Western European
WENRA
Nuclear Regulators Association

WENRA Handbook on Reactor Safety

Part 2: WENRA Statements and Safety Reference Levels

Version 2; December 2014

Content

Western European
WENRA
Nuclear Regulators Association

WENRA

STATEMENT on updated Safety Reference Levels

October 2014

27.10.2014

WENRA Statement

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WENRA Statement regarding the revision of the Safety Reference Levels for existing reactors taking into account the lessons learned from the TEPCO Fukushima Dai-ichi Nuclear Accident

October 2014

WENRA Statement

regarding the revision of the SRLs for existing reactors taking into account the lessons learned from the TEPCO Fukushima Dai-ichi Nuclear Accident

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One of the objectives of WENRA, as stated in its terms of reference, is to develop a harmonized approach to nuclear safety and radiation protection issues and their regulation in Europe. A significant contribution to this objective was the publication, in 2006, of a report on harmonization of reactor safety in WENRA countries. This report addressed the nuclear power plants in operation and it included "Safety Reference Levels" (SRLs), which reflected expected practices to be implemented in the WENRA countries. The SRLs were updated twice in 2007 and again in 2008.

The SRLs have been established for greater harmonization within WENRA countries raising the level of nuclear safety in Europe by their implementation in the national regulatory framework and in the nuclear power plants (NPPs). The emphasis of the SRLs has been on nuclear safety, primarily focussing on safety of the reactor core and spent fuel. The SRLs specifically exclude nuclear security and, with a few exceptions, radiation safety.

WENRA members are committed to continuous improvement of nuclear safety in their countries. Within this spirit WENRA emphasizes identifying the insights from the Fukushima Dai-ichi accident in March 2011 and operators improving NPP safety accordingly. For this purpose, WENRA mandated its Reactor Harmonization Working Group (RHWG) to review and revise the SRLs for existing reactors with the aim to integrate the lessons learned from the 2011 Fukushima Dai-ichi accident.

The SRLs that have been developed represent, in addition to good practices in WENRA countries, objectives for safety improvements to take account of the lessons learned from the Fukushima accident.

The national regulators make a commitment to improve and harmonize their national regulatory systems, by implementing the new SRLs until 2017 as a target date.

WENRA strives for openness and keeps all interested parties informed of the progress made in this work.

Western European
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Recommendation

15.08.2013

WENRA Recommendation in connection with flaw indications found in Belgian reactors

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WENRA
Summer 2013

01
Background

1. Doel 3 and Tihange 2 are Pressurised Water Reactors (PWR) owned and operated by Electrabel in Belgium. During summer 2012 Electrabel completed the 30-year in-service inspection of the reactor pressure vessel (RPV) at Doel 3. These inspections identified a large number of flaw indications in the vessel wall, located principally in the base material of the lower and upper vessel ring forgings. These nearly laminar indications are mostly planar in orientation and have been assessed by Electrabel as originating from, "hydrogen flaking", a metallurgical phenomenon that can occur during the steelmaking process.
2. FANC informed the WENRA members at the plenary meeting of 23-24 October 2012 on the state of the on-going investigations. It was decided that WENRA should decide on possible actions or recommendations as soon as the results of the Belgian investigations are available.
3. As soon as detailed and confirmed results were made available about the indications found at Doel 3, the Belgian nuclear safety authority (FANC) informed its counterparts in other countries. To increase transparency and cooperation between potentially interested countries and to benefit from external insights on the case, FANC decided in August 2012 to set up several national and international working groups. Three working groups were constituted of international experts made available by foreign nuclear safety authorities or related organizations and explored the following topics:
    a. Non-destructive testing techniques;
    b. Metallurgical origin of flaw indications;
    c. Structural mechanics and fracture mechanics.
4. The suggestions, observations and conclusions of national and international working groups were evaluated by FANC. Wherever appropriate and relevant, FANC decided to use this input in the formulation of their conclusions and specific requirements for the licensee.

02
Safety verification of European RPV's in the light of findings in Doel 3 and Tihange 2

5. Based on the reports from Belgium[1], some nuclear safety authorities have decided to request a safety verification of RPVs in their countries.
6. With the purpose of harmonizing the RPV related activities in Europe, WENRA recommends the nuclear safety authorities in Europe to take actions as outlined below.

[1] http://www.fanc.be/nl/page/dossier-pressure-vessel-doel-3-tihange-2/1488.aspx?LG=2

03
Recommendations to WENRA members

7. WENRA recommends the following two-step verification of materials quality and structural integrity of the RPV:

Step 1: Comprehensive review of manufacturing and inspection records

8. The national nuclear safety authorities shall request the licensees to review and compile a chronological and comprehensive documentation of all processes and steps of the manufacturing and controls of the RPV forgings and to evaluate this documentation with respect to the hydrogen flaking issue. This evaluation shall cover at least:

    a. All records and certificates of intermediate and final heat treatment, chemical analysis and impurity content requirements, material testing, and pre and in-service inspection results.

    b. Identification of any potential non-conformity or gap in the documentation.

    c. Records of the workshop and site inspections as required by the national nuclear safety regulations.

    d. Surveillance approach in the workshop and on site; and inspection findings if required by the national nuclear safety authority.

    e. The applied quality management system.

Step 2: Examination of the base material of the vessels

9. WENRA recommends performing additional non-destructive testing to reassess the quality of RPV forging base material of the vessels. The decision as to whether this should be undertaken rests with the national nuclear safety authorities and will depend upon the strength of the information presented from the Step 1 review. This decision should also take account of the results of in service inspections, if any, and national or foreign experience feedback. The national nuclear safety authorities shall agree with the licensee, or specify, the testing scope, volume and non-destructive test method.

10. The following information should be taken as guidelines for the national nuclear safety authorities regarding additional non-destructive testing. The additional testing may be carried out in connection with the regular in-service inspection of the RPV. The inspections should cover a representative volume of RPV forging base material in areas known to be potentially susceptible to hydrogen flaking. If these inspections reveal evidence of hydrogen flaking the inspections should be extended appropriately. For inspection of the RPV forging material, a method should be used which has been demonstrated to be sufficiently sensitive to detect hydrogen flaking.

Final considerations

11. Further measures are up to the national nuclear safety authority to decide upon. For example, extending the scope of analysis to other primary equipment (Steam Generators, Pressurizer).

12. The national nuclear safety authority should review the outcome of the work to address these recommendations.

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WESTERN EUROPEAN NUCLEAR
REGULATORS ASSOCIATION

RHWG
REACTOR HARMONISATION
WORKING GROUP

WGWD
WORKING GROUP ON WASTE
AND DECOMISSIONING

Western European
WENRA
Nuclear Regulators Association

WENRA

TERMS OF REFERENCE 2010

March 2010

26 March 2010

TERMS OF REFERENCE

OF THE

WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION

(WENRA)

1. We the Heads of Nuclear Regulatory Authorities (signatories) of European countries with nuclear power plants:

   - drawing from the experience already gained with WENRA and noting its achievements,
   - recognizing that the current regulatory challenges in Europe lead to envisage the activities of WENRA in a broader perspective,
   - re-affirming the need for increased co-operation between us, and
   - maintaining our independence,

   have again revised the previous Terms of Reference of the Western European Nuclear Regulators' Association (WENRA), which were signed on 4 February 1999 and revised on 14 March 2003.

2. With the general aim of improving nuclear safety, has the following objectives:

   - to build and maintain a network of chief nuclear safety regulators in Europe,
   - to promote exchange of experience and learning from each others best practices,
   - to develop a harmonized approach to nuclear safety and regulation, in particular within the European Union,
   - to discuss and, where appropriate, express its opinion on significant safety and regulatory issues.

3. Decisions in the name of WENRA are taken by consensus.

4. WENRA will keep the European Union Institutions informed about its activities, and is prepared to consider requests from these institutions for advice on nuclear safety and regulatory matters.

5. Heads of the regulatory authorities (or corresponding) in other European countries, which have expressed an interest, are invited as observers to WENRA. Observers have the right to express their opinion at the WENRA meetings but can not participate in the decision making. Observers may send suitably qualified participants to the working groups.

6. WENRA will develop and maintain, when appropriate, suitable relations with regulatory authorities from other countries as well as with international organisations.

7. WENRA will ensure appropriate opportunities for stakeholders to comment on its work.

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Czech Republic Dana DRÁBOVÁ	Finland Jukka LAAKSONEN
France André-Claude LACOSTE	Germany Gerald HENNENHÖFER
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WENRA

POLICY STATEMENT

December 2005

WENRA Policy Statement

We, the heads of the national Nuclear Safety Authorities, members of WENRA, commit ourselves to a continuous improvement of nuclear safety in our respective countries.

Nuclear safety and radiation protection are based on the principle of the prime responsibility of the operators. The role of national regulators is to ensure that this responsibility is fully secured, in compliance with the regulatory requirements.

In order to work together, we created the Western European Nuclear Regulators' Association (WENRA) with the following main objectives:

- *to build and maintain a network of chief nuclear safety regulators in Europe;*
- *to promote exchange of experience and learning from each other's best practices;*
- *to develop a harmonized approach to selected nuclear safety and radiation protection issues and their regulation, in particular within the European Union;*
- *to provide the European Union Institutions with an independent capability to examine nuclear safety and its regulation in Applicant Countries.*

In order to develop a harmonized approach, we are:

- *sharing our experience feedback and our vision;*
- *making efforts to further exchange of personnel, allowing an in-depth knowledge of working methods of each other;*
- *developing common reference safety levels in the fields of reactor safety, decommissioning safety, radioactive waste and spent fuel management facilities in order to benchmark our national practices.*

We recognise the IAEA standards form a good basis for the continuous improvement of national nuclear regulatory systems and nuclear safety.

The reference levels that we have developed represent good practices in our countries from which we can also seek to learn from each other to further improve nuclear safety and its regulation. Hence, we are committed:

- *by the year 2010 to improve and harmonise our nuclear regulatory systems, using as a minimum the reference levels;*
- *to influence the revision of the IAEA standards when appropriate;*
- *to regularly revise the reference levels when new knowledge and experience are available.*

We strive for openness and improvement of our work. For that purpose we will:

- *keep the European Nuclear Safety and Radiation Protection Bodies not belonging to WENRA, and the EU Institutions, informed of the progress made in our work;*
- *make our public reports available on the Internet (www.wenra.org);*
- *invite stakeholders to make comments and suggestions on these reports.*

Signed in Stockholm December 2005

J-P. Samain, Belgium	S. Tzotchev, Bulgaria
D. Drabova, Czech Republic	J. Laaksonen, Finland
A-C. Lacoste, France	W. Renneberg, Germany
I. Lux, Hungary	S. Giulianelli, Italy
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COMMON VIEWS ON THE SIGNIFICANCE OF NATIONAL RESPONSIBILITY FOR NUCLEAR SAFETY

July 2003

WESTERN NUCLEAR
REGULATORS ASSOCIATION

WENRA COMMON VIEWS
ON
THE SIGNIFICANCE OF NATIONAL RESPONSIBILITY FOR NUCLEAR SAFETY

July 2003

NATIONAL RESPONSIBILITY FOR NUCLEAR SAFETY REGULATION IS EFFECTIVE FOR ENSURING HIGH LEVEL OF SAFETY AND CONTINUOUS SAFETY IMPROVEMENTS

Principle of strong national regulations is one of the cornerstones of the International Convention on Nuclear Safety

All current and all new Member States of the EU operating nuclear power plants are Parties to the Convention on Nuclear Safety. Ratification of the Convention implies that they are legally committed to a high and internationally recognised level of safety in the nuclear power plants under their jurisdiction. The Parties to the Convention are further committed to a national responsibility for safety as well as independence of their regulatory bodies.

The principle of national responsibility and the requirements on regulation were included in the Convention on Nuclear Safety on the basis of positive experience in countries where strong national regulations have been emphasized since start up of the use of nuclear energy. In such countries the safety record is good and indicates high safety level.

Effective regulatory control requires in-depth knowledge of the facilities being regulated; this knowledge is only with the national regulators

Safety of a nuclear power plant is a complex technical and human issue that cannot be reached through mere conformity with rules and regulations and that cannot be measured with simple numerical values. Safety cannot be verified in a straightforward and objective manner. There are alternative means to achieve the same safety goal, and the final judgement on adequacy of safety level is always subjective.

All plants in operation to-day have been designed and constructed individually, and the relative safety importance of their parts cannot be judged from direct comparison with other plants. A necessary condition for a qualified safety

judgement is a thorough understanding of how various safety relevant factors are integrated to a whole. Assessment of nuclear safety therefore requires not only an in-depth knowledge on related
technical and physical issues but also a thorough familiarity with the details of each nuclear facility and the technology used. Furthermore, it is important to know the infrastructure and technical culture in which the plant is operated. Such knowledge and familiarity exists today with national regulators and their technical support organisations.

Achievement of the level of knowledge that is required for making credible safety judgements on foreign nuclear facilities is not possible without several years of work experience from those facilities. This is why there is a wide consensus documented in the Convention on Nuclear Safety that a conclusion on adequate nuclear safety can be drawn only by the national authorities.

Harmonised safety practices, that are followed in all countries of the world are needed

There is a general agreement among the EU institutions and the EU Member States that no new technical regulations and definitions should be introduced at the regional (EU) level. Instead, a respect of the IAEA nuclear safety standards has to be ensured. All European countries operating nuclear power plants are to day involved in the IAEA's nuclear safety standards work.

The IAEA standards are written with the aim to document best available safety practices, and the clearly stated objective of the standards programme is to enhance the level of nuclear safety worldwide. In this work each country can systematically benchmark its own situation to the international practice.

Safety must not be stagnant, but safety must be continuously improved

Some of the guidance given in the IAEA safety standards is of fundamental nature and needs to be taken as mandatory when issuing national regulations. Other IAEA standards are to be taken as commendable goals that are met by different countries and by different plants to varying degree. These goals are moving targets, and are therefore not fully achievable at any point of time. Their value is that they serve as a driving force for improvement and as a commendable goal for everybody. As such they are a sound support for the national regulators.

National responsibility for safety provides the fastest way to consolidate improved practices into safety regulations. Modifications are needed on one hand to implement the new principles written in the IAEA safety standards. In addition, national regulators must promptly react to new safety concerns that may be identified through operating experience or from research.

A possibility to make modifications to regulations separately in each country is the most efficient way to observe the principle generally adopted by European nuclear regulators: there has to be a continuous striving for enhanced level of nuclear safety.

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Well-established peer review mechanisms support the national regulators to adopt the best international practices

Notwithstanding the principle of national responsibility, it is necessary that national nuclear safety regulators interact and learn from each other. To day there are several peer review mechanisms for this purpose. Among these are the regular review meetings of the Convention on Nuclear Safety, the IAEA service called International Regulatory Review Teams (IRRT), and the nuclear safety harmonisation work agreed by the European national nuclear safety regulators.

The IRRT service is a well-established peer review mechanism between the IAEA Members States. The IRRT missions use the IAEA safety standards as the basis of their work, and thus build on worldwide consensus on good nuclear regulation. The scope of an IRRT mission is much wider than the scope proposed by the EC for similar European reviews, and also the number of man-days worked on an IRRT mission is higher by factor of 20-30. The experience from IRRT missions has shown that even with such large effort it has been difficult to understand all essential features in the work of the reviewed regulatory organisation, and therefore the need to fulfil recommendations of an IRRT mission are left to the discretion by the reviewed country.

Close regular co-operation has been established since 1999 between all European nuclear regulators. They have jointly formed an association called WENRA, and this association has already proven to be an effective instrument for assessing and enhancing the level of harmonisation of the safety regulations in its member states. In parallel with harmonisation of the regulations, the WENRA members are committed to develop their national practices so that the agreed reference safety level will truly be achieved in each country.

WENRA work of harmonisation is based on the IAEA safety standards, and thus provides feedback on the applicability of the IAEA standards. The joint positions of WENRA can thus be used also for improving the IAEA standards.

WENRA is willing to prepare regular reports on the status of nuclear safety in EU-countries and the achievements to higher harmonised safety levels. These reports are available for the European Institutions.

Nuclear safety is improved in a number of co-operation forums between regulators

The national nuclear regulators of the EU Member States have co-operated in a joint working group since the 1970's. This group and its subgroups have produced many valuable recommendations to the European and worldwide safety developments, and these are generally well implemented in the EU Member States.

Among other joint forums between the national regulators it is necessary to point out the CNRA and CSNI of the OECD Nuclear Energy Agency, and the numerous working groups under those Committees. These form a professional network between countries with advanced nuclear programmes. Since the 1970's this network has been used efficiently to benchmark and improve national safety practices.

Western European
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WENRA

REPORT

October 2000

WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION

GENERAL CONCLUSIONS OF WENRA

On nuclear safety in the candidate Countries to the European Union

We, Heads of the Nuclear Regulatory Authorities assembled in WENRA, considering the status achieved on nuclear safety in the candidate countries to the European Union and taking into account the results of the investigations of experts from WENRA and from French and German technical support organisations, come to the following conclusions:

BULGARIA

Status of the regulatory regime and regulatory body

At present, the regulatory regime is not in line with Western European practice because it does not provide sufficient independence to the regulatory body. The resources of the regulatory body are also insufficient to allow it to carry out its responsibilities.

Nuclear power plant safety status

Kozloduy units 1-4 (VVER-440/230)

Although improvements have been made, the Kozloduy 1-4 units have not reached an acceptable level of safety. Among others, a concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. Even if a solution could be found to this issue, significant time and effort would be required to achieve the necessary improvements to bring them up to equivalent Western European reactor standards. The Bulgarian Government has announced its decision to close down Kozloduy units 1-2 before 2003.

Kozloduy units 5-6 (VVER-1000/320)

If their modernisation programmes are carried out properly, the Kozloduy 5-6 units should reach a level of safety comparable to that of Western European reactors of the same vintage.

CZECH REPUBLIC

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in the Czech Republic are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

Dukovany units 1-4 (VVER-440/213)

Already in the early years of operation, improvements were implemented to remove safety deficiencies of the original design. An extensive modernisation programme has been established and it will allow Dukovany units 1-4 to reach a safety level comparable to that of Western European reactors of the same vintage. All issues, except the modernisation of the Instrumentation and Control systems, will be completed by 2004.

Temelin units 1-2 (VVER-1000/320)

The safety improvement programme for Temelin units 1-2 is the most comprehensive one ever applied to a VVER-1000 reactor. Standard Western practices were used to integrate Eastern and Western technologies and to deliver the corresponding authorisations. The on-going commissioning process has to confirm the integration of the different technologies. A few safety issues still need to be resolved. If these are resolved, Temelin units 1-2 should reach a safety level comparable to that of currently operating Western European reactors.

HUNGARY

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Hungary are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

Paks units 1-4 (VVER-440/213)

A major safety improvement programme has been implemented at Paks units 1-4, bringing these units to a safety level that is comparable to that of Western European reactors of the same vintage. An extensive modernisation of the Instrumentation and Control system is underway for further enhancement of safety.

LITHUANIA

Status of the regulatory regime and regulatory body

The legal and regulatory system has substantially developed over the past years. A licensing system is in place. However, further efforts are needed to reach a level comparable to Western European practice. In particular, the legal status of the plant need to be changed in such a way that operating organisation is given full responsibility and authority for the safety of the plant. The resources and technical support of the regulatory body need to be strengthened and its independence need to be maintained in the ongoing reorganisation of governmental institutions.

Nuclear power plant safety status

Ignalina units 1-2 (RBMK 1500)

The Ignalina units 1-2, although they have been much improved, cannot realistically reach a safety level comparable to that of Western European reactors of the same vintage. A decision has already been taken to shutdown unit 1 before 2005. The current financial situation of the plant needs to be improved in order not to delay the ongoing safety improvement programme.

ROMANIA

Status of the regulatory regime and regulatory body

Romania is taking the appropriate steps to establish a regulatory regime and regulatory body comparable with Western European practice. Further efforts are needed to ensure the necessary safety assessment capabilities, to develop the emergency response organisation within the regulatory body and to revise the pyramid of regulatory documents..Nuclear safety in EU candidate countries **- 7**

WENRA - October 2000

Nuclear power plant safety status

Cernavoda unit 1 (Candu 6)

The Candu 6 reactor of Cernavoda is similar to those in operation at Gentilly 2 and Point-Lepreau
in Canada. The main concern is with the financial situation of the plant: under the current situation, the plant management may have serious difficulties in ensuring and maintaining an adequate level of safety.

SLOVAKIA

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Slovakia are comparable with Western European practice. However, the human and financial resources of the regulatory body need to be further improved in order to provide reasonable work conditions for the staff.

Nuclear power plant safety status

Bohunice V1 (VVER-440/230)
A major upgrade programme is nearing completion, which has made significant improvements to reactor safety. A concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. If a solution can be found to this issue, the plant should reach a safety level comparable to that of Western European reactors of the same vintage. The Slovak Government has announced its decision to close down these units in 2006 and 2008.
Bohunice V2 (VVER-440/213)
Since 1990, significant improvements have been implemented at Bohunice V2 (units 3-4). Once the on-going upgrading measures have been implemented, i.e. around 2002, the safety level of these units is expected to be comparable to that of Western European reactors of the same vintage.
Mochovce units 1-2 (VVER-440/213)
Compared to earlier reactors of the same type (VVER 440-213), the Mochovce units 1-2 included several modifications already at the design stage. Although some residual work is still needed to confirm all parts of the safety analysis, the safety level of the Mochovce units 1-2 is comparable to that of nuclear power plants being operated in Western Europe.

SLOVENIA

Status of the regulatory regime and regulatory body

In order to be fully comparable with Western practice, the nuclear legislation needs to be revised, addressing the identified deficiencies. The regulatory body has evolved and operates in general accordance with Western practice and methodologies, however the budget and financial situation need to be improved in order to increase its independent safety assessment capability.

Nuclear power plant safety status

Kr ko (Western PWR)
The Kr ko plant is a Western design pressurised water reactor and its safety level is comparable with that of nuclear power plants in operation in Western European countries. A large modernisation programme has been recently completed. The safety implications of the long-term.
plant ownership need to be assessed. In addition, the evaluation of a few technical issues needs to be finalised.

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REPORT

January 2000

WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION

Nuclear safety in EU candidate countries

October 2000

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FOREWORD

The Western European Nuclear Regulators' Association (WENRA) is the association of the Heads of nuclear regulatory authorities of Western European countries with nuclear power plants, namely Belgium, Finland, France, Germany, Italy, the Netherlands, Spain, Sweden, Switzerland [(*)] and the United Kingdom. The association has the following objectives:

- To develop a common approach to nuclear safety and regulation, in particular within the European Union,
- To provide the European Union with an independent capability to examine nuclear safety and regulation in candidate countries,
- To evaluate and achieve a common approach to nuclear safety and regulatory issues which arise.

Nuclear safety in the candidate countries to the European Union is a major issue that needs to be addressed in the framework of the enlargement process. Therefore WENRA members considered it was their duty to offer their technical assistance to their Governments and the European Union Institutions. They decided to express their collective opinion on nuclear safety in those candidate countries having at least one nuclear power plant: Bulgaria, the Czech Republic, Hungary, Lithuania, Romania, Slovakia and Slovenia.

The report is structured as follows:

- A foreword including background information, structure of the report and the methodology used,
- General conclusions of WENRA members reflecting their collective opinion,
- For each candidate country, an executive summary, a chapter on the status of the regulatory regime and regulatory body, and a chapter on the nuclear power plant safety status.

Two annexes are added to address the generic safety characteristics and safety issues for RBMK and VVER plants. The report does not cover radiation protection and decommissioning issues, while safety aspects of spent fuel and radioactive waste management are only covered as regards on-site provisions.

In order to produce this report, WENRA used different means:

- For the chapters on the regulatory regimes and regulatory bodies, experts from WENRA did the work,
- For the chapters on nuclear power plant safety status, experts from WENRA and from French and German technical support organisations did the work,
- Taking into account the contents of these chapters, WENRA has formulated its general conclusions in this report.

WENRA's methodology for reaching the collective opinion expressed in the general conclusions

[(*)] The Swiss member of WENRA did not take part in establishing the present report has been to compare the current situation in the candidate countries to that in Western European countries using a common format which is reflected in the structure of the chapters. All major safety issues identified in past international co-operation have been considered. For each candidate country, a comparison was made with the current Western European practices and, whenever appropriate, discrepancies or deficiencies were clearly identified.

WENRA has not made a detailed safety assessment of the different nuclear power plants. Nuclear safety is a national responsibility and it belongs to the regulatory body of the various candidate countries to regulate the safety of all nuclear installations on their national territory, in line with the national legislative and regulatory framework.

WENRA's collective opinion on the regulatory systems is based on generic preconditions for an independent and strong regulatory regime such as a comprehensive nuclear legislation, the existence of an adequate licensing system, appropriate resources and technical support. WENRA's collective opinion on nuclear power plant safety is based on widely applied standards in Western European countries for the defence-in-depth and associated barriers. Quantitative comparisons of probabilistic safety assessments have not been used as the available results are of different depth and quality.

A first version of this report was issued in March 1999. It was solely based on the direct evidence WENRA had gathered through the different activities of its members (participation in multilateral assistance programmes, and in particular the Phare programmes and the IAEA extra-budgetary programme, and in bilateral contacts). In particular, information necessary to formulate an opinion on the regulatory regimes and the regulatory bodies were in many cases derived from the regulatory assistance projects of the RAMG implemented under the Phare programme. With regards to the safety status of nuclear power plants, WENRA had to recognise that in some cases the direct information was not sufficient to formulate an opinion.

For the present version, WENRA took the appropriate steps to collect the necessary information. In addition to the direct evidence already available, supplementary information was gathered through meetings with the candidate countries' regulatory bodies and plant operators. In particular, an ad-hoc Task Force was established to gather and evaluate additional information on VVER-440/230 reactors.

GENERAL CONCLUSIONS OF WENRA

ON NUCLEAR SAFETY IN CANDIDATE COUNTRIES TO THE EUROPEAN UNION

We, Heads of the Nuclear Regulatory Authorities assembled in WENRA, considering the status achieved on nuclear safety in the candidate countries to the European Union and taking into account the results of the investigations of experts from WENRA and from French and German technical support organisations, come to the following conclusions:

BULGARIA

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At present, the regulatory regime is not in line with Western European practice because it does not provide sufficient independence to the regulatory body. The resources of the regulatory body are also insufficient to allow it to carry out its responsibilities.

Nuclear power plant safety status

<u>Kozloduy units 1-4 (VVER-440/230)</u>
Although improvements have been made, the Kozloduy 1-4 units have not reached an acceptable level of safety. Among others, a concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. Even if a solution could be found to this issue, significant time and effort would be required to achieve the necessary improvements to bring them up to equivalent Western European reactor standards. The Bulgarian Government has announced its decision to close down Kozloduy units 1-2 before 2003.

<u>Kozloduy units 5-6 (VVER-1000/320)</u>
If their modernisation programmes are carried out properly, the Kozloduy 5-6 units should reach a level of safety comparable to that of Western European reactors of the same vintage.

CZECH REPUBLIC

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in the Czech Republic are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

<u>Dukovany units 1-4 (VVER-440/213)</u>
Already in the early years of operation, improvements were implemented to remove safety deficiencies of the original design. An extensive modernisation programme has been established and it will allow Dukovany units 1-4 to reach a safety level comparable to that of Western European reactors of the same vintage. All issues, except the modernisation of the Instrumentation and Control systems, will be completed by 2004.

Temelin units 1-2 (VVER-1000/320)
The safety improvement programme for Temelin units 1-2 is the most comprehensive one ever applied to a VVER-1000 reactor. Standard Western practices were used to integrate Eastern and Western technologies and to deliver the corresponding authorisations. The on-going commissioning process has to confirm the integration of the different technologies. A few safety issues still need to be resolved. If these are resolved, Temelin units 1-2 should reach a safety level comparable to that of currently operating Western European reactors.

HUNGARY

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Hungary are comparable with Western European practice. A well-defined licensing process according to Western practice is in place.

Nuclear power plant safety status

Paks units 1-4 (VVER-440/213)
A major safety improvement programme has been implemented at Paks units 1-4, bringing these units to a safety level that is comparable to that of Western European reactors of the same vintage. An extensive modernisation of the Instrumentation and Control system is underway for further enhancement of safety.

LITHUANIA

Status of the regulatory regime and regulatory body

The legal and regulatory system has substantially developed over the past years. A licensing system is in place. However, further efforts are needed to reach a level comparable to Western European practice. In particular, the legal status of the plant need to be changed in such a way that operating organisation is given full responsibility and authority for the safety of the plant. The resources and technical support of the regulatory body need to be strengthened and its independence need to be maintained in the ongoing reorganisation of governmental institutions.

Nuclear power plant safety status

Ignalina units 1-2 (RBMK 1500)
The Ignalina units 1-2, although they have been much improved, cannot realistically reach a safety level comparable to that of Western European reactors of the same vintage. A decision has already been taken to shutdown unit 1 before 2005. The current financial situation of the plant needs to be improved in order not to delay the ongoing safety improvement programme.

ROMANIA

Status of the regulatory regime and regulatory body

Romania is taking the appropriate steps to establish a regulatory regime and regulatory body comparable with Western European practice. Further efforts are needed to ensure the necessary safety assessment capabilities, to develop the emergency response organisation within the regulatory body and to revise the pyramid of regulatory documents.

Nuclear power plant safety status

Cernavoda unit 1 (Candu 6)
The Candu 6 reactor of Cernavoda is similar to those in operation at Gentilly 2 and Point-Lepreau in Canada. The main concern is with the financial situation of the plant: under the current situation, the plant management may have serious difficulties in ensuring and maintaining an adequate level of safety.

SLOVAKIA

Status of the regulatory regime and regulatory body

The regulatory regime and regulatory body in Slovakia are comparable with Western European practice. However, the human and financial resources of the regulatory body need to be further improved in order to provide reasonable work conditions for the staff.

Nuclear power plant safety status

Bohunice V1 (VVER-440/230)
A major upgrade programme is nearing completion, which has made significant improvements to reactor safety. A concern remains about the ability of the confinement system to cope with the failure of the large primary circuit pipework. If a solution can be found to this issue, the plant should reach a safety level comparable to that of Western European reactors of the same vintage. The Slovak Government has announced its decision to close down these units in 2006 and 2008.

Bohunice V2 (VVER-440/213)
Since 1990, significant improvements have been implemented at Bohunice V2 (units 3-4). Once the on-going upgrading measures have been implemented, i.e. around 2002, the safety level of these units is expected to be comparable to that of Western European reactors of the same vintage.

Mochovce units 1-2 (VVER-440/213)
Compared to earlier reactors of the same type (VVER 440-213), the Mochovce units 1-2 included several modifications already at the design stage. Although some residual work is still needed to confirm all parts of the safety analysis, the safety level of the Mochovce units 1-2 is comparable to that of nuclear power plants being operated in Western Europe.

SLOVENIA

Status of the regulatory regime and regulatory body

In order to be fully comparable with Western practice, the nuclear legislation needs to be revised, addressing the identified deficiencies. The regulatory body has evolved and operates in general accordance with Western practice and methodologies, however the budget and financial situation need to be improved in order to increase its independent safety assessment capability.

Nuclear power plant safety status

Krško (Western PWR)
The Krško plant is a Western design pressurised water reactor and its safety level is comparable with that of nuclear power plants in operation in Western European countries. A large modernisation programme has been recently completed. The safety implications of the long-term

plant ownership need to be assessed. In addition, the evaluation of a few technical issues needs to be finalised.

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A.C. LACOSTE Director Nuclear Installation Safety Directorate (DSIN) France	W. RENNEBERG Director General for Nuclear Safety Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU) Germany
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L. WILLIAMS HM Chief Inspector Nuclear Installations (HSE) United Kingdom	

REPORT

Executive summaries

BULGARIA

Status of the regulatory regime and regulatory body

Since the early 1990s, there have been significant improvements in the legislative basis and in the capabilities of the nuclear regulatory body (the Committee on the Use of Atomic Energy for Peaceful Purposes - CUAEPP).

However, much remains to be done to bring the regulatory regime up to Western European standards. The Bulgarian Governments needs to enact legislation that will make explicit the independence of the CUAEPP from bodies concerned with the promotion of nuclear power. The government needs to provide adequate funding to the CUAEPP to enable the recruitment and retention of adequate numbers of qualified staff. Funding is also needed to enable the development new technical support facilities for the CUAEPP. Resources need to be committed to the drafting and introduction of necessary new and revised legislation.

Nuclear power plant safety status

There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased. However, the lack of Safety Analysis Report of any Bulgarian nuclear power plant is a serious shortcoming for judging the safety. Therefore to confirm the improvements implemented a consistent safety case has to be established and it has to be reviewed by the CUAEPP.

Kozloduy 1-4
Despite the significant safety improvements already achieved considering the present safety status of the plant, there are still some major safety issues which are closely linked to the original basic design of the VVER-440/230 reactors and which are difficult to be removed, such as the limited confinement function and capability and the vulnerability against common cause failures. The work at Kozloduy is at least three years behind that at Bohunice and consequently safety improvement is not as far advanced. At present in view of the large amount of work required to be carried out it is difficult to have a final judgement on the adequacy and feasibility of all measures foreseen. It seems, however that financial provisions for continued safety improvements are inadequate. For Kozloduy 1 and 2 the implementation of relevant measures cannot be expected taking into account the announced closure dates.

Kozloduy 5-6
An extensive programme for further upgrading of these units with assured financing has been reviewed by Western TSOs and is at an early stage of implementation. Safety assessments done by Western TSOs for similar plants indicate that with the completion of the planned safety upgrades, it could be possible to achieve a level of safety for units 5 and 6 which is in line with international recognised safety practices.

CZECH REPUBLIC

Status of the regulatory regime and regulatory body

The nuclear legislative framework in the Czech Republic is comparable with Western European practice. It is considered that the SÚJB has a status comparable to that of Western European regulatory bodies. The SÚJB has developed a series of regulatory practices, including a well-defined licensing process, which compare favourably with those of Western European nuclear regulators.

Further improvements could arise from the following suggestions. It is recommended that the Government of the Czech Republic consider giving high priority to the implementation of the new Act on emergency preparedness and planning. The SÚJB should be asked to make proposals in view of removing too detailed requirements from the high level documents of the regulatory pyramid. Also, the contracting rules of the SÚJB need to be adapted so that it can obtain, when appropriate, the necessary high quality technical support on a long term basis.

Nuclear power plant safety status

<u>Dukovany NPP</u>
In the early years of operation, modifications were carried out to remove safety deficiencies in the original design. An extensive modernisation programme, called MORAVA, will be implemented by 2004 with the exception of I&C replacement.

The safety culture appears to be adequate. Safety assessments and verification documents, e.g. periodic safety reviews, are conducted in a way which is comparable to Western practice.

After full implementation of the modernisation programme it is expected that Dukovany NPP will achieve a safety level comparable to that of NPPs of the same vintage operating in Western Europe.

<u>Temelin NPP</u>
The safety improvement programme for Temelin NPP is the most comprehensive which has been applied to a VVER-1000/320 plant.

International co-operation has had a considerable influence on the plant's safety improvements (design, operation, safety approvals), and on the development of safety culture.

The combination of Eastern and Western technologies was successfully managed. Interfaces between the different technologies were considered throughout the modernisation programme and a standard Western practice was used to combine Eastern and Western technologies. The commissioning process will need to confirm the integration of the different technologies.

Some safety issues still need further clarification but if these issues are resolved, Temelin NPP will achieve a safety level that is comparable to that of operating Western PWRs.

HUNGARY

Status of the regulatory regime and regulatory body

The Hungarian approach to the licensing, regulation and control of nuclear facilities has developed strongly in the last ten years. A proper licensing process is in place, legislation and regulations are up-to-date, and the Hungarian regulatory practices are comparable with those of Western European countries.

However, there are some issues that need further consideration by the Hungarian Government. These are:

- The fact that the Minister of Energy Affairs is also the HAEC President creates an apparent conflict of interest, even though the formal mandate of HAEC President precludes this,
- The number of different authorities with direct responsibilities in the regulation of nuclear facilities increases the risk that important issues may be overlooked and reduces the efficiency of the regulatory work.

The NSD needs to continue its efforts to develop the inspection approach towards process oriented comprehensive team inspections.

Nuclear power plant safety status

The basic technical structure of Paks NPP is good from the safety point of view and the key safety systems are comparable to Western plants of the same vintage. No major shortcomings in the present safety systems have been identified in any of the several, independent, in-depth assessments done so far. Also the performance of the bubbler condenser containment in the case of large break LOCA has been verified in full-scope tests. There is still need for detailed analysis of the experimental results and for complementary tests of other design basis accidents (steam line break and small LOCAs). Paks containment structures provide adequate protection against design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU. However their leak rates are somewhat higher than those that are typical of Western European reactor containments.

Operational safety aspects are generally comparable to Western plants of the same vintage. However, management changes related to the political changes in the Government cause some concern. Periodic safety reviews are conducted in line with Western practices and have already led to an increase in safety.

It is expected that after the implementation of safety improvements already scheduled, the plant will reach a level of safety that compares favourably with plants of the same vintage in Western Europe.

LITHUANIA

Status of the regulatory regime and regulatory body

The legal and regulatory system has developed substantially over the last years. A licensing system is in place and the regulatory body VATESI has developed its approaches to safety assessment and inspection. Further efforts are needed, however, in order to be comparable with Western European practice.

The Lithuanian government needs to consider the legal status of Ignalina NPP, in order to give the operating organisation the full responsibility and authority to handle all financial and other management issues and thus to make the organisation able to take the full responsibility for safety. The legal obligation of VATESI to formally license suppliers needs to be changed, given a reasonable transition period. The imposed reduction of resources to VATESI, in terms of budget and staff, needs to be compensated as soon as possible and the resources successively strengthened in order for VATESI to handle all normal regulatory tasks and to contract the necessary technical support. In the reorganisation under way, of governmental institutions reporting directly to the Prime Minister, special attention needs to be given the independence of VATESI.

VATESI needs to give high priority to the development of the internal Quality Management system and take the final steps in separating the roles of the regulatory body and the operator in all supervisory activities.

Nuclear power plant safety status

The two units of Ignalina NPP (INPP) belong to the more advanced and improved design generation of RBMK reactors. In addition, the original design has been considerably improved through different safety improvement programmes. Most of the generic safety concerns with RBMK reactors have been satisfactorily addressed. More measures will be implemented, for instance the installation of a new diversified and independent shut down system at unit 2. However, weaknesses remain with respect to the last barrier for protection of the environment, especially in case of a severe accident. The weaknesses have to do with a less robust design of the confinement of the INPP reactors as compared with Western light water reactors. It is not realistic to make the INPP confinement system comparable. Consequently, regarding mitigation of accidents, a safety level comparable to light water reactors of the same vintage in operation in Western Europe will not be reached at Ignalina NPP. Therefore special attention needs to be given the prevention of accidents during the remaining operating time, including the need to ensure a high level of operational safety.

The financial situation of INPP needs to be much improved in order to cover all operational expenses as well as implementing the safety improvement measures considered necessary for the remaining operating time. Issues relating to safety culture need a stronger implementation. The symptom based emergency operating procedures need to be finalised and implemented without further delay. Due to the decision on decommissioning of unit 1, special attention needs to be given to keep a sufficient number of technical specialists, as well as maintaining the motivation of the staff, for the remaining operating time of both reactors.

ROMANIA

Status of the regulatory regime and regulatory body

Romania is taking appropriate steps to establish a regulatory regime and a regulatory body comparable with Western European practice. Roles, duties and responsibilities of organisations involved in nuclear safety are in line with those assigned to similar organisations in Western Europe. The independence of the regulatory body from the organisations involved in the use and promotion of nuclear energy is fully established by the law and is sufficiently reflected in the practice. The regulatory regime and the regulatory body have both improved during the licensing process of Cernavoda NPP.

However some improvements are necessary to reach a situation comparable with the practice in Western European countries:

- The independent assessment capability, the inspection practice and the technical support of CNCAN need to be strengthened. The salaries at CNCAN need to be further improved to preserve suitably qualified staff. Adequate resources need to be assigned to set up and implement a training programme for new staff. Existing agreement with the Canadian nuclear safety authority needs to be more effectively used for training purposes and for seeking advice on regulatory issues specific of the CANDU technology. A strategic plan could support the assignment of existing limited resources to higher priority needs,
- National organisations that would have a role in a nuclear emergency need to make their emergency procedures and lines of communication more effective. In addition, CNCAN needs to further develop its competence and staff numbers in this area and establish an emergency response centre,
- The responsibility for auditing and approving vendors and suppliers should rest with the operating organisation and not with the regulatory body.

Nuclear power plant safety status

Romania has only one NPP into operation. It is a CANDU 6 reactor similar to those in operation at Gentilly 2 and Point Lepreau in Canada. The plant was constructed and commissioned under the responsibility of a Western Consortium (AECL, Ansaldo). The Cernavoda plant managers and operators have a professional attitude and have assimilated a western safety approach and culture.

It is important that the Romanian Government ensures that the current financial problems of the utility do not affect the ability of the management to maintain an adequate level of safety at the plant. Western support, especially from Canadian experts, should be made available when it is needed in the future.

Based on available information it is apparent that additional assessments are needed to confirm design safety margins against seismic events and the adequacy of fire protection. Also, the resolution of specific safety issues for similar plants that have been addressed or are currently under discussion in Canada need to be noted and incorporated where necessary into an improvement programme. The current high level of qualification and safety culture of the plant managers needs to be preserved in the longer tem. The plant management safety culture should be extended to all plant personnel and to the necessary service and support interfaces existing in the country. There is finally a need for improvement in some areas of plant operation such as training, emergency preparedness and accident management.

SLOVAKIA

Status of the regulatory regime and regulatory body

The nuclear legislative framework in Slovakia is in line with Western European practice. The ÚJD has made significant progress over the recent years and has taken the appropriate steps to develop a series of regulatory practices comparable with those of Western European nuclear regulators. It is considered that, in general, the ÚJD status is comparable to that of regulatory bodies in Western European countries. On-going developments will improve its effectiveness.

It is recommended that the government of Slovakia consider the following suggestions. The ÚJD financial resources need to be further increased, in particular but not only, to maintain the independent assessment capability which was initiated under Swiss assistance. In order to retain highly qualified staff, the salaries at the ÚJD need to be made comparable with those of the operator's staff. It is suggested that the government give a high priority to the adoption of the national emergency plan. Also, the Atomic Act should be amended to remove some duties of the ÚJD that are not directly dealing with nuclear safety.

Finally, it is recommended that the ÚJD pay particular attention to ensure a clear separation between the technical support it receives and that provided to an operator.

Nuclear power plant safety status

The safety of Slovakian nuclear power plants has been improved since the early 1990´s in a determined manner with a strong national commitment, and significant investments have been made in technical upgrades. Guidance received from the IAEA has been used efficiently. Operational practices at all Slovakian nuclear power plants are consistent with those in Western Europe.

The following conclusions can be made:

<u>Bohunice V1 (units 1-2)</u>
The revised design requirements provide a coherent target for safety improvement of the plant. The utility has made significant progress towards establishing a new design base and implementing the relevant measures. Some work remains to be done but no technical obstacles in completing it are foreseen. It will be completed in 2000.

If a solution can be found to the concern related to the confinement ability to cope with the double ended guillotine break LOCA, the safety level of these units is expected to be comparable with that of units of the same vintage in Western European Countries.

<u>Bohunice V2 (units 3-4)</u>
Since 1990, significant improvements have been implemented at Bohunice V2. However, in order to achieve adequate reliability of safety systems in all operating situations, an extensive modernisation programme is planned for implementation between 1999-2006, with the major upgrades relating to safety being completed by 2002.

The safety of Bohunice V2 units seems generally adequate. Once the ongoing safety upgrades have been implemented (by about year 2002), the safety level of these units is expected to be comparable with that of units of the same vintage in Western European countries.

Mochovce (units 1-2)

Compared to their VVER-440/213 predecessors, units 1 and 2 of Mochovce included several modifications during the design phase. The most important of these are the use of higher quality equipment and the improvement of systems used in accident situations. However, some design weaknesses remained, and a dedicated nuclear safety improvement programme was developed for the Mochovce NPP in 1995. This programme, which is almost complete, was reviewed by Western European Technical Safety Organisations.

Although some residual work (e.g. bubbler condenser qualification, Mochovce site seismicity characterisation) is still needed to confirm all parts of safety analysis, the safety level of Mochovce units is comparable to that of the nuclear power plants being operated in Western Europe.

SLOVENIA

Status of the regulatory regime and regulatory body

The Slovenian Nuclear Safety Administration (SNSA) operates, in general, according to Western practice and methodologies. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, there are some issues that need to be addressed.

It is recommended that the Government of the Republic of Slovenia addresses the fact that the existing legislation on nuclear and radiation safety is not fully in line with current Western European practice, and its review needs to be completed. In addition, the lack of a final resolution of issues related to shared ownership of Krško NPP may affect the plant's long term financial situation, and have an impact on safety. Furthermore, the legal and financial situation of SNSA needs to be improved in order to increase its independent safety assessment capability. Finally, the national response to nuclear and radiological emergencies needs to be improved by implementing an integrated national emergency plan, paying special attention to the interface with the Croatian authorities. The SNSA, on its side, needs to develop further its own technical capabilities in order to be able to make better independent decisions, and needs to continue defining its regulatory requirements to allow it to make the licensing decisions.

Nuclear power plant safety status

Slovenia has one nuclear power plant located in Krško. The design of the Krško NPP is similar to other Westinghouse PWRs of the same type operating in the USA, Belgium, Switzerland, Korea and Brazil. The safety of the Krško NPP is comparable to that of nuclear power plants of the same vintage into operation in Western Europe. The NPP has had a continuous backfitting and upgrading programme and a large modernisation programme, including the replacement of steam generators and a full scope simulator. The site organisation and the operational safety practice are similar to those in Western Europe.

For the future the following issues need to be addressed. The implications on safety of the ownership for the long term and the upcoming privatisation process of the energy sector need to be carefully assessed. In addition, efforts to strengthen the engineering capability of the utility need to be continued, including resources to ensure the necessary technical support from foreign organisations. Closer contacts with Western European utilities would also be beneficial. Finally, the evaluation of a few issues, like the seismic characterisation of the site and the onsite storage of spent fuel like need to be finalised and further attention is deemed necessary to the performance of a periodic safety review.

Detailed chapters

BULGARIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. The primary legislation for nuclear safety, the Act on the Use of Atomic Energy for Peaceful Purposes, was enacted in 1985 and amended in 1995 and 1998. Enforcing regulations, which give interpretation and meaning to the early primary legislation came into force in 1985 but, due to shortage of resources, there has been slow progress in making revisions to reflect the 1995 and 1998 amendments. The Act gives responsibility for licensing and regulation to the Committee on the Safe Use of Atomic Energy for Peaceful Purposes (CUAEPP).

2. A Programme on the Development of a Comprehensive Legislative Framework on Safety of Spent Nuclear Fuel and Radioactive Waste Management was adopted by the Council of Ministers in December 1999.

3. A three-year programme for the development and revision of much of the present regulatory documentation was agreed in 1998 by the nuclear regulator and relevant Ministries. This will lead eventually to greater consistency and facilitate the adoption of a less prescriptive regulatory approach. However, the high workload of the regulatory body means that this programme is already behind schedule and is likely to be delayed even further. In 1999 a new Act setting out further ambitious amendments to the earlier Act was developed by the CUAEPP in co-operation with other Ministries. However, this was rejected by the Council of Ministers in February 2000, which will delay improvements in the legislative basis by at least one year. The proposed amendments aim to bring about the harmonisation of the Bulgarian and West European legislation on the safety of nuclear facilities and in the field of accounting and control of nuclear material leading to less prescriptive Bulgarian nuclear legislation. A new Act on the safety in the use of Nuclear Energy is under development by a joint working group under the leadership of the CUAEPP. The new Act will cover in detail the management of radioactive waste, spent nuclear fuel and decommissioning of nuclear facilities. It will also establish a stable legal mechanism for the financing of the regulatory body.

4. Existing legislation adequately defines the legal obligations of the Operator (Kozloduy NPP) giving it responsibility for the safe control of the plant and for civil liabilities under the Vienna Convention. The primary legislation also requires the operating company to make payments into funds for dealing with radioactive waste and for decommissioning. The regulations to implement this came into force only in 1999 and the first contributions to this fund were made in the same year. Kozloduy NPP is wholly owned by the State.

5. All the key international conventions related to nuclear safety have been ratified and incorporated into national legislation.

Status of the regulatory body and technical support infrastructure

6. The current legislation places a dual role on the CUAEPP. First as a State Body with membership from organisations concerned with the promotion of nuclear power and the

operation of the power stations, and secondly as a legal entity charged with regulation of safety. This implies a lack of independence of the CUAEPP as a safety regulator. The Council of Ministers decided in April 1999 that the CUAEPP should be replaced by an Agency of the Government, the State Atomic Energy Agency (SAEA). This would ensure regulatory independence from those organisations promoting nuclear energy and would give SAEA sole responsibility for regulating the nuclear facilities and the storage and transport of nuclear material. However, the legislation to implement this was rejected in February 2000.

7. Funding for the CUAEPP comes from the State Budget and is controlled by the Ministry of Finance. It is currently inadequate. Budget restrictions imposed in 1996 reduced the CUAEPP staff by about 25% to 77. Of these, 50 posts were allocated to the Inspectorate on the Safe Use of Atomic Energy (ISUAE), the enforcement and inspection division of the CUAEPP. CUAEPP salaries are still approximately 20% of those in the nuclear industry, and this makes recruitment and retention of well qualified staff difficult. Currently, there are not enough staff to adequately carry out all necessary safety assessment and site inspection duties and the low salaries make the CUAEPP vulnerable to the loss of more experienced personnel. The training for inspectors and succession planning will need to be improved when resources are available. A Council of Ministers decision in April 1999 approved step by step increase of the CUAEPP personnel to 88 in 1999, 102 (currently 80) in 2000 and a figure of 110 is under negotiation for 2001. Proposed legislation would fix the regulator's salaries at a minimum 80% of the equivalent industry level but this is not yet in place. Consequently the CUAEPP continues to lose staff.

8. In the past, frequent changes in senior management positions put additional strain on the CUAEPP. However, it is now benefiting from greater managerial stability, particularly with respect to the position of the Chairman. This is assisting the CUAEPP to plan and implement the improvement process.

9. Existing legislation gives the CUAEPP some enforcement powers. Penalties for contravention of regulations are defined in the primary legislation. Enforcement relies heavily on fines. The CUAEPP also has the responsibility to authorise suppliers of equipment to the licensees.

10. The CUAEPP currently has limited resources to perform technical evaluations and relies to a considerable extent on external support. In the last few years local technical support organisations have developed in number and expertise. However, there are still only a limited number in the country, which means they are sometimes contracted to work for both regulator and the utility. Following the enactment of the proposed new national legislation, the CUAEPP intends to have a permanent body dedicated to the technical support of the regulatory authority. There is currently a need for continuing external assistance and this need for support is unlikely to change in the near future.

11. The funds available to the CUAEPP for nuclear safety research and support are provided by charges levied on the licensees. Compared with equivalent funds typically available to Western regulators, this is at a fairly low level.

12. The status of the regulatory body is not yet comparable with its Western European counterparts. However, if political commitment would be achieved and resources made available, well-prepared plans exist to transform the CUAEPP.

Status of regulatory activities

13. An internally generated improvement plan issued by the CUAEPP in 1998 set out an ambitious programme for codifying the CUAEPP's nuclear safety and licensing requirements. This will gradually replace the prescriptive legacy of the former Soviet Union and bring the Bulgarian regulator in line with a Western European approach. The aim is to create a strong and independent regulatory body with sufficient funding to carry out the full range of regulatory activities, including the making of regulations, site inspection, assessment and enforcing the utility to implementation of periodic safety reviews and draw up safety analysis reports as a basis for licensing. Some good progress has already been made, but unfortunately, because the CUAEPP has insufficient resources, implementation of the plan is already falling behind schedule. On the positive side, the CUAEPP is making much better use of its site inspectors by adopting a Western approach in which the licensee carries out routine, qualification inspections of pressure parts and lifting equipment, under a general supervision by the CUAEPP. The CUAEPP management has recognised the importance of introducing an internal quality management system and it is making progress in the development of a manual and documents to support the regulatory work.

14. In general, the CUAEPP is staffed by technically competent personnel. However, since 1992, due to the limited resources, the CUAEPP has had to rely on external independent technical assessments in carrying out the licensing of plant modifications and improvements. Much of this has been provided under assistance projects funded by the EC, the IAEA and bilateral programmes. In the absence of significant increases in CUAEPP resources, such assistance will continue to be needed, at least until the completion of the modernisation of units 5-6.

15. Regulatory decisions on the future upgrading of the Kozloduy units 1-4 need to be taken against a clear licensing plan which has yet to be fully established and implemented. This will require the development of the relationship between the utility and a competent and fully recognised regulator. At present the safety justification is expected to be completed in 2001. The CUAEPP needs to give clear guidance on the compliance targets with consistency between initiating event classification, analysis assumptions and acceptance criteria.

Emergency preparedness on governmental side

16. In the past there were too many regulatory documents, produced over a number of years, to provide for an effective, consistent emergency response. A new Regulation for Emergency Planning and Preparedness for Actions in Case of a Radiation Accident was approved by a Decision of the Council of Ministers in March 1999. This regulation defines the responsibilities for planning, advising and decision making in the case of a nuclear emergency. In October 1999 the CUAEPP started to up-date the National Emergency Plan for Action in Case of the Nuclear Accident at Kozloduy NPP. This is scheduled for completion at the end of 2000. The role of the CUAEPP will continue to be to monitor the situation and to give advice to the central committee which takes the decisions. The CUAEPP has received international assistance in the development of its own Emergency Preparedness Manual. The CUAEPP has completed the modernisation of its Emergency Response Centre (ERC) with new offices, a diesel-generator for emergency power supply and new telephone and computer systems. Bulgaria is a participant in the IAEA Regional Assistance Project to promote harmonisation of emergency planning in Central and Eastern Europe.

17. Currently, there are annual communications-only exercises involving all relevant national authorities, and there is a site emergency exercise each year. But the new emergency planning regulation requires a full national nuclear emergency exercise every 5 years. A full national emergency exercise will be organised following the bringing into operation of the upgraded ERC. Bulgaria has participated in the last three INEX-2 international exercises organised by the OECD.

Conclusions

18. There have been significant improvements in legislation, organisation and operation of the CUAEPP. However, many of the weaknesses identified previously still remain. Lack of progress in several areas is no doubt partly due to the severe economic situation facing the country. Low wages, combined with a high workload and poor working conditions, have a negative effect on staff morale and the loss of further valuable staff is likely.

19. In order to reach Western European standards, it is recommended that the Government of Bulgaria considers the following issues, several of which were addressed in the draft Act amending the CUAEPP that was rejected by the Council of Ministers in February 2000:

- There is no substitute for a strong, independent and competent regulator. Over recent years, the technical competence, strength and continuity of the Bulgarian regulator has been strongly supported by Western experts. Major efforts are still needed to ensure that the regulatory authority achieves a status that is comparable with that considered acceptable in Western European countries. The independence of the CUAEPP from bodies concerned with the promotion and supply of nuclear power needs to be made explicit,
- Managerial and organisational stability of the regulatory body should be maintained,
- Budget and salaries for the CUAEPP need to be increased to allow it to recruit and retain sufficient numbers of competent staff even though agreement has been given to increase staffing levels. In addition, the CUAEPP needs funds to obtain independent technical support when required and to support independent nuclear safety research,
- There remains a shortage of competent technical support organisations within the country. The technical capabilities available to the regulator need to be enhanced and should be independent from those used by the licensees.

20. In addition to the progress that has already been made, the CUAEPP needs to:

- Ensure that sufficient resources are committed to continue the drafting and introduction of new and revised legislation identified in the CUAEPP Improvement Plan, and to the programme for codifying CUAEPP's basic nuclear safety and licensing requirements.

Chapter 2: Nuclear power plant safety status

Data

1. On the site at Kozloduy, Bulgaria has in operation six nuclear power plants operated by the state owned company Kozloduy NPP. The Bulgarian Government has announced the closure of Kozloduy units 1 and 2 not later than 2003.

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Kozloduy 1	VVER-440/230	1970	1974	2004
Kozloduy 2	VVER-440/230	1970	1975	2005
Kozloduy 3	VVER-440/230	1973	1980	2010
Kozloduy 4	VVER-440/230	1973	1982	2012
Kozloduy 5	VVER-1000/320	1980	1987	2017
Kozloduy 6	VVER-1000/320	1984	1991	2021

2. On the Belene site, construction of two VVER-1000/320 units was started in the 1980's but the work was frozen in 1990.

(i) Kozloduy units 1-4

3. The main parts of the information summarised in this chapter are based on knowledge and experience acquired by the Technical Safety Organisations (TSOs) during the Kozloduy short term upgrading programmes in the early nineties and during the TSOs assistance to the regulatory body within the framework of EBRD's Nuclear Safety Account (NSA) programme.

   Information on more recent issues was acquired during the WENRA-Task Force Mission in October 1999.

Basic technical characteristics

Design basis aspects

4. The first four units on the Kozloduy site are VVER-440/230 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2. Units 3 and 4 are more advanced type VVER-440/230 reactors having some of the design improvements of the later VVER-440/213. These include three-way redundancy and better segregation of safety systems, an Emergency Control Room and a low pressure core cooling system. During the early 1990's the utility implemented reconstruction programmes on all units based on IAEA recommendations and TSO advice. This involved installation of additional safety systems with the objective of eliminating or diminishing major safety shortcomings. The aim of these programmes was:

- To establish the Reactor Pressure Vessel status,
- To improve the behaviour of the confinement system,
- To improve the plant behaviour in respect of internal and external hazards,
- To improve systems and equipment reliability,
- To improve organisation and operational safety.

Two further major items are being implemented on a longer-term basis up to 2002:

- To demonstrate the ability of the plant to cope with Loss of Coolant Accidents larger than the current (100 mm) Design Basis using conservative analysis; the licensing analyses of the new design basis accident (200 mm) have been performed and are currently under review at the CUAEPP,
- To demonstrate the ability of the plant to cope with the complete rupture of a main primary coolant pipe (as Beyond Design Basis Accident) using best estimate analysis; analyses are expected to be available by the end of the year 2000.

Reactor pressure vessel and primary pressure boundary

5. The current condition and the inspection programme of the reactor pressure vessels (RPV) appear adequate. The RPVs of units 1 and 3 were annealed in 1989 and the RPV of unit 2 in 1992. Measurements of impurity concentrations in the weld near the core, and recent experimental results on irradiated samples taken from units 1 and 2 RPV, indicate that under the current design basis with postulated 100 mm break LOCA further annealing of RPV 1 and 2 would not be needed. However, investigations of additional samples seem to be necessary to confirm the re-embrittlement behaviour. Whilst internal cladding prevents direct sampling of RPV 3, it can be established from the original test coupons that chemical composition of the key weld of RPV 3 is bounded by those for unit 2. For unit 4, lower impurity contents in the affected weld mean that RPV embrittlement will not be a problem during its operational life. As part of the revision of the design basis to a 200-mm break, an extended pressurised thermal shock analysis of the RPV is necessary.

6. The utility operating Kozloduy 1-4 has implemented measures to reduce the probability of a large primary circuit break. The present design basis covers pipe ruptures up to 100 mm including primary to secondary leakages in the steam generators (SG). Pipework above 100 mm, i.e. 200-mm pressurizer surge lines and 500-mm main coolant circuit pipework, together with major primary circuit components such as main coolant pumps and valve bodies are covered by a state-of-the-art leak-before-break case (LBB). The LBB case has been performed by a Western industrial company under a Phare contract and has been accepted by the CUAEPP. The proposed extension of the DBA to cover ruptures up to 200 mm, if implemented, will provide some overlap between the prevention and mitigation measures. The calculations required to demonstrate fulfilment of LBB criteria for the 500-mm and the 200-mm primary circuit pipework have been carried out. The LBB case is underwritten by an in-service inspection programme and by two suitable instrumentation systems to detect incipient leaks. The LBB criteria commonly used in the Western countries require that three independent reliable and fast leak detection systems be used. A third independent system is currently being considered to replace another less sensitive one. The risk of a large primary to secondary leak caused by a steam generator collector head lift has been reduced by the use of flow limiters and by specific maintenance including in-service inspection. With the above-mentioned installation of the third leak detection system it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

Confinement

7. Despite recent efforts that have led to significant reductions (by a factor of 10), the confinement system leak rate is still excessive, and effort is required to further reduce it. A necessary confinement improvement, the jet vortex condenser discharging through a water pool, is planned to ensure the confinement's structural integrity in case of large break LOCA accidents up to 500-mm breaks. Implementation on units 3 and 4 is planned for

installation before 2002. This design solution is completely different from that already installed at Bohunice. However, the jet vortex condenser still requires confirmation of the claimed performance and a proof of the absence of unwanted side effects under the whole spectrum of conditions. Therefore, for Kozloduy 1 and 2, implementation cannot be expected taking into account the declared shutdown dates of these units.

Safety systems and hazards

8. By 1997, with assistance from the EU (Phare) and Nuclear Safety Account, substantial short term safety improvements have been implemented on all four units, e.g. improvements to reactivity control and additional reactor protection signals, measures to ensure the integrity of pressurised components, measures for improving protection against hazards in general (e.g. fire protection), and improvements to emergency power supply. In order to provide reliable cooling of the reactor circuit a new emergency steam generator feed water system (2x200%) has been implemented for units 3 and 4. Extension of this system to units 1 and 2 was provided in the year 2000. This system allows the cooling down of the corresponding unit and maintaining it in the cold shutdown state. A primary bleed and feed capability is available for all units. The steam lines are fixed and protected against multiple breaks in the non-isolatable part as well as inside the turbine hall downstream the isolating valves. Upgrading for protection against earthquakes is going on to achieve new seismic requirements of 0.2 g.

9. Units 3 and 4 are already equipped with a low-pressure core cooling system that facilitates Design Basis Accident extension. Compared to the VVER-440/213 type design, however, accumulators are absent and ECCS pumps and confinement spray pumps are located in the common boron compartment room.

I&C systems and emergency power supply

10. Replacement of safety related I&C (Reactor Protection System) will be necessary if it cannot be demonstrated that the reliability of the old relay based system complies with current international standards. It has to be noted that this replacement will be impractical for units 1 and 2 due to their limited residual lifetimes. The emergency power supply system fulfils international requirements, e.g. IAEA Safety Guides. For each unit the system is redundant and single-failure proof, and the equipment is qualified for accidental conditions.

Beyond design basis accidents and severe accidents

11. A series of safety improvements have been introduced in recent years in order to cope with some BDBA conditions such as the installation of a new emergency feed water system, emergency feed water supply by mobile pumps, implementation of equipment and procedures for primary bleed and feed. Consideration should also be given to a mitigative severe accident management strategy when the prevention-related work is reasonably complete.

Safety assessments and programmes for further improvements

12. In the early 1990's a consortium of Western TSOs assessed the safety status of units 1-2 and units 3-4 separately, and reviewed the corresponding modernisation programmes designed for safety during short-term operation. The TSO consortium gave recommendations for short-term safety upgrading measures under the condition of limited operational time, which were additional to those already identified by the utility.

    In 1997 the utility proposed a more extensive safety-upgrading programme for units 1-4.

Several modifications have already been introduced, with the aim of operating these units up to the end of their design life. This programme has undergone several updates, but has not been reviewed systematically by the regulatory authority. For internal review by the utility, a plant modification procedure exists in the frame of the NPP QA programme.

Safety assessment and documentation

13. The lack of Safety Analysis Reports (SAR) to Western standards for units 1-4 is a significant shortcoming, even though many different analyses were performed in the past. In the early nineties, international ad-hoc teams or foreign expert organisations rather than Bulgarian experts carried out a major part of the safety assessment used as a basis for safety upgrades.

14. A limited number of (mainly) generic safety analyses are available for units 3-4. In support of the CUAEPP, Western TSOs in collaboration with Bulgarian institutions have recently developed the requirements for a detailed Safety Report for units 3-4. This so-called Safety Substantiation Report (SSR) has to be provided to the regulator on completion of the extended modernisation programme (expected in 2002). The NPP has recently submitted a first revision of the Safety Substantiation Report for units 1-4 to the regulatory body. At present this report is undergoing a second revision.

Probabilistic safety assessment

15. Level-1 PSAs of varying levels of complexity have been carried out, considering the plant design status after short term upgrading and covering initiating events at full power. Separate PSAs for units 1-2 and 3-4 were performed by Bulgarian institutions, in collaboration with those from Spain and Russia, partly based on generic data from Russian NPPs and also data from the IAEA. In the PSA for units 3-4 seismic effects and internal fires were also considered. At present they are in the process of verification after IPERS missions. A PSA level-1 for shutdown states is currently underway. In-depth review is still outstanding.

Decommissioning

16. Bulgarian regulations, based on rules inherited from the former USSR, require the utility to provide documentation for decommissioning at least five years before the planned shutdown of a reactor. At present the preparation of technical proposals for units 1-2 decommissioning is underway in the frame of a Phare project with completion planned in 2000.

Operational safety

17. There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased.

Organisation, procedures, operation and maintenance

18. A number of upgrading measures have been fully implemented in Kozloduy units 1-4, or are well advanced:

- The training of existing and new personnel is now based on a systematic approach; management training has been introduced; operating personnel now have access to a modern multi-functional simulator of the Kozloduy training centre with well trained instructors,
- The technical specifications for operation have been significantly upgraded and are now unit specific,

- Symptom-oriented accident procedures for units 1-4 are under development in the frame of an international programme for VVER-440/230 reactors and are planned to be implemented by the end of the year 2000.

Safety culture and management, quality assurance

19. Since 1992, with Western assistance to the utility and the safety authority, plant management has pursued the objective of improving operational safety. The main goal of the management is to motivate personnel to continue the gradual increase in the safety and reliability of operation in order to reach a level comparable to Western practices. Results of the OSART mission of the IAEA to Kozloduy units 1-4 in January 1999 show that the status of operational safety has significantly improved. OSART gave a series of recommendations and encouraged NPP management to continue these improvements. A follow up OSART mission was agreed for the end of 2000.

20. There is competent staff at the plant dedicated to the continuous safety upgrading process. The management structure has been reorganised, the responsibilities clearly defined, and a Quality Assurance (QA) programme established. In the past the utility and the plant management have made significant progress in the implementation of a modern safety management system but improvements are still needed. In early 2000 the structure of previous NPP management (EP-1 and EP-2) was reorganised and now the units 1-4 (VVER-440) and units 5-6 (VVER-1000) have a common management.

21. The announcement of closure dates for units 1 and 2 present a new challenge for the utility and the plant management. Appropriate measures will be needed to ensure that motivation of staff for safe operation remains adequate during the remaining period of operation.

Operational experience

22. A systematic analysis of operational experience feedback (from Kozloduy and from other PWRs) has been ongoing since the early nineties.

Emergency preparedness

23. There is an on-site emergency plan in place. However, the national approach to emergency planning as a whole is currently under review.

(ii) Kozloduy units 5-6

24. The statements presented in this chapter regarding the safety of Kozloduy units 5-6 are based on the knowledge gained through active TSO involvement in the plant modernisation, IAEA mission records, and the information received through the VVER regulators forum.

Basic technical characteristics

25. The units 5-6 on the Kozloduy site are VVER-1000/320 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2.

26. In principle, the main safety features of units 5-6 are similar to the design of Western PWRs of the 1970's. In the early years the units suffered from frequent disturbances mainly due to the low quality of some equipment. With the replacement of some control valves, such as Feed Water control valves and a number of items of I&C and electrical equipment, as well as modification of the Steam Generators, a reasonable performance has now been achieved. This is important for safety because the frequency of disturbances that might

initiate an accident has been reduced.

Safety assessments and programmes for further improvements

27. A plant specific safety assessment is not yet available, although insights gained from TSO assessments of similar plants (e.g. Rovno 3) may be applicable to units 5-6. In developing the extended modernisation programmes for the VVER-1000 reactors, the utility has performed some plant specific safety analyses based on both deterministic and probabilistic approaches. But it has also used IAEA recommendations and operational experience at similar plants.

28. The PSA for Kozloduy units 5-6 is the first one performed in Bulgaria by its own experts. It is a level-1 study covering initiating plant events at full power, and also including fire and seismic events. It has undergone an IPERS mission review and a review by Western TSOs. In the frame of the modernisation programme, the operating organisation intends to adapt the PSA to the new plant status taking into account the TSO recommendations.

29. A programme for further upgrading of the units 5-6 is at an early stage and has been reviewed by Western TSOs. The main safety improvements relate to fuel and control rod optimisation, long term cooling including measures for prevention of sump filter clogging, electrical systems, instrumentation and control, containment integrity and radiation monitoring. The programme involves major Western and Russian partners and is planned for completion in stages over the next few years. Safety assessments by Western TSOs for similar plants in Ukraine and the Russian Federation have indicated that, after safety upgrading, it should be possible to achieve a level of safety in line with international recognised safety practices. However, to confirm this, a consistent safety case needs to be established and an adequate safety analysis needs to be made. Both will need to be reviewed by the CUAEPP.

Operational safety

30. Information and conclusions presented above for the Kozloduy 1-4 units are also generally applicable for units 5-6.

National industry infrastructure for technical support

31. In Bulgaria there are only limited resources of independent technical support organisations in support to Kozloduy NPP. These include Energoproject Sofia, several institutes of the Academy of Science, Riskengineering, ENPRO consult and BEQE.

On-site spent fuel and waste management

32. Spent fuel of the VVER-440 reactors is stored in an on-site fuel store erected in the 1980's. For the VER-440 units there is currently an agreement with Russia which permits transport of this spent fuel back to the Russian Federation. Presently the spent fuel store is being modified to accept VVER-1000 fuel from units 5-6. This fuel is currently stored in pools within the containment and the storage space is nearly full. Radioactive wastes originating at Kozloduy are stored in interim storage facilities and an on-site cementation plant for liquid wastes is being built, although with considerable delay.

Conclusions

General remarks

33. There have been significant improvements in the standards of operational safety at all units and staff awareness of safety issues has demonstrably increased.

34. The lack of SAR is a serious shortcoming for judging the safety of NPP.

Kozloduy units 1-4

35. The short term upgrading measures implemented at units 1-4 have significantly improved the safety of these units. The measures taken so far have been directed mainly to the prevention of incidents and accidents.

36. Despite the safety improvements already achieved and considering the present safety status of the plant, there are still some major safety issues which are closely linked to the original basic design of the VVER-440/230 reactors and which are difficult to be removed. Among these are the limited confinement function and capability and the vulnerability against common cause failures. For Kozloduy 1-2 the implementation of relevant measures cannot be expected taking into account the announced closure dates.

37. Further safety improvements are being implemented or planned. The current safety-upgrading programme includes the extension of the design basis to a 200-mm break and the consolidation of the confinement system improvements. The utility and its technical support are motivated to the implementation of these improvements and have announced their intention to implement safety-upgrading programmes to mirror those that have been implemented at Bohunice V1. However, the work at Kozloduy is at least three years behind that at Bohunice and consequently safety improvement is not as far advanced.

38. In view of the large amount of work required to be carried out in the next modernisation stage it is difficult to have a final judgement on the adequacy and feasibility of all measures foreseen in this programme. It seems that financial provision for continued safety improvements are inadequate for Kozloduy 1-4.

Kozloduy units 5-6

39. In principle, the main safety features of these units are similar to Western PWRs. A programme for further upgrading of these units is at an early stage and has been reviewed by Western TSOs.

40. Safety assessments done by Western TSOs for similar plants in Ukraine and Russia indicate that, with the completion of the planned safety upgrades, it could be possible to achieve a level of safety for units 5-6 that is in line with international recognised safety practices. However, to confirm this, all safety measures from the programme have to be implemented and a consistent safety case has to be established. Both have to be reviewed by the CUAEPP.

References

1. Convention on Nuclear Safety (CNS), Answers to Questions on the National Report of Bulgaria, April 1999.
2. International Conference on the Strengthening of Nuclear Safety in Eastern Europe, Vienna 14 - 18 June 1999, IAEA-CN-75.

CZECH REPUBLIC

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. A new Atomic Act (law on peaceful utilisation of nuclear energy and ionising radiation) came into force in 1997. It confirms the SÚJB as the responsible body for supervising the utilisation of nuclear energy and ionising radiation. It defines the competencies of the SÚJB for the licensing of nuclear installations as well as the assessment, inspection and enforcement activities.

2. The Atomic Act states that the operator is responsible for the safety of its installations. The company that operates the nuclear power plants is a share holder company in which the state controls the major part.

3. Since the new Atomic Act came into force in 1997, the SÚJB has prepared or revised all regulations arising from the Atomic Act. The issuing of these regulations is an important accomplishment of the SÚJB.

4. The Czech Republic is a contracting party to all key international conventions dealing with nuclear safety.

5. The nuclear legislative framework in the Czech Republic is comparable with Western European practice.

Status of the regulatory body and technical support infrastructure

6. The SÚJB is a central agency of the State Administration reporting to the Government. Its President may participate in the meetings of the council of ministers. If needed, the Vice-Prime Minister, in charge of economy and finance, ensures the link between the council of ministers and the SÚJB. The SÚJB is funded from the State budget, approved by the parliament.

7. The SÚJB is responsible for nuclear safety, radiation protection, transport of nuclear and radioactive material, international notification of incidents and accidents, the provision of information to the public, nuclear material accountancy, and the import and export of dual purpose equipment. The SÚJB plays an important role in the emergency preparedness and planning in conjunction with other administrative departments.

8. The SÚJB has the power to issue and withdraw authorisations. It also has the power to impose penalties on the operators for any violation of the conditions of an authorisation. Enforcement actions by individual inspectors can be appealed to the SÚJB President, the next level of appeal being the court of justice.

9. The SÚJB considers that its current budget is sufficient. It obtained a 13% increase in 2000 to facilitate the licensing work for the Temelin nuclear power plant. The SÚJB has a special budget for research, which is divided equally between radiation protection and nuclear safety. Certain administrative constraints arise for the SÚJB when it contracts for technical

support. Except for small contracts or matters that are urgent from a safety point of view, the SÚJB is obliged to go through an open tendering process. This does not favour the long-term contractual technical support that the SÚJB needs.

10. The SÚJB was able to recruit 30 new staff over the last 3 years, which led to a total of 161 staff (as of 1st January 2000) engaged on nuclear safety and radiation protection activities. The National Radiation Protection Institute, with a staff of 110 providing technical support on radiation protection, is under direct SÚJB supervision. From 1st January 2000 another TSO with 45 staff came under the direct control of the SÚJB. Although its main field of operations is in the area of non-proliferation of nuclear, biological and chemical weapons, it also has considerable capabilities in the area of radiation protection and emergency preparedness.

11. Technical support for nuclear safety is provided by the Nuclear Research Institute (ÚJV), Institutes of the Academy of Science of the Czech Republic, universities, private companies and foreign organisations (for example from Slovakia). But there are a limited number of experts available from within the Czech Republic, which leads to the SÚJB needing to share competencies with the operators. Moreover, the contracting procedures with which the SÚJB must comply reduce the possibility of having long-term contracts for dedicated regulatory technical support. For the future, the SÚJB would prefer to replace some of the short-term contracts with individual contractors by long-term agreements with an extended scope of support. The administrative rules should be adapted in order to make this possible.

12. It is considered that, in general, the SÚJB has a status comparable to that of Western European regulatory bodies.

Status of regulatory activities

13. Since 1992, a number of national and international evaluations of the SÚJB have taken place. The recommendations of the various missions and support programmes have been used effectively in the development of Czech regulatory activities. The SÚJB takes an active part in international regulatory co-operation.

14. The Atomic Act authorises the SÚJB to draft subordinate regulations which, after approval by a legal advisory group of the Government, are signed by the SÚJB President. The laws and decrees issued in the Czech Republic contain very detailed requirements. The SÚJB needs to provide the Government with feedback on the application of the current regulatory pyramid and, if appropriate, propose the necessary changes. The SÚJB intends to continue developing technical guidance documents on the application of these regulations for the operators as soon as resources are available after the licensing of the Temelin nuclear power plant.

15. A well-defined licensing process for nuclear installations according to Western practice has been set up in the Czech Republic. It is governed by the Atomic Act and the Construction Act and includes the steps of siting, construction, operation and decommissioning. Major licences for siting, construction and permanent operation are issued by the District Authorities of the region where the installation is located. Such licences cannot be granted if the SÚJB issues a negative opinion regarding the safety of the plant. The District Authorities collect opinions from all other involved bodies of the state administration, including SÚJB. In addition to this process, there is a set of individual SÚJB approvals,

which have to be granted (in accordance with the Atomic Act), for individual steps within the siting, construction, operation and decommissioning phases of a nuclear installation. The environment impact assessment, which is part of the licensing process, includes a statement on the decommissioning options.

16. The methodology for assessment of safety related documentation is derived from US NRC practice. In addition to the assessment of the safety analysis reports, the SÚJB also assesses and approves such documents as plant technical specifications, the physical protection plan and the utility's quality assurance programme. Requirements for periodic safety reviews are included in licence conditions, usually requesting a review after 10 years of operation. However, when a plant is undergoing a modernisation programme, the periodic safety review is regarded as part of that programme.

17. The SÚJB inspection activities also derive from US NRC practices. They are based on a biannual inspection plan. The inspection plan and the inspection committees are the foundation of the SÚJB system of experience feedback. The SÚJB has established event-reporting requirements for the licensee and has developed a system for analysis and feedback of the licensee's operating experience. This is similar to Western European practice. The SÚJB also actively participates in the INES and international event reporting systems. In addition to its participation in the VVER regulators' forum, the SÚJB has an international agreement with Slovakia and Hungary to share the experiences gained at Dukovany, Bohunice, Mochovce and Paks.

18. The SÚJB has established two advisory committees, one for nuclear safety, the other for radiation protection. This provision is recognised as a good practice. In addition, special advisors have also been contracted for the licensing of the Temelin nuclear power plant.

19. In summary, the SÚJB has developed a series of regulatory practices that compare favourably with those of Western European nuclear regulators. The SÚJB is giving high priority to the licensing of the Temelin nuclear power plant and will resume the development of guidance documents after this period of intensive activities.

Emergency preparedness on governmental side

20. The new Act in the field of emergency preparedness and planning was passed by the Parliament in June 2000. In the case of an emergency situation of any kind, the co-ordination of all activities is the responsibility of the Inter-Ministerial Crisis Co-ordination Committee. This is composed of sub-committees such as the one for protection of the public, of which the SÚJB President is a member.

21. In the case of a nuclear emergency, the SÚJB has a role to advise the authority responsible for the protection of the public. To this end it has created an emergency response centre.

22. On-site emergency plans are approved by the SÚJB. It also ensures their consistency with the off-site plans that are approved by the head of the District Authority.

23. Neighbouring countries, e.g. Austria, have been invited as observers during emergency exercises. The national organisation for emergency preparedness needs to be further tested during exercises. However, the SÚJB considers that it will be difficult to test the national organisation in an exercise prior to the implementation of the new Act. The Czech Republic has participated in INEX-2 international exercises.

24. It is concluded that the SÚJB has taken the appropriate steps to fulfil its role in emergency preparedness.

Conclusions

25. The regulatory regime and regulatory body in the Czech Republic are comparable with those in Western Europe. Nuclear Safety legislation establishes the roles and responsibilities of the utility and the regulatory body. The regulatory body is well engaged in the state control of nuclear activities and the national emergency organisation is defined. A well-defined licensing process according to Western practice has been set up in the Czech Republic.

26. It is recommended that the Government of the Czech Republic consider the following:

- The implementation of the new Act on emergency preparedness and planning needs to be given a high priority,
- It seems that the documents in the regulatory pyramid in some cases may contain too detailed requirements. The SÚJB should be requested to suggest simplifications,
- The contracting rules of the SÚJB need to be adapted so that it can obtain, when appropriate, the necessary high quality technical support on a long term basis.

Chapter 2: Nuclear power plant safety status

Data

1. The Czech Republic has two nuclear power plants (NPP) at Dukovany and Temelin. Temelin NPP is the only plant within EU candidate countries, which is not yet in operation. Fuel loading of unit 1 started on 5 July 2000, fuel loading for unit 2 is planned to be approximately 15 months later.

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Dukovany: (in operation)				
Unit 1	VVER-440/213	1974	02/1985	2015
Unit 2	VVER-440/213	1978	01/1986	2016
Unit 3	VVER-440/213	1978	11/1986	2016
Unit 4	VVER-440/213	1978	06/1987	2017
Temelin: (under construction)			Fuel loading	Design lifetime
Unit 1	VVER-1000/320	1986	07/2000	30 years
Unit 2	VVER-1000/320	1987	11/2001	30 years

2. The plants are owned by CEZ a.s. (Czech Power Company), a joint stock company. CEZ is the sole license holder for the construction and operation of nuclear power installations in the Czech Republic.

(i) Dukovany units 1-4

3. The information given in this report on Dukovany NPP is based on the general knowledge on VVER-440/213 plants (summarised in Annex 2), the Czech National Report for the Convention on Nuclear Safety (April 1999), IAEA documents and information provided by the SÚJB and the NPP.

   The plant specific technical statements mainly rely on information provided by the operator on the occasion of a two days expert meeting with the SÚJB and the operator in June 1999 at Dukovany. Major safety issues were discussed and a summary list of upgrading measures (already implemented or planned in the near future) was provided by the operator. A second meeting of TSO expert organisations with the regulatory authority and the operator took place in May 2000. Since Dukovany NPP was not supported by large Western TSO projects in the past, both expert meetings were most worthwhile in providing technical information on the safety status of the NPP. Other background documentation which has been used is listed in the references.

   An in-depth safety assessment of Dukovany NPP, in particular a review of the modernisation programme (MORAVA), has not been made by Western TSOs. The operator, however, offered to give further help in confirming and expanding the technical information on the plant's safety status given so far.

4. The initial design lifetime for each unit as a whole is 30 years from first criticality. For each of the reactor pressure vessels the design lifetime is 40 years.

Basic technical characteristics

Design basis aspects

5. All units of Dukovany NPP are second generation VVER-440/213 type reactors. Generic safety characteristics of these reactors are presented in Annex 2.

6. For the primary circuit and the safety-related systems, the basic design was made by Russian organisations. The specific plant design was developed and carried out by Energoprojekt Prague, a Czech company which, under Czech law, became the only responsible organisation for the design. All major parts of the primary equipment (except the main circulation pumps) as well as the equipment of the whole secondary circuit were manufactured in the former Czechoslovakia, mainly by Skoda Plzen, Vitkovice, etc. Domestic companies were also engaged in the quality control during manufacturing and construction. Since the nineties fuel manufacturing in Russia is also under Czech quality control. No major quality concerns have been identified in tests and inspections carried out since start of operation. Since the first years of plant operation, safety improvements have been made continuously. A major back-fitting programme had already started in 1991 based on the safety assessment of Greifswald unit 5, analyses and supporting programmes of the IAEA and WANO, and other international co-operation. Major safety improvements were focussed on fire protection, electrical supply, secondary side feedwater supply, and the installation of an emergency response centre. Further improvements are either under design or planned.

Reactor pressure vessel and primary pressure boundary

7. Reactor pressure vessel integrity (especially safety margins against radiation embrittlement) appears to be adequate for all units. Due to the well-balanced composition of material impurities (low content of Phosphorus, Copper) and the protection measures to lower the embrittlement rate, it is expected that annealing will not be necessary for any of the vessels during the design lifetime. To ensure pressure vessel integrity various measures have been introduced, e.g. low leakage core configuration and pressure vessel embrittlement monitoring by a surveillance programme. In-service inspections of the reactor pressure vessels and the primary piping are conducted with state-of-the-art techniques.

8. The piping systems were designed in accordance with Russian and Czech standards. A set of primary pipe whip restraints has been partially installed. A partial leak-before-break (LBB) implementation exists, but it is not relied upon in the safety case. Several preventive measures on steam generator (SG) integrity have been implemented or are underway (e.g. N16 activity measurement on each steam line, measures for exclusion of corrosion damage at flange connections, new feedwater distributors (inside the SG) in order to exclude primary collector thermal fatigue). Accident analyses have been performed and corresponding emergency operating procedures have been revised.

9. After completion of pipe whip restraints, the integrity of the primary pressure boundary is considered to be adequately safe.

Confinement

10. The leak rates have continuously decreased since the commissioning but they are still slightly higher than those that are usually accepted in Western PWR containments. For design basis accidents, however, radiological consequences would not exceed those accepted within EU countries. The performance of the bubbler condenser system in case of Large Break LOCA has been verified in full-scope tests in the frame of the Bubbler

Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents.

Safety systems and hazards

11. In terms of capacity and redundancy the design of the safety systems is in general comparable to Western reactors of the same vintage (see Annex 2). Several measures against hazards (e.g. fire protection) have been taken in order to improve the separation between redundant trains. Further upgrading protection measures are completed or under way. Protection against sump screen clogging has been implemented. Secondary pipe whip restraints are scheduled to be added at the 14.7-m level in accordance with US standards, based on results of a recent analysis.

12. For improving the original generic VVER-440/213 design at Dukovany NPP, an independent Emergency Feedwater System has been installed in a separate building. Former shortcomings have been eliminated.

13. A systematic fire hazard analysis and a flooding analysis were carried out in 1997. Major weak points already have been eliminated (e.g. fire prevention measures). Further measures are underway or planned to be completed in 2000. Measures to cope with high-energy pipe breaks are under development, the completion is scheduled for 2003.

14. Seismic qualification of existing equipment is ongoing in the frame of the MORAVA Project; all new implemented equipment is qualified to withstand 0.1g which is acceptable for this site according to Western practice.

I&C systems and emergency power supply

15. Many improvements on I&C and electrical equipment have already been introduced or are underway. Based on insights gained from reliability analyses, proposals for modifications in the safety related I&C have been developed and will be implemented in 2001. Under current plans of the utility major upgrading of the I&C with digital systems is foreseen by 2010.

16. Various means for condition monitoring of mechanical components, e.g. vibration monitoring of reactor internals, lose part monitoring, on-line operational load measurement as well as ageing monitoring for key components, have been introduced.

Beyond design basis accidents and severe accidents

17. Analyses on some representative beyond design basis accidents (e.g. ATWS, total loss of heat sink, total loss of electrical power) were completed in 1998. The results of these analyses were used in the development of symptom based emergency operating procedures. Analyses on selected severe accidents with core melt scenario have been performed within the scope of a regional Phare project and in the frame of a level-2 PSA.

Safety assessments and programmes for further improvements

Safety assessment and documentation

18. In 1991 the former Czechoslovak Atomic Energy Commission (CSKAE) established conditions for licensing unit 1 for continued operation beyond 10 years (after 1994). In particular, this required the operator to provide a revised SAR, the so-called Operational Safety Analysis Report (OSAR). OSARs also have been prepared for units 2-4. Based on

the OSAR, the SÚJB issues time-limited licences for further operation.

The structure and content of the OSAR are in compliance with the Regulatory Guide n°5 from 1988 and, to a major extent, with the later IAEA guide for periodic safety reviews (IAEA Safety Series 50-SG-O12).

19. All the modifications and safety improvements implemented at Dukovany NPP have to be included continuously in the safety analysis reports of the corresponding unit.

Probabilistic safety assessment

20. In 1992 the first version of a level-1 PSA study for Dukovany NPP was developed by Nuclear Research Institute Rez (NRI), in co-operation with several Czech and Slovak research institutes. In 1994 the updated level-1 PSA for Dukovany NPP was completed. The study was the first level-1 PSA completed for a VVER-440/213 reactor by a Western contractor. Since 1995 NRI has regularly updated the Dukovany level-1 PSA under a living PSA project. The current version of the level-1 PSA includes internal initiating events, fires and floods. The results were used for confirmation and scheduling of upgrading measures within the scope of the MORAVA programme (see § 24) and for refining of the emergency operating procedures. Finally in 1998 the level-1 PSA study was reviewed by an IAEA IPERS mission.

21. In addition, a shutdown PSA (SPSA) has been carried out. The results of the SPSA indicate that the contribution to the total core damage frequency is comparable with that of operation at full power. The results of the SPSA are being used to improve procedures for shut down accidental conditions. First results of a level-2 PSA study are already available and they will be used as an input for severe accident guidelines.

Safety measures and further assessments

22. The Dukovany NPP is involved in international co-operation. Several IAEA missions (OSART, ASSET, IPERS, etc.) have been performed to assess plant operational safety.

    All important safety issues have been addressed in the existing safety programme and are either resolved or are underway. It is intended that the relevant measures will be resolved according to a schedule and will be complete by the year 2002 [1].

23. The Dukovany NPP is practising an extensive exchange with WANO and participates in common activities with other VVER-440/213 operators.

Programmes for safety improvements

24. An extensive modernisation programme (MORAVA) has been established based on Western nuclear safety standards and evaluation of operational experience [2]. The whole modernisation programme will be fully implemented by 2010. The major safety modifications, except I&C, will be completed by 2004. Upgrading of the safety related parts of the I&C with digital systems is planned to be implemented during refuelling outages and will be complete by 2010. The main objective of the programme is to achieve a safety level that is fully comparable with international safety standards and NPPs operating in EU countries.

25. Major upgrading measures which have already been implemented or are under way are for example:

- Automatic protection against primary circuit cold overpressure,
- Protection against sump screen clogging,
- Modification of equipment on the 14.7-m floor as pipe whip restrains, protection against missiles, replacement of valves, two additional steam relief valves, replacement and re-routing of pipes are under way,
- Modifications on the emergency feedwater system (e.g. pipe whip restraints, qualification of valves, et al.) are underway.

Furthermore, additional measures for assuring safe operation are underway, e.g.:

- Reconstruction and extension of diagnostic monitoring equipment,
- Installation of a full scope simulator.

Reconstruction of the I&C system is under preparation.

Operational safety

<u>Organisation, procedures, operation and maintenance</u>

26. Staff responsibilities within the NPP are clearly defined. Nuclear safety and production are separate divisions within the management organisation. The head of the nuclear safety division is a deputy director.

27. Until now the plant operational personnel have been trained at the full-scope simulator of the VUJE Education and Training Centre (Slovakia). At Dukovany, a plant specific full-scope simulator has been installed and training is planned to start there from the beginning of 2001.

28. Symptom oriented emergency operating procedures (EOPs) have been developed in co-operation with Westinghouse. The new EOPs were fully introduced in November 1999.

<u>Safety culture and management, quality assurance</u>

29. The safety culture of Dukovany NPP has been continuously improved. Two OSART missions in 1989 and 1991 noted a high level of nuclear safety and a professional management with competent and trained personnel. A WANO peer review was performed in 1997.

30. A comprehensive quality assurance programme (QA) was established in compliance with IAEA recommendations and regulatory requirements. A management system has been set up in order to assess the safety significance of plant modifications and to ensure their proper implementation.

<u>Operational experience</u>

31. The reliability of plant operation since its first start-up is an indication of the good quality of the equipment.

32. Over the last ten years the average number of unplanned shutdowns (scrams) per unit has been less than 1 per year. A system has been established to ensure efficient feedback of operational experience from Dukovany NPP and other NPPs, especially from VVER reactors.

Emergency preparedness

33. The emergency plan is regularly updated and exercises are carried out annually. The Dukovany Crisis Centre is equipped with necessary computerised support systems. The level of preparedness achieved is adequate.

(ii) Temelin units 1-2

34. Originally it was planned to build 4 VVER-1000 type reactors at Temelin. Construction of the first two units started in 1986. In the early 1990s the original plan, however, was revised. In 1993 the former government decided to complete only units 1 and 2. This decision was re-approved last year by the current government.

35. Background information on Temelin NPP is available from several IAEA documents and to some extent from bilateral co-operation with institutions from EU countries. Furthermore additional generic information on the main safety features of VVER-1000 derives from Tacis and Phare projects on other VVER-1000 plants (e.g. Rovno 3, Kozloduy 5-6).

Several IAEA documents have been used for the assessment given in this chapter, e.g. IAEA report on VVER Safety Issues Resolution at Temelin NPP (1996) [3], review mission reports on Temelin NPP - PSA and External Events (1995, 1996) [4], [5]. A general overview on the Temelin NPP safety status and Safety Analysis Report (SAR) was presented by Czech experts at the IAEA Conference on Strengthening of Nuclear Safety in Eastern Europe, Vienna 1999 [2].

Recently a conceptual safety assessment study on selected safety issues of Temelin NPP was carried out by German expert institutions (GRS et al.) in close co-operation with the SÚJB. For the purpose of the study, plant specific information was made available at several bilateral expert meetings (Dec. 1999 - May 2000) by the SÚJB and the NPP. The study performed by GRS, however, does not replace an overall safety review of the plant.

Basic technical characteristics

Design basis aspects

36. Both units of Temelin NPP are of the standard VVER-1000/320 type. Their design concept is similar to Western PWRs of the same vintage. General safety characteristics of the VVER-1000/320 are presented in Annex 2.

37. The Local Civil Constructions Authorities issued the construction permits for units 1 and 2 in 1986 based on statements of the authority (former Czechoslovak Atomic Energy Commission, CSKAE). These construction permits however were given under some specified conditions, e.g. a re-analysis of all design basis accidents using qualified calculation tools. These conditions have been fulfilled.

38. Qualified Czech (former Czechoslovak) companies manufactured major parts of the equipment (e.g. reactor pressure vessel, steam generators, pressurizer, all equipment on the secondary side). A large part of the systems and their supporting plant (e.g. electrical supply) were designed, manufactured and installed by Czech organisations. Domestic companies were also engaged in quality control of the main equipment and thereby gained knowledge and experience of quality verification.

39. The design of Temelin NPP has been the subject of continuous improvements and

modifications, which have been reviewed by many international expert groups. Numerous individual improvements were implemented already before 1990. Further safety improvements in Temelin NPP have been strongly influenced by international co-operation.

40. Major safety design changes include:

- Replacement of I&C,
- Replacement of core and nuclear fuel,
- Replacement of the original radiation monitoring system,
- Replacement and supplementing of the diagnostic system,
- Replacement of original cables with fire-retardant and fire-resistant ones,
- Significant changes in the electrical design (electrical protections, addition of 2 non-safety grade diesel generators, increased discharge time of batteries).

The most important design modifications (core design and I&C) were supplied by a Western vendor. According to the SÚJB and the NPP, the combination of Eastern and Western technologies did not cause major problems because I&C was replaced completely and there was one main contractor for accident analysis, core design and I&C. However the interfaces between the different technologies were considered continually during design and implementation.

41. Re-assessment of generic safety issues for VVER-1000 reactors was performed just before fuel loading and results were provided to IAEA as an open report to the member states [6].

42. The safety improvement programme implemented in Temelin NPP is the most comprehensive one that has been applied to a VVER-1000/320 plant so far.

<u>Reactor pressure vessel and primary pressure boundary</u>

43. The high quality of the reactor pressure vessel, manufactured by Skoda, Plzen, is well documented. The Nickel impurity content, however, is somewhat higher than today's more stringent specifications. To determine the effect of neutron irradiation on the material, a special irradiation programme covering end-of-life fluence condition has been performed. However, due to some uncertainties, a final assessment with regard to expected changes of material properties currently cannot be made. Therefore close attention has to be given to the monitoring of the embrittlement of the RPV during operation.

To reduce the rate of embrittlement and to ensure pressure vessel integrity, various measures have been introduced, e.g.:

- Pressure vessel embrittlement monitoring is carried out by an adequate surveillance programme which includes irradiation samples being inserted between the reactor core and the pressure vessel wall in the range of maximum neutron flux,
- Preheating of the water of emergency injection systems.

These measures minimise the risk of the RPV brittle fracture; moreover the surveillance will allow the identification of a possible acceleration of material ageing.

44. In service inspections (both from inside and outside) of the reactor pressure vessel (every four years), of the steam generators, the primary piping and other main equipment will be

conducted with state of art techniques.

45. To ensure SG integrity, design modifications and operational measures have been introduced reducing the possibility of primary to secondary leaks. Also a modified technology for the collectors manufacture has been used and, due to the replacement of the Cu alloys in the turbine condenser by pipe bundles from titanium, the water chemistry conditions (pH) of the secondary circuit have been improved.

46. Despite these improvements a leak (up to the maximum flow area of about 14 cm^2) is analysed as design basis accident. The leak size corresponds to the leakage coming from collector header lid lift up.

47. According to the results of a first PSA that was carried out in the early nineties, leakages from the primary to the secondary side have been the main contributors to the total core damage frequency. This was mainly due to conservative assumptions. Considering further updating of the PSA and thereby taking into account all preventive measures that have been introduced in the plant, it can be expected that the initiating frequency of a primary to secondary leak would be assessed more realistically, thus leading to a lower contribution to the core damage frequency.

48. LBB has been applied to the main primary piping (including the pressurizer surge line, low pressure ECCS, residual heat removal system and passive emergency cooling system) in order to reduce the probability of large primary breaks and to avoid the need for further reinforcement of existing pipe whip restraints. Therefore it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

Confinement

49. Constructive improvements on the pre-stressing of the containment tension cables as well as improved monitoring of the pre-stressing system and the concrete structure have been introduced. The measured containment leak rate of unit 1 is comparable to that of Western reactors. It can be concluded that the containment function can be ensured for the postulated DBA.

Safety systems and protection against hazards

50. In terms of its capabilities, the redundancy and separation of the safety systems (e.g. ECCS, EFWS, AC and DC emergency power supply) is comparable to that of Western PWR of the same vintage (see Annex 2).

51. A comprehensive I&C modernisation has been carried out on unit 1 and is underway on unit 2. The old I&C system (original design) was replaced by the new one which is based on modern technology (digital I&C system). Systems which are important for safety have been modernised such as Reactor Trip System, Engineered Safety Features Actuation System and Post-Accident Monitoring System. The new I&C system also covers the Reactor Power Control and Limitation System, the Unit Control System and the Unit Information System. Systematic monitoring is carried out to register failures, using such tools as automatic testers, self-diagnostics, data quality and validity tests, communication diagnostics and manual tests.

52. To ensure correct interactions between the new I&C and the original equipment, all stages of the design and implementation were carried out jointly by Energoprojekt and Westinghouse. Basic design, detailed design and the systematic analysis (functional design)

are verified with best estimate analysis. A special and comprehensive independent verification and validation programme similar to the one at Sizewell B NPP was implemented and accepted by the SÚJB.

53. Additional monitoring of the reactor coolant pressure boundary integrity has been introduced e.g. vibration monitoring of reactor internals, lose part monitoring, on-line operational load measurement as well as ageing monitoring for key components.

54. Measures to address the ECCS sump screen clogging issue in the case of a medium or large LOCA have been introduced but their effectiveness needs to be verified.

55. Far reaching measures for protection against internal hazards have been implemented:

- A systematic fire hazard analysis and a flooding analysis have been carried out,
- Comprehensive measures to increase fire protection (e.g. replacement of the original cable with fire resistant cables),
- Measures preventing consequential failures due to high energy pipe breaks. To compensate for missing spatial separation, additional pipe whip restraints have been installed at the 28.8m level as a protection against postulated steam piping and feedwater line rupture (following current US regulations).

Some specific issues, however, e.g. protection against postulated steam piping or feedwater line rupture on the 28.8-m level, need further consideration.

56. Plant-specific safety demonstration for the functioning of the main steam relief valves and the main steam safety valves under dynamic loading with a steam-water mixture still has to be fully verified. This action is underway. This function is needed to control specific primary to secondary leaks.

57. The seismic re-evaluation of the Temelin NPP locality was performed in accordance with the IAEA methodology (Safety Series 50-SG-S1) using design value of g=0.1. The new seismic analyses were performed for all safety important buildings, components, control systems, I&C and electrical systems. Based on this seismic re-evaluation, modifications and changes were carried out (e.g. installation of additional dampers at the pressurizer pipeline for cold spray).

<u>Beyond design basis accidents and severe accidents</u>

58. Compared to the original design, representative BDBAs have been analysed (e.g. station black out, total loss of heat sink, ATWS). Where it was considered necessary, corresponding measures were implemented (e.g. additional Diesel Generators, pressurizer safety valves withstanding water/steam mixtures for bleed and feed, confirmation of the gas removal system effectiveness for bleed and feed).

59. A systematic analysis of severe accident scenarios selected based on the preliminary analyses and results of PSA studies was performed which permitted the proposal and evaluation of the accident management strategies. Advanced severe accident analysis codes of Western European and US origin have been applied. At present the accident analysis is oriented to support the development and validation of methods and procedures for severe accidents management.

Safety assessments and programmes for further improvements

Safety assessment and documentation

60. Before the commissioning of Temelin NPP the Pre-Operational Safety Analysis Report (Pre-OSAR) was available. Structure and content of the Pre-OSAR is in accordance with the US NRC Regulatory Guide 1.70 taking into account characteristics of the VVER-1000 design and plant specific modifications. Additional requirements of the SÚJB were addressed.

61. The Pre-OSAR is one of the preconditions for fuel loading. For the licensing of plant operation the Pre-OSAR will be amended by results of the commissioning process (tests, etc). This new document will be the final Operational Safety Analysis Report (OSAR). According to regulatory requirements for periodic safety reviews the OSAR has to be updated after each ten years of operation.

Probabilistic safety assessment

62. In the beginning of the nineties a US consultant performed a PSA (level-1 and 2) in co-operation with the utility. The level-1 PSA also includes events during shutdown states. Interim PSA results have been used to complement design related and operational safety upgrading measures. Further updating of the PSA is underway, taking into account plant modifications, Emergency Operating Procedures, more realistic plant specific input data and results from recent accident analyses.

63. The level-2 PSA includes an analysis of the containment strength, a determination of the impact of the core melt progression on the containment structure and an evaluation of the fission product release (timing, frequency and magnitude) for various accident sequences. The Temelin level-2 PSA is one of the first analyses performed for VVER-1000 plants. The analysis involved standard scope, approach and procedures.

    All Temelin PSA models (including level-2) will be updated in order to reflect all design modifications and safety improvements, and the real operational status of the plant.

Safety missions and further safety improvements

64. During construction of Temelin NPP a series of IAEA and others missions have taken place covering various safety aspects: plant construction practice, safety systems evaluation and safety analyses, fire protection, quality assurance, resolution of safety issues, etc.

65. Investigations on selected safety issues (e.g. accident analysis) were also carried out by several Western institutions. Some preliminary Western safety assessments on Temelin NPP were carried out during bilateral co-operation.

Operational safety

Organisation, procedures, operation and maintenance

66. The training of the operating staff of Temelin NPP basically follows the same scheme as that of Dukovany NPP. Plant operators have been trained on a plant specific full scope VVER-1000 simulator at the site. The simulator will be adapted based on commissioning results and first operational experience.

67. Symptom oriented emergency operating procedures have been developed to support operator actions during accident conditions.

Safety culture and management, quality assurance

68. The plant management is committed to develop and maintain a strong safety culture. Support on this subject has been received from IAEA and Western organisations and companies.

69. The utility has developed its own competence achieving independence from the original Russian supplier. Nonetheless, the present situation at the NPP is characterised by good relations with the original designer and close co-operation with Russian experts.

70. A comprehensive quality assurance programme (QA) has been established in compliance with IAEA recommendations and has been approved by the SÚJB.

71. Final tests by NPP staff before commissioning indicated no major concerns and good quality of equipment.

Operational experience

72. Operational experience from other VVER-1000 NPPs and Western PWRs with comparable parts of equipment (e.g. digital I&C) has been examined. Further relevant information has been gained from OSART missions to operating VVER-1000 NPPs, several other IAEA missions and co-operation with WANO.

Emergency preparedness

73. The on-site emergency plan is based on the plan for Dukovany NPP but has been revised and updated. For the programme supporting emergency preparedness at Temelin, technical and normative documentation of other countries and IAEA were taken into account.

National industry infrastructure for technical support

74. The Czech Republic has a strong national infrastructure in the nuclear field due to fact that Czechoslovakia has developed its own reactor in the past and later it was the supplier of main components for VVER reactors. This infrastructure includes also research and general design at Energoprojekt Prague. Skoda is responsible for design work in the frame of the components it supplies. The Nuclear Research Institute Rez provides technical support in different areas, e.g. component integrity, especially the Reactor Pressure Vessel. As part of the infrastructure, one could also note the remaining links with the Slovakian institutions, especially the VUJE Institute.

On-site spent fuel and waste management

75. At present spent nuclear fuel from Dukovany NPP is stored for a 6-year period in the reactor storage pool and subsequently transferred into CASTOR casks. In the early years of operation, spent fuel was temporary stored in the interim storage at Bohunice NPP. All spent fuel has now been transferred back to Dukovany NPP. In 1997, at the Dukovany site a spent fuel interim storage facility with a capacity of 600 tons has been built and commissioned by CEZ.

    The radioactive liquid and solid waste is reconditioned and stored at the site.

76. It is planned that Temelin spent fuel elements will be stored in the reactor storage pool of the plant (inside the containment) for about 10 years. Subsequently the spent fuel elements will be transferred to the interim storage facility.

Conclusions

(i) Dukovany units 1-4

77. The following conclusions can be drawn:

- In the early years of Dukovany NPP operation, modifications were carried out to remove safety deficiencies in the original design,
- Dukovany´s containment structures provide adequate protection against design basis accidents and the overall radioactive releases would not be higher than would be accepted within the EU. However their leak-tightness is not as good as that of typical containments in Western Europe. This would have some influence in the progress and consequences of potential severe accident scenarios,
- Safety assessments and verification documents, e.g. periodic safety reviews, are conducted comparable to Western practice,
- Extensive PSA studies have already been performed or are currently underway,
- The safety culture has been continuously improved and appears to be adequate,
- An extensive modernisation programme (MORAVA) has been established and expected to be implemented within the next 10 years by 2010. All safety improvements except I&C replacement should be implemented by 2004.

78. By full implementation of the Modernisation Programme, it is expected that Dukovany NPP will achieve a safety level comparable to that of NPP of the same vintage operating in Western Europe.

(ii) Temelin units 1-2

79. The following conclusions can be drawn:

- The safety improvement programme for Temelin NPP is the most comprehensive one that has been applied to a VVER-1000/320 plant,
- From the beginning of plant construction improving the nuclear safety, radiation protection and accompanying safety evaluation was a continuous process,
- The combination of Eastern and Western technologies was successfully completed. I&C was replaced completely and there was one main contractor for accident analysis, core design and I&C. The interfaces between the different technologies were considered. A standard Western practice was used to combine Eastern and Western technologies including safety assessment. The commissioning process has to confirm the integration of the different technologies,
- Some safety issues still need clarification with respect to the safety of piping at the 28.8-m level and with respect to the verification of steam relief valves,
- If these issues are resolved, Temelin NPP will achieve a safety level that is comparable to that of operating Western PWR.

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HUNGARY

Chapter 1: Status of the regulatory regime and regulatory body

The information given here is based on experience gained through bilateral and multilateral assistance programmes and co-operation such as RAMG and CONCERT, IAEA missions and other open sources.

Status of the legislative framework

1. The first Hungarian regulations on nuclear safety were issued in the form of ministerial decrees in 1979, when the first two Units at Paks were under construction. These gave a framework for nuclear power plant licensing and safety inspections, and also contained technical requirements for nuclear safety. The first Atomic Energy Act was issued in 1980. The Paks nuclear power plant is owned by the state.

2. Revision of the nuclear legislation and the regulatory framework started in the early 1990s. IAEA guidance was used to help with this process and in addition the Hungarian authorities and experts became acquainted with the legislation and regulatory practices in a number of Western European countries. This international experience was reflected in the new legislation. The new Act on Atomic Energy was adopted by the Parliament of Hungary in December 1996 and the revised governmental decrees came into force in June 1997. The Atomic Energy Act establishes, among other pertinent things, the licensing process.

3. The governmental decrees issued under the Atomic Energy Act specify the duties and authority of the Hungarian Atomic Energy Commission (HAEC) and the Hungarian Atomic Energy Authority (HAEA). The Government decrees also mandate the Nuclear Safety Directorate (NSD) of the HAEA to act as the regulatory body for nuclear safety matters. Nuclear Safety Regulations, published as appendices to the governmental decree (108/1997), comprise the detailed rules and requirements for the nuclear facilities in licensing, quality assurance, design and operation as well as requirements for research reactors.

4. The Act clearly states that the operating organisation carries full responsibility for safety, and the legal status of the utility as an operating organisation is defined in the Act. The role of the regulatory body is to verify that necessary actions to ensure nuclear safety are taken by the operating organisation. The responsibilities of the regulatory body are generally well separated from the responsibilities of the operating organisation.

5. All the key international conventions related to nuclear safety have been ratified and included in national regulations. Peer evaluations of the Hungarian legislation and regulatory framework have been carried out by an IRRT mission of the IAEA.

6. It can be concluded that the legislative framework in Hungary provides a similar level of control to that generally found in Western European countries that have nuclear programmes. The legislation and other regulatory documentation is modern and comprehensive but some changes could be considered regarding their contents to strengthen the independence and co-ordination of the regulatory activities, as discussed in

the next section.

Status of the regulatory body and technical support infrastructure

7. The governmental supervision of the safety of the nuclear facilities is ensured by the HAEC, the HAEA, and relevant Ministers.

8. The role of the HAEC is to prepare proposals on nuclear issues for Government decisions, and to co-ordinate the work of state institutions in the nuclear field. The HAEC is a commission composed of senior officials of the ministries and the heads of the central public administrative organisations that perform regulatory tasks under the Act on Atomic Energy. The President of the HAEC is nominated by the Prime Minister from the members of the Government. Because of its broad mandate, the HAEC is also involved in promotion of nuclear technology activities. The President of HAEC monitors and supervises the activities of the HAEA, and reports annually on nuclear safety to the Parliament. At present the President of the HAEC is the Minister of Economic Affairs, and in this role he is in charge of energy policy.

9. The dual role of the Minister of Economic Affairs is a source of concern regarding independence of HAEC, although the statute of HAEC explicitly requires the HAEC President to act independently of his/her affiliation.

10. The HAEA is an executive body for regulation of nuclear safety and the control of nuclear materials. It has two main parts, each being headed by a deputy of the Director General. The Nuclear Safety Directorate (NSD) has responsibility for licensing, safety assessment and inspection of nuclear facilities. The General Nuclear Directorate has responsibilities regarding the safeguards of nuclear and radioactive material, transport of radioactive materials, and nuclear export and import. The Director General of the HAEA and his deputies are appointed by the Prime Minister.

11. In 1997, the work scope of HAEA was extended to include areas that were previously exclusively under the responsibility of different authorities: civil structures, radiation protection, emergency preparedness, fire safety and physical protection. These authorities still play a role in regulatory control of nuclear facilities. The HAEA also works in co-operation with authorities that regulate regional planning and environmental issues. Governmental decrees give only generic rules for co-operation with other authorities. The appropriate Ministry is responsible for nominating the co-authority. Despite recent improvements, the legal and governmental infrastructure of Hungary, with its distributed regulatory responsibilities, could be better co-ordinated in order to avoid any omission or overlap and to provide for effective co-operation between authorities.

12. Currently, HAEA has about 90 staff. Of these, about 40 are technical experts within the NSD. Taking into account the number and variety of nuclear plants, this number is comparable with Western European countries. NSD staff generally have a high level of technical competence, but in the areas of new responsibilities and inspection practices, as discussed below, there is need for training and/or an increase in the number of staff.

13. The NSD is authorised to take appropriately strong enforcement measures to ensure safety, such as ordering the shutdown of a reactor. It can also oblige the licensee to pay a fine for a violation of rules, although a need for such enforcement measure has not yet been necessary.

14. The Director General of the HAEA is the second instance for all appeals concerning the regulation of nuclear facilities and activities, and this may to some extent compromise the independence of the NSD in its regulatory decisions.

15. Radioactive waste treatment facilities as well as the interim storage of spent fuel at the plant site are regulated by NSD. However, the ultimate disposal facilities for radioactive waste and spent fuel are not classified as nuclear facilities by the present Hungarian legislation and are currently regulated by the Ministry of Health.

16. Funds for the NSD are specified in the Government budget (the major source of funds is the fees paid by the licensee), and it seems that the availability of funds has not been a limiting factor in their daily work. However, to ensure long-term stability, it would be more satisfactory if the NSD salaries were similar to those of the utility.

17. The technical know-how and staff resources within the NSD permits in-depth assessment of key safety issues. Support for this work is available from the national institutes such as KFKI AEKI (Atomic Energy Research Institute) and VEIKI (Institute for Electric Power Research), which together possess an independent, advanced safety analysis capability. KFKI and VEIKI both support the regulator and the utility; independence of work being assured by administrative rules at expert level. The technical support available to the regulatory organisation is competent and sufficient. The regulator has good access to the results of both national and international research programmes.

18. The current regulatory policy of the NSD strongly emphasises the reliance on national resources. NSD can fulfil its duties successfully without foreign assistance.

19. It can be concluded that the NSD of HAEA has duties, authorities and competence that are broadly comparable with those found in Western European nuclear regulatory organisations.

Status of regulatory activities

20. Since 1992, a number of national and international evaluations of the HAEA have taken place. The recommendations of the various missions and support programmes were effectively used in the development of Hungarian regulatory activities. The NSD has for a long time participated actively in international exchange of information.

21. The Director General has issued the Organisational and Operational Code of the HAEA, which describes in detail the regulatory regime, organisational matters, and the ways of working within HAEA. Another document issued by the Head of the NSD establishes the regulatory strategy and serves as a basis for internal Quality Assurance within the NSD. At the level of HAEA the written procedures and guidelines for internal QA, and the planning processes for management of regulatory activities, are in a development phase.

22. In addition to mandatory regulations, the regulatory documents include a set of safety guidelines that are issued by the Director General of the HAEA. There is an active programme for developing new safety guidelines and for upgrading the existing ones, as the need arises from experience. HAEA issued 33 guidelines between 1997 and 1999.

23. In connection with the Periodic Safety Reviews and the related Paks NPP operating license renewal, the NSD has developed a systematic safety assessment process. In addition, the

NSD established an extensive inspection programme. This programme is carried out by an inspection department of six experts who are permanently stationed at the Paks site. Currently, the inspection practices are being reviewed and are increasingly focusing on the work processes of the operating organisation, operating experience feedback, and integrated inspections that cover a certain area as a whole.

24. The regulatory system for analysis and feedback of operating experience from domestic events is similar to common Western European practices. HAEA/NSD is a member of the regional co-operation between the Czech Republic, Slovakia and Hungary, and a member of the VVER Regulators' Co-operation Forum.

Emergency preparedness on governmental side

25. HAEA/NSD acts in an advisory role in emergency situations. The Governmental Co-ordination Committee for emergency planning and preparedness for all kinds of emergencies is chaired by the Minister of the Interior. The Director General of HAEA is the vice-chairman in the specific case of a nuclear emergency. Under the Committee, there is an extensive national system for planning rescue measures, radiation monitoring, and providing information to the general public. HAEA/NSD has established a dedicated centre for emergency response, training and analysis (CERTA).

26. The national system for nuclear emergency preparedness has improved significantly over the past years. National exercises are carried out regularly. In addition, HAEA/NSD participates in the IAEA Central and Eastern Europe emergency-planning co-operation. NSD reviews and approves plant on-site emergency plans. The national emergency preparedness capability was tested in a large international exercise (INEX-2 HUN) in November 1998. The INEX-2 exercises and associated workshops have provided a good stimulus for recent improvements; in this respect it can be concluded that NSD has taken all appropriate steps to fulfil its role as a nuclear safety regulator.

Conclusions

27. The Hungarian approach to licensing, regulating and controlling nuclear facilities has developed strongly in the last ten years. A proper licensing process is in place. Legislation and regulations are up-to-date, and the Hungarian regulatory practices are comparable with those of Western European countries.

28. Issues that need to be considered by the Hungarian Government are the following:

- The fact that the Minister of Energy Affairs is also the HAEC President creates an apparent conflict of interest, even though the formal mandate of HAEC President precludes this,
- The number of different authorities with direct responsibilities in the regulation of nuclear facilities increases the risk that important issues may be overlooked, and reduces the efficiency of the regulatory work.

29. The NSD needs to continue its efforts to develop the inspection approach towards process oriented comprehensive team inspections.

Chapter 2: Nuclear power plant safety status

Data

1. Hungary has one nuclear power plant at Paks with four units:

Paks Unit	Reactor type	Start of construction	First grid connection	End of design life
Unit 1	VVER 440/213	1974	12/82	2012
Unit 2	VVER 440/213	1974	09/84	2014
Unit 3	VVER 440/213	1979	09/86	2016
Unit 4	VVER 440/213	1979	08/87	2017

2. The plant is owned by the Paks Nuclear Power Plant Ltd. which is 99.92% owned by the Hungarian Power Companies Ltd. The latter is owned by the state.

3. The following statements are based on information available in open literature, Convention on Nuclear Safety and other IAEA conference reports, and knowledge gained through many years of bilateral and multilateral co-operation between Hungary and WENRA member countries.

Basic technical characteristics of Paks NPP

Design basis aspects

4. Each unit has a design lifetime of 30 years from first criticality. Operating licenses have no final date of expiration, but are subject to renewal by the regulators every 10 years, based on a periodic safety review.

5. All units in Paks are second generation VVER-440/213 reactors. Generic safety characteristics of this type of plant are discussed in Annex 2.

6. During construction, the quality of the main equipment was controlled by the Hungarian experts. Under the then prevailing political constraints, independent quality verification during manufacturing was not possible to the extent that would have been required in the West. However, no major quality concerns have been identified in tests and inspections carried out since the plant began operating. Also the high reliability of plant operation since its first start-up is an indication of the good quality of the equipment.

7. Since the start up of the plant, many safety improvements have been made and this will continue as a matter of policy throughout the plant lifetime. One of the early improvements was a Hungarian designed core monitoring system (called VERONA). This was installed in 1988, and has since been extended to provide plant operators with information on other important safety parameters.

8. Following a thorough safety evaluation project (called AGNES), a systematic safety enhancement programme was launched in mid-1994. Among the measures already implemented are:

- Relocation of the emergency feed water system outside the turbine building. This removed the major concern about possible complete loss of decay heat removal capability as a consequence of fire or high-energy pipe break in the turbine building,

- Replacement of a number of components to improve the system performance and reliability, and to ensure adequate environmental qualification,
- New systems to improve accident management capabilities,
- Major upgrade of fire protection.

9. In 1996, a review of Paks against IAEA generic safety issues [1] was carried out by the plant staff and IAEA experts, and reached generally favourable conclusions. Although some issues will require continued attention and actions in the future, they are not considered to be significant risk factors today.

<u>Reactor pressure vessel and primary pressure boundary</u>

10. Pressure vessel embrittlement is monitored by an adequate surveillance programme. To date, all vessels have maintained their material toughness with adequate safety margins. Should annealing become necessary in the future, the required technology is available. In-service inspections of the reactor vessels and primary piping are conducted with state-of-the-art techniques. Paks is also taking measures to reduce the possibility of a large primary to secondary leak via the steam generator collector. By these means, it is considered that the integrity of the primary pressure boundary is adequately safeguarded.

<u>Confinement</u>

11. The measured leak rates reflect little variation in construction quality between units. These leak rates are generally smaller compared to other plants of the same type, although they are somewhat higher than leak rates usually associated with Western European reactor containments. The containment internal pressure driving the leak is effectively limited by the bubbler condenser function in the case of design basis accidents, and the overall radioactive releases are not higher than what is accepted within the EU.

12. The performance of the bubbler condenser system in the case of a Large Break LOCA has been verified in full-scope tests in the frame of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA).

<u>Safety systems and hazards</u>

13. In terms of their number, type, and redundancy, the Paks safety systems (diesel generators, emergency core cooling system, emergency feed water system and containment spray system) are comparable to Western reactors of the same vintage. Paks has taken proactive measures to address primary-to-secondary leaks, with modifications currently being implemented, and sump clogging during a LOCA (implemented 1996-1997, and addressing all known related concerns). Hazards including fires, floods, and high-energy pipeline breaks were analysed in the context of the Periodic Safety Review associated with recent license renewal.

<u>I&C systems and emergency power supply</u>

14. All safety-related I&C systems are being upgraded to state-of-the-art digital technology. Reactor protection signals have been modified to provide actuation through two different physical parameters for each initiating event. These signals are processed by different trains and diverse programmes of the protection systems. Emergency power supplies are already up to Western standards.

Beyond design basis accidents and severe accidents

15. The station has studied beyond design basis accidents (such as station blackout, total loss of feedwater, and Anticipated Transients Without Scram, ATWS) with help from the Hungarian research organisations and has developed guidance for operators on how to avoid severe core damage. Severe accident management procedures for Paks are being developed for implementation in the near future. These will be introduced after the completion of new symptom-oriented emergency operating procedures, in parallel with some associated improvements in plant hardware. Additional work is needed to investigate containment response to severe accident phenomena. One Phare project on these issues has been recently completed and another is underway addressing feasibility of filtered containment venting and hydrogen handling. Based on the results of these projects, a comprehensive strategy for managing severe accidents will be developed.

Safety assessments and programmes for further improvements

Safety assessment and documentation

16. The original safety-related documentation supplied with the plant followed vendor practice which then differed from the Western European approaches to safety analyses and reporting. A thorough safety evaluation of Paks was conducted in the AGNES project, which started in late 1991 and was completed by mid-1994. Both deterministic analyses as required in the licensing of Western plants and a level-1 PSA study encompassing internal events, low power and shutdown states, and flooding and fire events have been completed. A seismic PSA is underway. Much attention has been given to appropriate validation of the analysis tools. The results were well documented and were used later to prepare the technical documents of the Periodic Safety Review. They have also been used to update the Final Safety Analysis Report (FSAR). According to regulations issued in 1997 the contents of the FSAR follow the US NRC Regulatory Guide 1.70, accommodating VVER specific features. The first complete version of the FSAR has been submitted to the NSD, and approval is expected by the end of 2000.

17. A periodic safety review (PSR) of all Paks units is required by the current Act and regulations. The first PSR of Paks was initiated by a Ministerial Order issued in 1993. Specifications for that PSR took into account the relevant IAEA safety guidelines. The PSR of units 1 and 2 was completed in 1997, and that for units 3 and 4 was completed by the NPP at the end of 1999. Regulatory review of the PSR is expected to be completed in 2000. The PSR has to be repeated every 10 years. Although one goal of the initial PSR was to update the Final Safety Analysis Report, the current regulations require the licensees to keep the FSAR continuously up to date.

18. A separate project to improve seismic capability has been underway for over five years. This has examined more than 10 000 plant items to determine their vulnerability to earthquakes. A significant part of the necessary modifications has already been done, and the entire project will be completed in 2002.

Programmes for safety improvements

19. According to the best current understanding and available information, no single deficiency representing dominant risk factors remains to be addressed, but a number of measures to further reduce the remaining risk are currently being implemented. These include:

- The addition of equipment to the protection systems and engineered safety features, in order to provide diverse responses to postulated accidents such as large primary-to-secondary circuit leaks, and primary circuit overpressurization,

- Means to protect the reactor containment from phenomena that may occur after severe core damage,
- The development of a new set of emergency operating procedures,
- Condenser replacement to allow high pH secondary water chemistry in order to protect the steam generators from transportation of secondary circuit erosion products.

20. Implementation of these (and other relatively minor) measures has already been scheduled. Less urgent measures are being implemented at a rate of one unit per year, starting from 1999 for unit 1. This programme is expected to be completed by the end of 2002, when all modifications already planned will have been implemented on all units.

21. There are other refurbishment programmes which are part of plant regular maintenance and lifetime management. This mainly involves the replacement of ageing equipment with advanced modern equipment.

Operational safety

<u>Organisation aspects</u>

22. The financial situation of the company is stable. This is demonstrated by the fact that a significant share of annual turnover is regularly spent on investments in safety and reliable operation. Although the management of the Paks plant is today characterised by a strong commitment to safety and reliability, there is some concern that political changes in the Government tend also to induce changes in the station management. Examples of this have been seen in the past ten years.

23. Since the start of operation, the operating company has developed its competence with one aim of attaining independence from the original Russian suppliers. Today they are in a situation where the involvement of the Russian supplier organisation and its successors is no longer a necessity, but is one option in an open bidding process.

24. A positive indication of safety culture found at Paks is the extensive investment in local training facilities. A full scope plant-specific training simulator has been in use since 1988. A recent development is a maintenance-training centre where activities can be exercised before carrying out inspection and/or maintenance on real plant equipment.

25. The drive for increased safety and quality of operations is exemplified by extensive international co-operation. The plant has actively sought contacts with other utility organisations through WANO and especially with other VVER operators. A WANO team was invited to make a peer review of operating practices in 1992, with a follow-up mission in 1995. Since 1988, the plant has also received several safety missions offered by the IAEA, such as OSART, ASSET, and an IAEA safety improvement review. Both WANO and IAEA missions have made recommendations for improvements to operational safety, and these have been given due attention.

<u>Safety culture and management, quality assurance</u>

26. Close contacts are being maintained with European expert organisations and companies operating in the nuclear field. Support has also been received from the IAEA in the development of the plant safety culture. In addition, the plant has developed a QA system based on local regulations derived from IAEA codes and guidelines.

<u>Operational experience</u>

27. The reliability of plant operation and the low frequency of transient events places Paks at

the higher end of the performance ratings for the world's NPPs. This is evident from the small number of unplanned scrams and other operational events throughout the history of the plant.

Emergency preparedness

28. The level of emergency preparedness at Paks is comparable with that at plants in Western European countries, as demonstrated in OECD INEX-2 international exercise, which was based on an accident scenario at Paks.

National industry infrastructure for technical support

29. The VEIKI and KFKI institutes have several decades of high quality experience in fundamental safety-related research, including both reactor physics and system thermal-hydraulics. This ensures a sound domestic competence base and a strong technical support capability.

On-site spent fuel and waste management

30. Spent fuel from the early years of Paks operation has been transported to the Mayak reprocessing facility in Russia. These shipments ceased in 1995 and therefore domestic storage became necessary. Already before 1995, when the possibility to ship spent fuel to Russia became uncertain, and free storage capacity in the spent fuel pools was running low, the Paks NPP awarded a contract for the construction of a modular vault type dry storage (MVDS) system at the NPP's site. The licence for its construction was issued in February 1995 and the licence for commissioning of the first phase (3 modules, 450 assemblies each) of the project was issued in February 1997. Seven modules are now in operation, and construction of next four modules has started. These eleven modules will be able to handle up to 10 years accumulation of spent fuel from all four units, and extension of the MVDS can be made stepwise when the need for more space arises. The facility is designed for interim storage for a period of 50 years.

31. The interim storage facility for spent fuel is being administered by a special organisation designated by the Government, called Public Agency for Radioactive Waste Management (PURAM). The operating personnel are contracted from the NPP.

Conclusions

32. The following conclusions can be drawn:

- The basic technical structure of the plant is good from the safety point of view, and the key safety systems are comparable to Western plants of the same vintage. No major shortcomings in the present safety systems have been identified in any of the several, independent, in-depth assessments done so far. Also the performance of the bubbler condenser containment in case of large break LOCA has been verified in full-scope tests. There is still need for detailed analysis of the experimental results and for complementary tests of other design basis accidents (steam line break and small LOCAs),
- Paks containment structures provide adequate protection against design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU. However their leak rates are somewhat higher than those typical of Western European reactor containments,
- Operational safety aspects are generally at a level comparable to Western plants of the same

vintage. Management changes related to the political changes in the Government cause some concern,

- Periodic safety reviews are conducted in line with Western practices and have already led to an increase in safety.

References

1. IAEA-EBP-VVER-03, Safety issues and their ranking for VVER-440/213 nuclear power plants, April 1996.

LITHUANIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. Lithuania has established the basic laws and regulations related to nuclear safety. The Law on Nuclear Energy from 1996 contains general provisions about licensing, design, operation and decommissioning of nuclear facilities, export and import of nuclear materials, transportation, physical protection, accident management, civil liability, financing, labour relations and international relations concerning nuclear energy. The Law defines a licensing system under which the responsibility for safety is assigned to the licensee. The regulatory body, VATESI, is authorised to issue the licences. The development of the licensing procedure has been completed with international support and the system was successfully implemented in the licensing of Ignalina NPP (INPP) unit 1 in July 1999.

2. The Law on Nuclear Energy also requires the licensing of organisations delivering services and equipment to nuclear facilities.

3. In May 2000 a law on decommissioning of INPP unit 1 was approved by the Lithuanian parliament. The law prescribes that a programme and a plan for decommissioning should be prepared and that all the necessary preparatory measures shall be taken before January 1, 2005.

4. Besides VATESI, the Law on Nuclear Energy establishes the responsibilities of other governmental organisations with respect to the licensing of nuclear related activities. Under the Law, practical work arrangements need to be developed between the different organisations involved in licensing. A reasonable practice for co-ordination between these bodies was created in the mentioned licensing of unit 1.

5. The INPP, the only nuclear power plant in Lithuania, is owned by the state as represented by the Ministry of Economy. At present the operating organisation is not authorised to handle all management issues. For instance, the financing is controlled by the Ministry. In practice this means that the operating organisation cannot assume the full responsibility for safety. This shortfall in the legal status of INPP has been under discussion for several years without any resolution.

6. Lithuania has acceded to all key international conventions related to nuclear safety.

7. It can be concluded that the legislative system in general is in line with Western European practice. However, to be fully comparable with Western European, practice the formal licensing of vendors needs to be abandoned and the nuclear utility needs to be given a corporate legal status.

Status of the regulatory body and technical support infrastructure

8. The regulatory body, VATESI, was established in 1991 from a few specialists from the INPP organisation and the small site inspection group of USSR Gospromatomnadzor. VATESI's responsibilities and authorities are described in its statute and in the Law on

Nuclear Energy.

9. VATESI is advised by a Board appointed by the government. The Head of VATESI reports directly to the Prime Minister on regulatory matters. Consequently VATESI is independent from that part of the state (Ministry of Economy) which is responsible for the ownership of INPP.

10. VATESI is financed through the state budget. As a consequence of the present difficult economical situation in Lithuania, VATESI, as well as other governmental organisations, suffered budget reductions in 1999 and 2000. In 2000 salaries were reduced on average by 16%. At present this restricts VATESI's possibilities to further develop the organisation, to use external expert advice and to participate in international activities.

11. VATESI has a staff of 29 in its head office in Vilnius and a further 5 persons in its resident supervision group at Ignalina NPP. 19 of these are technical experts.

12. According to its statute, VATESI has enforcement powers to withdraw the INPP operating permit for safety reasons and to impose penalties on INPP staff in cases of violation of safety rules.

13. Although in general VATESI staff are competent, more technical staff are needed to handle adequately all normal regulatory tasks, as well as to develop internal procedures and to deal with the new task of decommissioning. The government has approved a plan to increase the number of VATESI's staff. However, due to budget restrictions, new recruitment has been stopped temporarily. The salary level, although significantly lower than at INPP for corresponding work but higher than for other governmental institutions, has been good enough to recruit qualified new staff, and staff turnover is low.

14. VATESI is now in a position to develop and implement an internal Quality Management system. Such a project has been defined and is underway with international support. However there is some concern that VATESI at present is unable to cope with this development work due to its limited staff resources. Important issues to be addressed in the Quality Management system are new integrated inspection procedures, development of regulations, procedures for safety assessment, documentation management, activity planning and human resource planning.

15. The national expertise that is available to VATESI from Technical Safety Organisations (TSO) has increased in number and competence during the recent years. This expertise comes mainly from the Lithuanian Energy Institute in Kaunas but also from the technical universities in Vilnius and Kaunas and from other organisations. A special TSO Council co-ordinates activities, and considers, among other things, whether a TSO involved in any specific matter is sufficiently competent and independent from the interests of the nuclear operator. The licensing process of Ignalina unit 1 engaged these TSOs in a broad co-operation with Western TSOs. This has been of great value for the transfer of Western methods and practices to Lithuania. However the national TSO resources cannot yet be regarded as sufficient to support VATESI. For instance, no national competence is available for supporting VATESI in human factors assessments.

16. VATESI has some access to research results through the universities and the Lithuanian Energy Institute and through international bilateral contacts. International contacts are very important for Lithuania, as they are for other small nuclear countries, and need to be

strengthened.

17. It can be concluded that the resources of VATESI need to be strengthened in order to carry out its regulatory duties.

18. A reorganisation of governmental institutions reporting directly to the Prime Minister is under way in Lithuania. There is a plan to include VATESI into the regulatory sphere of the Ministry of Environment. In this reorganisation special attention needs to be given to the managerial and financial independence of VATESI.

Status of regulatory activities

19. At an early stage, VATESI introduced a system of annual permits for the operation of INPP. This practice has enabled VATESI to exercise strict regulatory control of the plant. In 1999 the INPP unit 1 was licensed in compliance with the Act on Nuclear Energy. The licence is valid until July 2004 and contains a number of conditions. The licensing was a major effort for VATESI and unique for RBMKs, with respect to the scope of safety analysis and regulatory review carried out with international co-operation. This review was done according to Western practice. The work is continuing now with the follow up of the implementation of licensing conditions and the corresponding licensing of unit 2. This licence is planned to be issued in the end of 2002. VATESI has learnt considerably from this process which has contributed to its development as a competent regulatory body.

20. To date VATESI has issued a number of licences for Lithuanian and foreign suppliers to INPP. It should be mentioned that VATESI only makes a general assessment of the Quality Management and the competence of the vendor, and it is the responsibility of the operator to make a more detailed assessment before a contract is signed.

21. During recent years VATESI has developed new regulations, allowing the resident inspector group to apply a more system oriented inspection methodology, in order not to be involved as much in plant daily activities. This development continues and is estimated to require a few more years before it is fully implemented. The end result is expected to be a clearer separation of activities between VATESI and the operator regarding the safety of INPP.

22. VATESI has an internal commission of experts that reviews on a regular basis event reports from INPP and makes recommendations on regulatory measures.

23. The decision taken to close INPP unit 1 before 2005 will require a number of actions from VATESI. The planning of these is underway. A most important task is to make sure that safety is not compromised during the last operating years. This includes technical issues as well as organisational and safety management issues.

Emergency preparedness on governmental side

24. Lithuania has adopted a national emergency preparedness plan that has been internationally reviewed. In the case of severe national emergencies, a Crisis Commission is established at governmental level for co-ordination of all rescue activities. The head of VATESI is a member of the Commission.

25. In case of an accident at INPP, the role of VATESI is to give advice to the national rescue

authorities and to supervise the accident management at INPP without taking part in the operational decisions. VATESI has developed and exercised its own emergency preparedness plan. There is a 24-h cover for on-duty and decision making. Equipping of an Emergency Operating Centre is planned with Phare support. VATESI, as well as other responsible governmental organisations, have reviewed and approved the Emergency Response Plan of INPP.

26. Lithuania participates in the IAEA Emergency Preparedness Harmonisation Project and has also participated in several INEX exercises.

Conclusions

27. The legal and regulatory system has developed substantially over the last years. A licensing system is in place and VATESI has developed its approaches to safety assessment and inspection. Further efforts are, however, needed in order to be comparable with Western European practice.

28. The following issues need to be considered by the Lithuanian government:

- The legal status of INPP needs to be changed in such a way that the operating organisation is given the full responsibility and authority to handle all financial and other management issues and thus be able to assume the full responsibility for safety,
- The full responsibility to select and assess suppliers to nuclear facilities should rest with the operating organisation; hence the legal obligation of VATESI to formally license suppliers needs to be changed, given a reasonable transition period,
- The imposed reduction of resources to VATESI in terms of staff and budget needs to be compensated as soon as possible and the resources successively strengthened in order to handle all normal regulatory tasks, the contracting of necessary technical support and the full participation in international regulatory co-operation,
- The technical support structure and access to nuclear safety research should be further strengthened in order to provide VATESI with the necessary competence to review all major safety issues,
- In the ongoing reorganisation of governmental institutions reporting directly to the Prime Minister, special attention needs to be given the independence of VATESI.

29. The following issues need to be considered by VATESI:

- The development of the internal Quality Management system needs to be given a high priority. In this development work the final step needs to be taken towards an integrated regulatory supervision of INPP, clearly separating the roles of the regulatory body and the operator in all activities. This particularly applies to the role of the resident supervision group,
- Lessons learned from the licensing of unit 1 need to be incorporated in safety assessment and licensing procedures in order to strengthen the independent integrated assessment capability of the regulatory body.

Chapter 2: Nuclear power plant safety status

Data

1. Lithuania has one nuclear power plant with two units in operation at Ignalina (INPP). These are of a later RBMK design and have the highest power rating of any RBMK reactor:

Unit	Type	Present power level		Start of construction	First grid connection	End of design life
		MWth	MWe			
INPP-1	RBMK 1500	4200	1300	1977	12/1983	2013
INPP-2	RBMK 1500	4200	1300	1978	08/1987	2017

2. The plant is owned and operated by the state. The Lithuanian parliament has recently confirmed a governmental decision to shut down unit 1 before 2005.

3. The Information in this chapter is based on first hand knowledge gained by Western TSOs during their participation in the EBRD funded safety review of the Ignalina 1, in bilateral co-operation programmes, in Phare projects and through the IAEA Extrabudgetary Programme on RBMK reactors.

Basic technical characteristics

<u>Design basis aspects</u>

4. After the Chernobyl accident in 1986, design modifications were made at the INPP as well as at other RBMK reactors. These modifications were aimed at reducing the positive void coefficient, improvement of the reactor protection system and display of the reactivity margin (see Annex 1). Compared to other RBMK reactors of the same design generation, INPP has some additional safety features, for instance the following are included in the original design basis:

- A break of the largest pipe in the primary circuit (900 mm header) and,
- Well-diversified emergency core cooling systems.

The design basis for emergency core cooling is comparable to Western plants of the same vintage with regard to loss of coolant accidents (LOCA) and operational transients. Anticipated Transients Without Scram (ATWS) and events such as fires, station blackout and seismic events were not, however, fully covered in the original design basis. With respect to station blackout -due to the large water inventory, the large heat capacity of the graphite and the relatively low power density- the grace period is about four times longer compared to typical Western light water reactors. Hence, there is more time available at INPP for accident management measures. Recent investigations by Western experts indicate that the seismic risk might be lower than earlier considered.

5. Backfitting has improved the original design on several important points:

- The reactor cavity venting system of each reactor is now capable of tolerating 9 simultaneous fuel channel breaks, without any damage to the cavity which could result in significant radiological releases,
- A new system (DAZ-system) has been installed in unit 1, protecting the reactor against the

most frequent and most severe ATWS events, e.g. loss of preferred power, loss of main heat sink, etc. The DAZ-system will also be installed in unit 2 during the maintenance period in 2000. To remove the ATWS issue as a safety concern, a new diverse shutdown system (DSS) is planned to be installed in unit 2 and brought into operation in 2003. This has not been considered for unit 1 because of the limited remaining operating time,

- Additional signals for scram have been installed in both units (low flow through group distribution headers, low operational reactivity margin (ORM), fast pressure decrease in drum separators, high temperature in reactor protection cabinets). Worn out accumulator batteries have been replaced with Western equipment,
- The control room of unit 1 has been upgraded with a new process computer, which also contains a Safety Parameter Display System with 3D neutronics calculations. For unit 2 these modifications are planned to be completed in 2002,
- The fire safety has been much improved in both units by new sprinklers, detectors, cable coating, fire doors, fire ventilation and removal of combustible material.

Status of fuel channel pressure tubes

6. The top welds of the pressure tubes have been thoroughly examined and no safety significant deficiencies have been found. All the pressure tubes have been fitted with new seals. New equipment has been delivered and its use has been validated for more accurate ultrasonic measuring of the tubes and the associated gas gap. New equipment for visual inspection of the inner surface of the pressure tubes has also been delivered and used. Together with the earlier equipment, it is now possible to get complete information about the status of the tubes and the graphite and, as a consequence, a good possibility of assessing the remaining lifetime. Irradiated parts of one tube have also been examined in Sweden and the results were used in the licensing of unit 1. The examined parts were in good material condition with regard to hydration and possible embrittlement. These results have been confirmed by an international peer-review. Further investigations will be made of irradiated parts from unit 2.

Material verification

7. The INPP units have suffered from material defects and leakages in the primary circuit piping, although this has been less than at other RBMK units. The material problems and degradation mechanisms are of the same types that have been found in Western BWRs, especially intergranular stress corrosion cracking in certain piping. Since 1992, the primary circuit has been examined with modern methods and equipment. There is now a good, and for RBMK unique, knowledge about the material status of the large diameter pipes of the primary circuit. The work continues with more refined analysis of crack characteristics and upgrade of the old, as well as installation of new, leak detection systems in unit 2. A new IAEA extrabudgetary programme will address these issues for all RBMK plants with INPP as the reference. Most of the work on non-destructive testing (NDT) is now done by INPP's own staff who are certified according to European Standards (EN 473). Investigations have started regarding implementation of risk based NDT-inspection in unit 2.

Status and capabilities of safety systems

8. INPP has a high redundancy of front line engineered safety systems. In the original design, as in most older Soviet designs, inadequate physical and functional separation made some systems vulnerable to area events and common cause failures. At INPP the fire protection has been improved to protect vital electrical systems, control and protection systems, and the emergency core cooling pumps. Based on a fire hazard analysis, further improvements will be implemented in 2000. Important dependencies in the support systems, e.g. the

service water system, have been identified by PSA and modifications have been implemented. Environmental qualification of the safety-related components needs further consideration. Development and implementation of a system for environmental qualification is planned to be completed in 2002 following recent VATESI regulations on Ageing Management.

Reactivity control

9. As mentioned in § 5, additional protection signals have been installed in order to compensate for deficiencies in the earlier Control and Protection System (CPS) design. As also mentioned in § 5, it has been decided to install a new diverse shut down system in unit 2. This is expected to be financed by the EU. The new system (DSS) is planned to be in operation in 2003. The tender request specification is currently under international review and tendering is expected soon. The system specification includes control rods plus a liquid hold down system, two different sets of instrumentation, trip logic and different clutch mechanisms. The specific design and installation will be proposed by the tendering organisations. In order to address the generic risk of power increase after potential loss of coolant from the CPS channels, part of the existing manual control rods will be replaced with rods of the cluster type. In the new rods the space occupied by coolant is small, representing less potential reactivity insertion. Since 1995, a new type of fuel with higher enrichment and burnable absorber has been loaded into both units, improving fuel economy, facilitating reactor operation and having improved safety characteristics. This fuel together with the new scram signal (mentioned in § 5) makes it much easier to keep the operational reactivity margin within the safety limits.

Status and capabilities of safety systems

10. A pressure suppression type of partial confinement called the Accident Localisation System (ALS) protects the reactor and part of the primary circuit. This system has the following features:

- About 65% of the water volume of the primary circuit, the parts situated below the top of the core, is enclosed within the confinement,
- The confinement is made up of a large number of semi-interconnected reinforced concrete compartments,
- Condensation of steam occurs in ten water pools which are separated in two groups of five,
- Spray nozzles for steam condensation are installed in several compartments,
- A controlled venting system serves to reduce both the peak and the long-term pressure.

11. The design basis for the ALS is the isolation of the steam resulting from a LOCA after the largest pipe break. Hence, the condensation capacity of the INPP ALS system is larger than in other RBMK units.

12. Recent thermal-hydraulic and structural mechanics calculations show that the maximum pressure in the ALS for the ultimate design basis accident remains more than 0.1 MPa below the design pressure and that with the upper bound pressure from calculations the structural integrity of the ALS is not endangered. However, some aspects of the performance of the ALS remain to be addressed in order to verify design basis events at a level of Western European practice. Several studies are included in the ongoing safety programme, with the objective of finalising by 2001 at the latest. The leak-tightness and structural stability of the ALS was a licensing issue for unit 1. The NDT investigations made on the as-built structure indicated no serious deviations from the design

specifications. The strength analysis is planned for completion in 2000. However, the measured leak rate is much higher than observed in Western plants of the same vintage, significantly higher in unit 1 than in unit 2. Unit 2 has steel lining inside the ALS. INPP continues to enhance the leak-tightness of the confinements.

Beyond design basis accidents

13. The capacity of ALS to handle a core damage or an accident state is being analysed in an ongoing level-2 PSA. Several scenarios are identified in the analysis where there is a risk of the ALS bypassing through structural leakage or failing structures. More deterministic analysis is needed in order to make reliable conclusions. The reactor cavity, which is one part of the confinement system, has a relatively low design pressure and, moreover, its hypothetical failure could lead to pressure tube ruptures, consequential failure of the fuel and a release of radioactivity through the bypass. The probability of this pressure being exceeded is however rather low taking into account the newly installed additional protection signals and the increased relief capacity of the reactor cavity.

Ageing and lifetime assessments

14. The closure of the gas gap between pressure tubes and graphite blocks (see Annex 1) is an ageing issue with implications for the lifetime of the fuel channels. During the recent years several hundred pressure tubes of unit 1 have been measured during outages using Western designed equipment. Some 70 tubes have been removed for precise determination of the gap width and for further analysis of the tube material and the channel graphite. Based on present knowledge of the pressure tube and graphite behaviour, the date for gap closure at unit 1 is uncertain but is estimated not to happen before 2002. VATESI monitors the situation and controls operation with annual permits. VATESI has declared that operation will be stopped as soon as gap closure is confirmed. The empirical data basis for this decision will be significantly enlarged by the new gap measurements obtained by the removal of 100 pressure tubes during the maintenance period of unit 1 in year 2000. This will provide a much more accurate lifetime projection than was previously available. In a 1994 agreement with the EBRD, the Lithuanian government declared that re-tubing of the INPP units would not be performed.

Safety assessments and programmes for further improvements

Safety assessments and documentation

15. The International RBMK Safety Review 1992-94 used INPP unit 2 as one of its reference plants. This review considered the safety of RBMK reactors in nine technical areas and resulted in about 300 recommendations and extensive documentation.

16. Under an agreement with EBRD, unit 1 was subjected to a comprehensive safety assessment in 1994-96. It consisted of the production of a safety analysis report (SAR) and its independent review (RSR). The safety assessment was made to Russian regulations reviewed and amended for Lithuania, IAEA codes and guides and equivalent Western standards, but considered only a limited remaining operating time before gas-gap closure was expected. It was also limited due to financial and time restraints. The SAR was reviewed by an independent international team. The Ignalina Safety Panel, an independent group of senior experts, evaluated the findings and drew conclusions based on both the SAR and the RSR.

17. A number of deterministic analyses were carried out in the licensing of unit 1 in order to supplement the SAR, e.g. accident analyses (missing in the SAR), a fire hazard analysis, a single failure analysis of the Control and Protection System (CPS), safety cases for ALS and

the structural integrity of the primary circuit. Some of these analyses will be further developed in the safety improvement programme for 2000-2005. The licensing process for unit 2 has recently started. In this work, a new specific and extended SAR will be developed and reviewed for unit 2.

18. A PSA level-1 covering full power operation has been available for unit 2 since 1994. The quantitative results are based partly on plant specific data and partly on generic data. Limitations exist in the modelling of external events and dynamic effects of LOCA, as well as in the modelling of Common Cause Failures and Human Performance. Based on this study, a level-2 study is being prepared and expected to be completed by the end of 2000.

    An updated version (phase 5) of the PSA level-1 study was recently reviewed by an IAEA IPSART mission. As a result of the mission the study will be further improved.

<u>Programmes for safety improvements</u>

19. Following the safety review recommendations the Lithuanian Government committed itself to financing a new safety improvement programme (SIP-2). The programme contains design modifications, management and organisational development and safety analyses. A first part of the SIP-2 is now finished. According to INPP, 118 of 160 planned activities were implemented by the end of 1999 including 12 specific activities for unit 1. A second part of SIP-2 is now defined for implementation during 2000-2005, which addresses the remaining issues from the safety reviews, issues specifically directed at unit 2 and high priority issues as a result of decommissioning of unit 1. The SIP-2 programme has so far suffered from financial difficulties and there is a concern that it could be delayed.

Operational safety

<u>Organisation, procedures, operation and maintenance</u>

20. INPP is a state enterprise under the Ministry of Economy. The plant Director General reports directly to the Minister.

21. A major problem for INPP is the financial difficulties caused by limited remuneration for the electricity production. The problem of limited electricity demand and insufficient payment for deliveries has been significant during recent years. The payments are just about sufficient to keep the plant operating with regard to staff salaries and operational expenses. For 1999 INPP only received a small part of the planned funding for the SIP-2 programme and there is a concern that this will also happen in 2000. In that case, INPP will face further financial problems and some planned improvement measures will need to be postponed. In any case, the complete safety improvement programme cannot be financed without foreign support.

22. As a whole, the operational and technical support staff shows a high level of technical competence. A significant transfer of Western knowledge has taken place in the last years as a result of extensive international co-operation programmes. As a result of the decision on decommissioning of unit 1, INPP has prepared a programme to reduce the number of employees mainly by outsourcing activities, like district heating, transportation, maintenance, etc. Recently the number of employees was reduced from 5 000 to 4 800. A few plant specialists have left INPP, mostly for work abroad. There is a concern that more specialists will leave as soon as other opportunities arise.

23. The operating procedures have been improved as recommended in the SAR review. Further development of the operating procedures may result from the use of the new full-

scale simulator. New symptom based Emergency Operating Procedures (EOPs), meeting Western standards, are planned to be implemented by the end of 2000. To prepare for the training of the shift teams, INPP instructors have received training in modern control room work management and methods. The new full-scope simulator will considerably contribute to a successful implementation of the EOPs.

24. The Technical Specifications document addresses the necessary operational limits and conditions although the format differs from Western practice.

25. A new computerised maintenance management system is being implemented. There is also a new procedure and computer tools for the handling of plant modification drawings according to Western standards.

26. The training system is undergoing modernisation according to the IAEA Systematic Approach-to-Training model. Training of control room operators on the new full-scale simulator began in late 1998.

Safety culture and management, quality assurance

27. INPP has been subjected to one OSART, two ASSET missions, and a number of other IAEA activities. With the support of Western experts, considerable efforts have been made since 1994 to develop management, organisation, and safety culture at INPP. Several moves towards Western practice have been made for instance the establishment of a Plant Safety Committee. However, it has been difficult for the committee to be fully accepted within the INPP organisation. A new Quality Management System based on IAEA standards has been implemented during 2000 after four years of work. To achieve deeper changes to the old safety culture is proving to be a slow process and is also dependent on changes in Lithuanian laws and regulations. The difficult economical situation and the decision on decommissioning present further challenges to the management of INPP in the development of the safety culture.

Operating experience

28. The operating history of INPP shows decreasing trends for all events categories except for leaks in the primary circuit. Up to 1990, the collective dose of INPP staff was comparable to the world average. From 1990 the dose has increased a little, mainly due to the extensive safety upgrading works. The main categories of events during the 1990s have been equipment failures, leakage in the primary circuit and Control and Protection System problems. There was also a serious bomb threat in 1994, which led to an extensive project with Western support to upgrade the physical protection of the plant. Recent statistics show a decrease in the number of more serious events during the 1990-ties and a slight increase of number of minor events. Of 81 reported events in 1999, 68% were attributed to equipment faults, 17% personal errors, and 15% to deficiencies in procedures.

Analysis and feedback of operational experience

29. Procedures exist for the analysis of events and operational experience feedback. However the communication between the different plant departments and the internal experience feedback, needs to be improved. Regulations require the reporting of ”abnormal events” to VATESI. Event reporting is well implemented. There is an exchange of operating experience between INPP and the other RBMK plants through reports, at technical meetings and telephone conferences.

Emergency preparedness

30. The earlier on-site Emergency Response Plan has been thoroughly reviewed and modified in line with Western standards. Accident classification and alarm criteria have been developed according to IAEA's RBMK guidelines, and following a final review they will be included in the plan. The new plan was exercised for the first time in 1998 and was exercised again in October 1999 using the completely reconstructed Emergency Operating Centre. The plan is now under revision again to accommodate the experience gained.

National industry infrastructure for technical support

31. Prior to taking responsibility for INPP in 1991, Lithuania had little involvement in nuclear activities. Therefore, the country lacks deeper nuclear experience and tradition. The national technical support infrastructure for INPP is improving but will not be sufficient in the near future. In particular, further Western assistance and Russian consultation will be needed for qualified engineering work.

On-site spent fuel and waste management

32. A new interim dry storage facility for 72 spent fuel casks has been built close to the site and recently started operating. A safety assessment of the present facilities for storage of solid and bitumenised waste is going on. Further measures regarding nuclear waste management are included in the present safety improvement programme and have received a high priority as a result of the decommissioning decision.

Conclusions

Design issues

33. INPP belongs to the more advanced and improved design generation of RBMK reactors. In addition, the original design has been considerably improved through the safety improvement programmes and most of the generic safety concerns with RBMK reactors have been satisfactorily addressed. More measures will be implemented as a consequence of, for instance, the installation of a new diversified and independent shut down system at unit 2. The safety situation of INPP is also better known internationally and considerably better documented than for other RBMK plants.

34. The measured leak rate of the confinement system is much higher, in particular at unit 1, than observed in Western plants. Some further aspects remain to be addressed in order to reach a complete verification of the confinement performance for the design basis events. However, compared with Western European light water reactors of the same vintage, there remain weaknesses in the design of the confinement, especially in case of a severe accident:

- The reactor cavity, which is one part of the confinement system, has a relatively low design pressure and, moreover, its hypothetical failure could lead to consequential failure of fuel integrity and a release of radioactivity through the bypass. The risk of this pressure being exceeded is however rather low taking into account the increased relief capacity and the newly installed additional protection signals,
- A LOCA in the primary circuit outside the partial confinement could lead to a release of steam that it would not be possible to isolate. Such a potential accident sequence was accepted in the original design, due to its perceived low risk regarding radioactive release to the environment, but is not acceptable according to Western European design principles.

35. It is not realistic to upgrade the confinement to include the complete reactor pressure boundary. Consequently, regarding mitigation of accidents, a safety level comparable to light water reactors of the same vintage in operation in Western Europe will not be reached. Therefore special attention must be given to prevention of accidents, including the need to ensure a high level of operational safety for the remaining operating time.

<u>Operational safety</u>

36. With regard to the operational safety, the following issues need to be resolved in order to be comparable with Western European plants:

- The financial situation of INPP needs to be much improved in order to cover all operational expenses as well as implementing the safety improvement measures considered necessary for the remaining operating time,
- Issues relating to safety culture need a stronger implementation in order to prioritise safety in all organisational levels, not least after the decision to shut down unit 1,
- The symptom based emergency operating procedures need to be finalised and implemented without delay,
- Due to the decision on decommissioning of unit 1, special attention needs to be given to the keeping of a sufficient number of technical specialists, as well as maintaining the motivation of the staff, for the remaining operating time of both reactors.

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ROMANIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. Legislation regulating the peaceful use of nuclear energy has existed in Romania since 1974. The current nuclear Law, in force since December 1996, defines areas of application together with the roles, duties and responsibilities of organisations involved in the licensing process. A 1998 amendment to the Law was issued to remove inconsistencies with other national Laws and to strengthen the status and the role of the Regulatory Body (National Commission for Nuclear Activities Control - CNCAN).

2. The Law clearly assigns responsibility for the safe operation of NPPs to the operator. According to the Law, CNCAN is responsible for the regulation, licensing, and control of all nuclear facilities in Romania. The legislation requires CNCAN to license not only the operator but also its subcontractors with regard to quality assurance. This approach has the potential to obscure the operator's perception of its primary responsibility for safety at the plant. There is however a plan to revise this law in the near future in order to take advantage from the lessons learned after a few years of its application.

3. The company that operates the Cernavoda NPP is the National Nuclear Company "Nuclearelectrica". It is a state company reporting to the Ministry of Industry and Trade.

4. Romania has ratified the key international conventions dealing with nuclear safety.

5. The nuclear legislative framework in Romania is generally in line with Western European Practice, even if some specific improvements are still necessary (see § 2).

Status of the regulatory body and technical support infrastructure

6. CNCAN is led by a President with the rank of State Secretary, nominated by the Prime Minister and reporting directly to the Government. CNCAN is independent from ministries and organisations that have a role in the use and promotion of nuclear energy. However, some cases of CNCAN involvement in the selection process of licensee suppliers have been noted.

7. CNCAN's organisation and staff composition has undergone several changes. The organisational structure, approved by the Government in 1998, was implemented in spring 2000. An Advisory Committee is also envisaged but this is not yet operational. A licensing board is in place to support the President in the licensing decision process. Regulatory activities relating to the licensing of the nuclear power plant and research reactors are performed by the Nuclear Safety General Division. This currently comprises 31 experts but a significant recruitment programme aimed at hiring qualified experts or new graduates was implemented last year. CNCAN has taken over the function of monitoring the national environmental radioactivity and 174 people are involved in this activity. However, due to their different professional background, these personnel cannot be used for nuclear licensing activities.

8. CNCAN is funded by the Government and, in addition, it is also allowed to retain 50% of licensing fees paid by the applicants. In the last two years priority has been given to assign most of the budget to significantly improve the headquarters infrastructure and to purchase new equipment. By comparison, the development of the technical competencies of staff and the provision of external technical support for the assessment of key regulatory issues might not have received sufficient financial resources.

9. Personnel salaries have been recently increased and this should lead to a reduction in the turnover of qualified staff.

10. In the field of NPP licensing activities, CNCAN has a limited number of senior qualified experts. There is also a limited number of experienced inspectors. Most of the recently hired CNCAN personnel need to be trained in the safety and operational features of CANDU reactors, regulatory methodology and practice, and in inspection practice. The experience gained during the licensing of Unit 1 also needs to be transferred to the new staff.

11. CNCAN has the power to issue, amend, and revoke licences but the licensee can appeal against CNCAN decisions in the Romanian courts. CNCAN has the necessary enforcement power to carry out inspections on the site and to enforce remedial actions when violations are identified.

12. Progress in the further development of regulatory management needs to continue. This should include the finalisation of a Quality Assurance manual that is currently under development using, as reference, the working procedures of the Canadian nuclear safety authority.

13. In addition to insufficient internal technical assessment capabilities, CNCAN currently does not have qualified external technical support for all safety aspects. At present some support in specific areas is provided by the Canadian nuclear safety authority, the IAEA, and private consulting organisations operating in the country. CNCAN is also investigating the possibility to develop a national TSO.

14. Progress has been made by the Romanian Regulatory Body in the recent years and in particular during the licensing process of the Unit 1 of the Cernavoda NPP. Further improvements are however necessary to reach a status comparable to Western European practice.

Status of regulatory activities

15. CNCAN is currently revising the Romanian regulations to make them consistent with the new nuclear law and to bring them into line with Western European practice.

16. According to the nuclear Law, any activity related to a nuclear installation must be licensed by CNCAN. Since April 1999 the Cernavoda plant has been operating based on a two years licence. At present the major regulatory activity is related to the definition and the implementation of a strategy for the extension of the Cernavoda licence after May 2001.

17. As required by the law, CNCAN is involved in the licensing of suppliers to the licensee. Pending amendment of the law, CNCAN has started to develop an inspection practice of the licensee's QA programme, including the control of suppliers. This is more in line with

Western European practice.

18. The Romanian licensing practice has been developed during the construction, commissioning and initial operational phases of Cernavoda Unit 1 and is defined in a regulation under revision. The basic format of the safety documentation is the requirements of US NRC Regulatory Guide 1.70, with specific provisions derived from the Canadian practice. For the licensing activities of the Unit 1 of the Cernavoda NPP CNCAN nominated a licensing manager. Technical advice was also provided by the Canadian nuclear safety authority throughout the licensing process by the presence of a Canadian expert on the site. The CNCAN assessment activities were addressed towards verifying the compliance of the plant design basis with applicable regulations and were mainly based on engineering judgement supported by a few independent analyses. IAEA expert missions also took place to provide additional support for CNCAN decisions.

19. CNCAN has established event-reporting requirements for the licensee and has developed an internal system for the assessment of the plant operating experience. CNCAN is also actively participating in international event reporting systems.

20. At present the CNCAN independent assessment capability is limited because there are insufficient experienced staff members. Some support comes from the Canadian Nuclear Safety Authority, from IAEA national and regional projects, and in the framework of the CANDU Regulators group of which CNCAN is member. A first year of regulatory assistance has been provided under the EU Phare programme.

21. CNCAN site inspection practice was initially developed based on the Canadian approach. It has been improved through the advisory service of the Canadian nuclear safety authority and also through international missions organised by the IAEA. Inspection during construction and commissioning was based on an on-site inspection unit for day to day activities, plus team inspections from the headquarters to address specific areas. Since the start up of the Cernavoda Unit 1, site inspections have been performed mainly by headquarters personnel. It is now planned to assign a resident inspector to the site mainly acting as a liaison with the CNCAN headquarters. Inspection procedures based on the Canadian practice are under development together with a training programme for the inspectors.

22. CNCAN has made good progress in performing its regulatory activities. Some improvements are however still needed to be in line with the practice of Western European Regulatory Bodies.

Emergency preparedness on governmental side

23. A national emergency plan is in place. An inter-ministerial Committee has the responsibility for the control, evaluation, and approval of the national emergency plan. As a member of this Committee, CNCAN has the role of providing technical support and advice, and notifying the public about nuclear emergencies. CNCAN also has the responsibility for approving the on-site emergency plans of nuclear facilities. Romania has participated in a number of INEX international exercises. At present, however, CNCAN does not have enough experienced personnel in emergency preparedness and a dedicated emergency centre is not available at the CNCAN headquarters. There are however plans to address these issues in the future.

Conclusions

24. Romania is taking appropriate steps to establish a regulatory regime and a regulatory body comparable with Western European practice. Roles, duties, and responsibilities of organisations involved in nuclear safety are in line with those assigned to similar organisations in Western Europe. The independence of the regulatory body from the organisations involved in the use and promotion of nuclear energy is fully established by the law and is sufficiently reflected in the practice. The regulatory regime and the regulatory body have both improved during the licensing process of Cernavoda NPP. However some improvements are necessary to reach a situation comparable with the practice in Western European countries.

25. The following recommendations need to be addressed by the Government:

- Despite the critical economic situation the availability of technical support to the regulatory body need to be improved as well as the salaries of the regulatory body personnel,
- The full responsibility to select and assess suppliers to nuclear facilities should rest with the operating organisation; hence the legal obligation of CNCAN to formally license suppliers with regard to quality assurance needs to be changed, given a reasonable transition period,
- Procedures and lines of communications between national organisations involved in the response to a nuclear emergency should be improved.

26. The following recommendations are addressed to the regulatory body:

- The independent assessment capabilities and the inspection practice need to be improved, especially in view of the future licensing activities of Cernavoda Unit 2. To this purpose adequate resources should be assigned to set up and implement a training programme for new recruits,
- In order to be able to carry out its duties and responsibilities in the area of emergency preparedness, CNCAN needs to further develop by establishing an emergency response centre and a dedicated unit in the organisation,
- The existing agreement with the Canadian regulatory body needs to be more effectively used for training purposes and for obtaining advice on regulatory issues specific to CANDU technology,
- It is recommended that a strategic plan is developed to ensure that appropriate resources are allocated to the higher priority issues,
- The ongoing revision of the regulatory pyramid needs to be continued and completed.

Chapter 2: Nuclear power plant safety status

Data

1. Romania has only one NPP into operation. It is a CANDU 6 reactor located in the Cernavoda site. Construction of five CANDU 6 reactor units started at Cernavoda in 1980 and stopped at different stages of advancement (e.g. 46% for unit 1) following the 1989 political changes. Subsequently, it was decided to concentrate on the completion of the first two units.

2. In 1991, the Romanian Electricity State Company (RENEL) signed a contract with a Western Consortium (AECL and Ansaldo), transferring to it the responsibility to complete the construction of Unit 1 and to commission and manage its initial operation. The national industry participated in the construction of conventional systems under a qualification programme supervised by the Consortium. Operating responsibility was transferred from the Western Consortium to RENEL, with commercial operation starting in July 1997 under the conditions of a provisional operating licence issued by the Regulatory Body.

Reactor	Reactor type	Electrical power (MW)	Start of construction	First grid connection	End of design life
Cernavoda Unit 1	Candu 6	705	1980	1996	2026

3. Work on Unit 2 had stopped with 80% of the civil work and 5% of the mechanical work completed. However, the Romanian Government is now committed to complete the construction of the Unit and the financial means for achieving this are being worked out.

4. The Cernavoda NPP is owned by the National Nuclear Company "Nuclearelectrica". It is a state company reporting to the Ministry of Industry and Trade.

5. Due to the close involvement of the Western Consortium in the construction project, Cernavoda has not benefited from EU industrial assistance programmes. Furthermore, CANDU reactors are not in operation in any Western European Country so their detailed safety characteristics are not known to Western European safety organisations.

6. The statements presented in this report are primarily based on information provided by the Romanian regulator and the utility. In addition the Canadian regulatory body was contacted to collect some information on the design of the CANDU 6 reactor and the evolution of regulatory requirements in Canada.

Basic technical characteristics

<u>Design basis aspects</u>

7. The Cernavoda reactor uses natural uranium as fuel and heavy water as the coolant and moderator. It is based on a standard CANDU 6 design developed in Canada in 1979 and is similar to NPPs in operation at Point Lepreau and Gentilly 2 in Canada.

8. In a CANDU 6 reactor the moderator and the coolant are separated by two concentric tubes, the pressure tube and the calandria tube. The pressure tubes (380) and the calandria tubes are housed in a cylindrical tank (calandria) that contains heavy water moderator at

low pressure, surrounded by a concrete reactor vault containing light water for biological and thermal shielding. The pressure tubes contain the fuel bundles, and the coolant is circulated through these tubes. The calandria tubes prevent the moderator from coming into contact with the high temperature coolant. There is an annulus between the pressure tube and the calandria tube, filled with gas (CO_2). The gas is monitored to provide early detection of tube failure. Refuelling is carried out with the reactor at power. This allows removal of any defective fuel from the core as soon as it is identified, so helping to keep the heat transport system essentially clean from fission product activities.

9. A disadvantage of the CANDU reactor concept is that the reactivity void coefficient is positive. The adverse effects of this during transients and accident conditions are counteracted by two independent, diverse and equally capable shutdown systems.

Some advantages of the design are:

- Control devices cannot be ejected from the core, being located in the moderator at low pressure,
- There is the possibility of using the moderator as an emergency heat sink following severe loss of coolant scenarios with the emergency core cooling unavailable.

Pressure tubes and primary pressure boundary

10. The integrity of the primary pressure boundary of the Cernavoda NPP is monitored through a periodic inspection programme in accordance with Canadian Standards. The standard is based on the ASME code adapted to take into account the fact that CANDU is a pressure tube type reactor. In particular the methodology developed in the Canadian Standard is based on the inspection of samples chosen based on pre-established criteria. The pressure tubes of the Cernavoda NPP have been manufactured from a new type of material (Zirconium-Niobium 2.5%) which takes account of the lessons learned following a tube rupture at the Pickering 2 NPP in Canada. The Regulatory Body has approved the measures to ensure the integrity of the pressure tubes and of the primary system.

Safety systems

11. Plant systems are divided into process systems (some of which are safety related) and special safety systems. The special safety systems are the two shutdown systems (shut-off rods and liquid poison), the emergency core cooling system, the containment system and the related supporting systems. An unavailability target of 10^{-3} per year is requested for each Special Safety System as design requirement, while a strict application of the "single failure criterion" is not requested. To provide protection against common cause failures, the mitigating systems (Process and Special Safety Systems) have been divided into two independent groups such that at least one group will be capable of shutting down the reactor, cooling the fuel, containing fission products and monitoring the plant. The emergency power supply system comprises 4x50% stand-by diesel generators and 2x100% emergency power supply diesel generators. For the Cernavoda plant additional assessment is necessary to confirm the plant design margins against seismic events and the adequacy of fire protection.

Confinement

12. The containment provides a complete enclosure of the reactor primary circuit. It is a pre-stressed structure with plastic liner, equipped with an automatic spray system to reduce any pressure increase after an accident. The measured leak rate of the containment is comparable with that of reactors in operation in Western Europe.

Postulated design basis accidents

13 The Cernavoda NPP, like other CANDU reactors, is designed against a set of postulated events based on the concept of single/dual failure. Single failure (corresponding to a single initiating event in the terminology used in PWR plants) is a failure of any process system that is required for the normal operation of the plant. In this category there are events like large LOCA, single channel events, small LOCA, etc. Dual failure (corresponding to single initiating events with associated a degraded performance of engineered safety features in PWR reactors) is a combination of a single failure event described above and the simultaneous failure or impairment of one of the special safety systems (emergency core cooling or containment). For the single failure and the dual failure categories of events, maximum frequencies and reference dose limits for members of the public are established. Quantitative engineering limits (e.g. peak cladding temperature in case of LOCA) are not fixed at regulatory level. This is in line with the Canadian approach in which it is the licensee's responsibility to develop the general performance standards, established by the regulator, into more detailed design requirements. Conservative assumptions are adopted in the plant transients and accident analysis. Plant design bases include external events such as earthquake, flooding, missiles, and for the containment a reference aircraft impact.

Beyond design basis accidents and severe accidents

14 Beyond design basis events like Anticipated Transients Without Scram and Station Blackout are not analysed in the CANDU safety analysis. These types of scenarios are assumed to be prevented by the existing design safety features (two independent diversified and equally capable shutdown systems and a redundant number of standby and emergencies diesel generators). Concerning severe accidents the standard CANDU safety analysis already includes scenarios with the failure of emergency core cooling in which the heat removal is provided by the moderator. For scenarios with more core degradation, the capability of the calandria to provide a spreading of the corium and sufficient heat removal area for core debris as well as the additional capability of the concrete reactor vault as ultimate heat sink are still to be analysed and the corresponding management procedures defined.

In early 2000, Cernavoda NPP developed a new Safety Analysis Strategic Plan targeted towards acquiring and developing severe accident methodologies. This plan is under discussion with the Regulatory Body.

Construction and commissioning

15 The basic safety features of the CANDU 6 concept have not changed significantly over the years. When construction of Cernavoda Unit 1 restarted in 1991, design improvements were introduced similar to those already implemented in the twin plants of Wolsung (South Korea), Point Lepreau and Gentilly 2 as a result of their operating experience and PSA studies. The main improvements include for example better separation between control and shutdown system, modification of control room design and the provision for post LOCA sampling capability in the containment.

16 During commissioning, difficulties were experienced with equipment reliability. These problems mainly involved Romanian supplied equipment. The Western consortium managed the resolution of these problems often by replacing equipment. Of particular note, was the need to replace the Romanian supplied diesel generators sets with imported equipment. Deficiencies related to the construction were largely corrected through major programmes of piping/weld inspection and repair. All special safety systems and related

safety support systems were imported. From the collected information it appears that the commissioning of the plant was performed in a manner comparable to CANDU 6 plants in Canada.

Safety assessments and programmes for further improvements

17 The basic safety assessment of the plant is provided in the Final Safety Analysis Report, whose content is in line with the standard content of US and Canadian safety assessment documents. The probabilistic basis of a standard CANDU design is derived from a reliability analysis performed at system level to show compliance with established reliability targets. A level-1 PSA was developed in the past by national organisations. It is however still incomplete and not validated. The utility of Cernavoda NPP, in the framework of the new strategic plan for safety analysis, started an activity aimed at completing a PSA level-1 by years 2001-2. In the longer term it is planned to develop a level-2 PSA analysis that includes fire, flood and seismic hazards, and low power events.

18 At present there are some issues, applicable to CANDU 6 plants like Cernavoda, that have been addressed or are under discussion in Canada. These issues include fire hazard assessment, prevention of dangerous effects of secondary side pipe failure (control room habitability), clogging of containment sump filters, core cooling in absence of forced flow, hydrogen behaviour in the containment. For example, design changes for the sump filters are under evaluation at Cernavoda. The resolution of the above issues needs to be monitored both by the Operator and the Regulatory Body and an improvement programme established where necessary. At present in Cernavoda there is a continuous programme of plant modifications based on operational feedback. However, this modification programme and the possible improvement programme to address the issues discussed above, may be affected by the financial situation at the NPP (see § 21).

Operational safety

<u>Organisation, procedures, operation and maintenance</u>

19 Plant organisation and operational documents are based on the Canadian approach and culture. The document that replaces the traditional Technical Specifications for Operation is the Operating Policies & Principles (OP&P). This is a Canadian reference document, which defines the envelope for safe operation and also includes items related to the plant organisational structure. As a result of an agreement between CNCAN and the Utility, the OP&P now used in Cernavoda are more detailed than those used for similar Canadian plants.

20 The plant Human Resources Development Unit reports directly to the Station Manager. Training programmes have been established both for general topics and for specific job types. A training centre is available on the site, with a full-scope plant simulator, which is currently being adapted to the specific details of the Cernavoda plant. The training programme for operators now includes a supplementary programme developed together with the Polytechnic Institute. This complements the basic training for control room operators, which does not address engineering studies. A systematic training system based on the use of the simulator needs to be further developed.

21 Due to the general economic difficulties of the country, the Utility only receives a part of the payments due to it for the production of electricity. In addition, a large part of the income that is received goes to pay off the credit for plant construction. This is leading the NPP into financial difficulties.

Safety culture and quality assurance

22 The utility has developed a nuclear safety policy document, which covers both corporate and plant levels. It clearly states the overriding priority given to nuclear safety and the objective of the utility to promote a nuclear safety culture at Cernavoda NPP. It also sets out a number of policies for the achievement of good performance. On a yearly basis the Cernavoda management communicates strategic objectives of the NPP to its staff.

23 The Cernavoda NPP received an IAEA pre-OSART mission in 1994 and a WANO mission in August 1997. The WANO mission noted some positive points in the management processes, and some areas to be improved, such as maintenance, plant configuration control, training, and feedback from operating experience. Action has been taken which implements some of the recommendations, while others are still ongoing.

24 Following the transfer of operating responsibility to the Romanian utility, the on-site technical support from the Western Consortium was significantly reduced. In view of this, the current plant managers were selected from professional experts who had been working for the Cernavoda Project for several years. These received initial training in Canada and then on-the-job training while acting as deputies of the Consortium managers during the construction and commissioning phases. The plant management is aware that additional efforts are necessary to ensure that an adequate safety culture extends from the senior staff to all levels of plant personnel.

25 In Canada in 1997, an Independent Integrated Performance Assessment (IIPA) identified deficiencies in human performance and management at a number of Ontario Hydro stations. At the request of CNCAN, a systematic review of the IIPA recommendations was performed by the utility to ascertain their applicability to Cernavoda. The above mentioned WANO Peer Review also covered safety management aspects. It is planned to implement those recommendations that are applicable, but this will depend on budget availability. Both the utility and CNCAN are of the opinion that the most critical IIPA findings that led to the temporary shutdown of some Ontario Hydro units are already addressed or not applicable to Cernavoda.

26 A Plant Quality Assurance manual was issued in 1993. A revised version, based on current Canadian, Romanian, IAEA and ISO standards, is under evaluation by CNCAN.

Operational experience

27 A plant event database has been kept since start of commercial operation. These events led to a total of 7 unplanned shutdowns.

28 A new set of procedures has been in use since April 2000. These procedures will allow the systematic analysis of operational events with ASSET methodology and the assessment of external operating experience.

Emergency preparedness

29 There is an on-site emergency plan approved by CNCAN. In order to improve the evacuation routes from Cernavoda, a bridge is under construction over the Danube, although work on this has stopped because of financial problems. Periodic emergency drills and annual exercises are carried out at the plant.

30 At present there is no dedicated emergency centre available on-site or outside the NPP.

This is contrary to Western European practice.

National industry infrastructure for technical support

31 In Romania, engineering support for the nuclear programme is provided by the Centre of Technology and Engineering for Nuclear Objects (CITON), and research is carried out by the Institute for Nuclear Research (ICN). Both organisations have been supporting the national nuclear programme since the early 1970s. However, Western support and in particular Canadian consulting services are still necessary, particularly for engineering activities involving safety related equipment.

On-site spent fuel and waste management

32 Cernavoda NPP fuel is currently stored in the plant spent fuel pool whose capacity is sufficient for about 9 years operation. The project is underway for the development of an on-site facility for interim storage of spent fuel.

Low and intermediate solid wastes are currently stored in a facility located on the site. Its capacity is for 20 years of operation.

Conclusions

33 The Cernavoda NPP has a CANDU 6 reactor similar to those in operation at Gentilly 2 and Point Lepreau in Canada. The plant was constructed and commissioned under the responsibility of a Western Consortium (AECL, Ansaldo). The safety of the plant has been assessed in a complete safety analysis report approved by the Romanian regulatory body.

34 The NPP managers and the plant operators have a professional attitude and have assimilated a Western safety approach and culture.

35 The Government needs to consider the following:

- The current financial difficulties of the plant need to be solved. If not overcome they could seriously affect activities that are necessary to ensure and maintain an adequate level of safe operation,
- The national infrastructure for technical support and research needs to be improved. It is important that Western support (especially from Canadian experts) is made available when it is needed in the future.

36 The Cernavoda NPP needs to address the following:

- To confirm design safety margins against seismic events and the adequacy of fire protection,
- To monitor the resolution of specific safety issues addressed or currently under discussion in Canada for similar plants and to establish an improvement programme where necessary,
- To preserve in the longer term the current good level of qualification and safety culture of the plant managers. This safety culture should be extended to all plant personnel and to the necessary service and support interfaces existing in the country,
- To improve areas of plant operation such as accident management, emergency preparedness, training activity and feedback from operational experience,

- To undertake and complete the strategic plan concerning the development of PSA studies and severe accident strategy.

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2. IAEA IRRT Mission Report, February 1998.
3. Report of Romania to the Convention on Nuclear Safety.

SLOVAKIA

Chapter 1: Status of the regulatory regime and regulatory body

Status of the legislative framework

1. A new Atomic Act on the peaceful use of nuclear energy entered into force on 1 July 1998 and abrogated the previous Act of 1984. The nuclear regulatory authority (ÚJD) was established in 1993 as an independent state authority when Slovakia became an independent state. According to the Atomic Act, it is responsible for state supervision of nuclear installations, radioactive waste and spent fuel management, transport, nuclear materials, physical protection. It has a prominent role in the emergency preparedness and planning organisation in Slovakia. The ÚJD is not responsible for radiation protection or the supervision of the use of radioactive sources outside nuclear installations. The Atomic Act defines the competencies of the ÚJD for the licensing, assessment, inspection, and enforcement activities. Some overlapping of responsibility exist between the ÚJD and the occupational safety office, which may lead to conflicting requirements placed upon an operator.

2. The Slovak nuclear power plants are operated by Slovenské Elektrárne, a 100% state owned Shareholder Company. A partial privatisation could take place in the future. The legal status of the operator is well defined in the Atomic Act, which states that it is responsible for the safety of its installations. The Atomic Act also gives to the regulatory body the responsibility to deliver authorisations that have no safety significance. The regulatory body should be relieved from delivering them.

3. Slovakia is a contracting party to all the key international conventions dealing with nuclear safety.

4. The nuclear legislative framework in Slovakia is generally in line with Western European practice, even if some issues could be improved as indicated in § 2 above.

Status of the regulatory body and technical support infrastructure

5. The ÚJD Chairman reports to the government which, in practice, does not interfere with technical decisions. He has direct access to the Prime Minister and participates in the meetings of the council of ministers when the agenda includes a topic within the responsibility of the regulatory body.

6. The ÚJD is funded from the state budget. Following the recommendations from the RAMG exploratory mission in 1993, the ÚJD staff and budget were increased significantly. Taking into account Slovakia’s present nuclear programme, the ÚJD current financial resources are still not sufficient and the inclusion of the Slovak contribution to the Chernobyl shelter fund should not be considered as an increase of its budget. The ÚJD would better retain its experts if their salaries were more in line with those of the operator's staff. The technical competence of the ÚJD personnel is internationally recognised. The ÚJD has a staff of 82 and ÚJD has indicated that 5 additional experts would be desirable to fit with its development plans.

7. The ÚJD has the power to issue and withdraw authorisations. It also has the power to impose sanctions on the operators for any violation of the conditions of an authorisation. Significant decisions signed by the ÚJD Chairman such as plant shutdown can be appealed to court.

8. The ÚJD has good access to technical support of several organisations based in Slovakia and the Czech Republic. However, the same technical support organisations assist the operator and this may create a conflicting situation.

9. It can be concluded that, in general, the ÚJD status is comparable to that of regulatory bodies in Western European countries. On-going developments such as those on internal quality assurance will improve its effectiveness.

Status of regulatory activities

10. Since 1992, a number of national and international evaluations of the ÚJD have taken place. The recommendations of the various missions and support programmes were effectively used in the development of Slovak regulatory activities. The ÚJD takes an active part in international regulatory co-operation.

11. Significant effort has been made to issue regulations as a consequence of the new Atomic Act. At present, 9 regulations are issued out of a total of 16 planned documents. In addition to these regulatory documents, guidelines for the practical application of the regulations by the utility are starting to be produced. The ÚJD has adopted a pragmatic approach in the review process of these guides by introducing a one-year trial of the guide by the utility to integrate, in the final document, the experience feedback.

12. A rigorous licensing review is in place, based on a Safety Analysis Report established by the operator. A licence for a nuclear installation is not issued by the ÚJD but by the authorities of the region where the installation is located. Nevertheless, a licence cannot be issued without the formal agreement of the ÚJD. The licensing steps of siting, construction, operation, and decommissioning are set up in the Atomic Act of 1998. A statement on the decommissioning option is included in the environment impact assessment established at the beginning of the licensing process.

13. The safety assessment practice is well developed and was carried out through a bilateral programme with Switzerland. This activity is now funded by the ÚJD, at least in its 2000 budget: the ÚJD should be given the financial resources to continue this activity.

14. Based on the recommendations of Western nuclear regulatory authorities, the ÚJD has developed a comprehensive inspection plan to conduct routine and daily inspections, perform special inspections and event oriented activities, with a systematic recording of the findings. Inspection procedures are clearly understood and utilised. The inspection performance corresponds to Western European practice.

15. So far, the safety re-evaluation of an installation was made on a case by case basis. The ÚJD intends to introduce a periodic safety review system according to the international practice. To this end, a draft regulation is currently being reviewed.

16. The Ministry of Health is the Authority for radiation protection supervision. Some regulatory issues in the field of nuclear safety have implications in terms of radiation

protection and vice-versa. Both Authorities have established a memorandum of understanding so as to harmonise their respective regulations and actions.

17. The ÚJD has established event-reporting requirements for the licensee and has developed a system for analysis and feedback of operating experience from domestic events similar to common Western European practice. The ÚJD is also actively participating in the INES and international event reporting system.

18. In addition to its participation to the VVER regulator's forum, the ÚJD is part of an international agreement with the Czech Republic and Hungary to share the experience feedback gained at Dukovany, Bohunice, Mochovce and Paks.

19. In summary, the ÚJD has made significant progress over the recent years and has achieved a series of regulatory practices comparable with those of Western European nuclear regulators.

Emergency preparedness on governmental side

20. The National Commission for Radiation Accidents (NECRA) involves the various state bodies playing a role in case of the activation of an off-site emergency plan and gives advice to the local authorities. The first draft of the national emergency plan is expected by the end of the year 2000. Its review, which is being co-ordinated by the ÚJD, should be given high priority.

21. The ÚJD, whose Chairman is a member of the NECRA, is in charge of advising this Commission on all nuclear safety matters in case of an emergency situation. The ÚJD also reviews the on-site and off-site emergency plans from the point of view of nuclear safety.

22. The ÚJD is operating a well-equipped emergency response centre to acquire and process the technical information needed by the NECRA to advise the local authority in charge of managing the emergency situation.

23. Although the national emergency organisation was not yet formalised in a document at that time, a national emergency exercise involving the Bohunice plant, the ÚJD, the NECRA and the local authorities, was organised in 1997. More focused exercises were also organised in 1998 and 1999. Emergency exercises are foreseen to have a periodicity of 3 years in the national Emergency plan. As soon as this plan is issued, it is recommended to organise further emergency exercises in Slovakia. Slovakia has participated in INEX-2 international exercises.

Conclusions

24. The regulatory regime and regulatory body in Slovakia are comparable with Western European practices. The independence of the regulatory body from the organisations involved in the promotion of nuclear energy is fully established in the legislation, which also clearly specifies the prime responsibility for safety of the operator. A well-defined licensing system is in place. The regulatory body is well engaged in the state supervision of nuclear activities and the national emergency organisation is under preparation.

25. It is recommended that the government of Slovakia consider the following:

- The financial and human resources of the ÚJD need to be further increased. The salaries at the ÚJD need to be made comparable with those of the operator's staff,
- The ÚJD needs to be given the resources to maintain the independent assessment capability which was initiated under Swiss assistance,
- The adoption of the national emergency plan needs to be given a high priority,
- The Atomic Act should be amended to remove some duties of the ÚJD that are not directly dealing with nuclear safety.

26. It is recommended that the ÚJD consider how to ensure a clear separation between the technical support it receives and that provided to an operator.

Chapter 2: Nuclear power plant safety status

Data

1. On the two nuclear sites at Bohunice and Mochovce, the Slovak Republic has in operation the following six nuclear power plants owned by the Slovak state company Slovenské Elektrárne (SE):

NPP unit	Reactor type	Start of construction	First grid connection	End of design life
Bohunice V1:				
Unit 1	VVER-440/230	1974	1978	2008
Unit 2	VVER-440/230	1974	1980	2010
Bohunice V2:				
Unit 3	VVER-440/213	1976	1984	2014
Unit 4	VVER-440/213	1976	1985	2015
Mochovce:				
Unit 1	VVER-440/213	1983	1998	2028
Unit 2	VVER-440/213	1983	1999	2029

2. At Bohunice, a prototype gas cooled heavy water moderated reactor called Bohunice A1 was operated for a short time in the 1970´s. The reactor was permanently shut down in 1977 after an accident that led to partial core damage and it is now being decommissioned.

3. At Mochovce, the construction of two more units, similar to units 1 and 2, has been suspended (40-50% complete), and currently there is no schedule for their completion.

4. The Slovak government has decided to close the two units of Bohunice V1 in 2006 and 2008.

(i) Bohunice V1, units 1-2

5. The statements presented in this chapter regarding the safety of the Bohunice V1 plant are based on the general knowledge of VVER-440/230 plants summarised in Annex 2, on information provided by Slovak organisations (utility and safety authority), and on records of IAEA missions. The information is confirmed and complemented by the results of the WENRA task force to Slovakia from 12 to 15 October 1999, which focused on Bohunice V1.

Basic technical characteristics

Design basis aspects

6. The first two units on the Bohunice site are VVER-440/230 type nuclear power plants. Generic safety characteristics and safety issues of such plants are presented in Annex 2. Improvements have been carried out on both units continuously since they were commissioned. So far, more than 1200 modifications of various safety importance have been implemented, and this improvement process is continuing.

7. Based on the findings of a safety assessment in 1990, the Czechoslovak Atomic Energy Commission issued a list of urgent upgrading measures which have been implemented during the period 1991-1993 and they are known as Bohunice V1 small reconstruction

programme. During 1991-1992, a safety report for a large "gradual reconstruction" was completed for Units 1-2. Results of the review and assessment of this safety report were issued in 1994 in the form of ÚJD regulatory requirements. These were the basis for the development of a major safety-upgrading programme, known as Bohunice V1 gradual reconstruction programme. The aims of this programme were:

- To establish the reactor pressure vessel status,
- To improve the function of the confinement system,
- To demonstrate the ability of the plant to cope with Loss of Coolant Accidents larger than the Design Basis using conservative analysis up to 200-mm LOCA break,
- To demonstrate the ability of the plant to cope with the complete rupture of a main primary coolant pipe (a Beyond Design Basis Accident) using best estimate analysis,
- To improve the plant behaviour in response to internal and external hazards,
- To improve system and equipment reliability,
- To improve organisational and operational safety.

8. The revised design requirements provide a coherent target for safety improvement of the plant. Completion of the long-term improvement programme is expected in year 2000. By then, the safety level of the plants will have been significantly improved when compared with the standard VVER-440/230 described in Annex 2.

<u>Reactor pressure vessel and primary pressure boundary</u>

9. The current condition and the surveillance programme of the reactor pressure vessels appear to be adequate. Both reactor pressure vessels were annealed in 1993. Safety assessments, supported by measurements of weld impurity concentrations, indicate that further annealing will not be necessary up to end of the expected lifetime. Measures have also been implemented on Bohunice V1 to reduce the probability of a large primary break. The majority of the calculations required to demonstrate a leak-before-break case for the 500-mm and the 200-mm primary circuit pipework have been carried out. The leak-before-break case is supported by an in-service inspection programme and by suitable instrumentation to detect incipient leaks. Revised analysis of seismic loadings to take recent modifications into account is expected in the year 2000. This could lead to further minor plant changes. There is an overlap of safety arguments for the revised design basis which covers pipe failures up to 200 mm by conservative analysis and the beyond design base studies which covers pipe failure up to 500 mm by best estimate analysis. The leak-before-break case, which covers pipework from 200-mm to 500-mm diameter and primary circuit components (valves, pumps), supports both these studies. The risk of a large primary to secondary leak, as a consequence of a steam generator collector head lift, has been reduced by the use of a new sealing technology and specific in-service inspection. Thus it is considered that the integrity of the primary pressure boundary is safeguarded to an adequate level.

<u>Confinement</u>

10. Compared to the original design, the confinement capability has been upgraded by installing jet condensers and by improving the venting flaps and the leak-tightness by two orders of magnitude. Relevant experiments have been carried out on the behaviour of essential features like the jet condensers and the venting flaps to support the confinement analyses. According to the utility’s analysis for DBA and for 500-mm break LOCA (as BDBA), there is no large challenge to the confinement function. Regarding the hydrogen management issue, the related measures have been tailored to the level of hydrogen

production that would take place for calculated cladding heat-up during the postulated accidents (DBA and 500-mm break LOCA as BDBA). It is considered that a consistent approach has been followed to demonstrate the modified confinement capabilities against the postulated accidents. However, there are smaller margins with respect to the radiological confinement function compared to those of Western reactors.

Safety systems and hazards

11. Original deficiencies in terms of the capacity and separation of safety systems have been mostly corrected, and it is planned that the remaining deficiencies will be addressed during the year 2000. Extensive measures have been taken against fire risk. Regarding site seismicity, the final value of the peak ground acceleration has not yet been defined but a conservative high value has been assumed as a basis of the upgrading programme. Also improvements to address post-LOCA coolant re-circulation (sump filter clogging) have been implemented.

12. Isolated deviations from common Western practices have been identified. The most significant one is probably the lack of specific protection against the dynamic effects that could result from potential high-energy pipeline breaks. For instance, it is assumed that there would be no break in the main steam lines downstream the isolating valves, where the current mechanical calculations show only a small safety margin to a pipeline break in certain postulated abnormal operating conditions. Consideration of this type of break might call for installation of anti-whipping devices. The utility is aware of this problem and considers its solution firmly linked to the residual lifetime of the Bohunice V1 units.

I&C systems and emergency power supply

13. The original reactor protection system has been replaced with a completely new one that compares favourably with current international practices. The system is qualified in accordance with international standard IEC 880. A new post accident monitoring system has been installed that presents the most important parameters both in the main control room and on the emergency control panel. The re-designed emergency power supply system compares with most important international safety practices, and the system is properly qualified.

Beyond design basis accidents and severe accidents

14. Some limited preventive measures have already been implemented, and further actions for prevention and mitigation are planned once the current plant modernisation process has been completed. Therefore, with regard to beyond design basis accidents, the situation has been improved. A preliminary list of BDBAs including, in addition to 500-mm break LOCA, the total loss of steam generators feed water, small and intermediate primary breaks with loss of high pressure safety injection and station black-out has been already investigated by the utility. As a result, pressurizer safety valves have been replaced to allow the use of a primary feed and bleed procedure, and a source of power from the nearby hydro-plant at Madunice in the case of station blackout has been added. Further actions are still ongoing, e.g. the completion of the list of the BDBAs in the light of the PSA results and the definition of the dedicated measures (equipment and/or emergency procedures) in order to prevent core melt. Concerning severe accidents, there are no specific requirements yet because both the regulator and the utility have put emphasis on first implementing the DBA/BDBA prevention and mitigation measures to avoid core melt. This is considered as a reasonable approach, but development of a feasible severe accident management scheme would be a logical next step, though in the course of the confinement improvements, some solutions have already been considered on how to improve core melt sequence mitigation.

Safety assessments and programmes for further improvements

Safety assessment

15. A Safety Analysis Report was prepared for the V1 plant prior to the start of the long-term improvement programme. Following completion of this programme, a Safety Analysis Report, with a content similar to those of Western plants, was presented to the ÚJD in June 2000 for review.

16. A Probabilistic Safety Analysis level-1 was carried out before and after the Bohunice V1 small reconstruction programme, and will be repeated after completion of the long term improvement programme. The scope of this PSA has covered only full power mode but considers all important initiating events including internal fires and floods.

Programme for safety improvement

17. The long-term improvement programme should be complete in the year 2000. Thereafter, modifications will be implemented based on a case-by-case analysis.

Operational safety

Organisation, procedures, operation and maintenance

18. The organisation of the Bohunice plant, including Bohunice V1 and Bohunice V2, is similar to typical Western plants' organisation. The site Manager directs the operations division (two chief engineers, one for Bohunice V1 and one for Bohunice V2), the maintenance division, the economics and commerce division, the investments division, the human resources and services division and the technical support and safety division. Compared with Western European practices, it is considered that the organisational aspects and procedures used for the V1 plant are adequate.

19. The utility has been able to implement the V1 plant modernisation program as well as other necessary modernisation and maintenance activities. To ensure that maintenance is carried out efficiently, a maintenance Division has been set up with the necessary workshops, laboratories, equipment and tools. For on-the-job training, mock-ups facilities are also available.

20. Overall, the qualification of the plant staff appears satisfactory. A comprehensive training system is in place and a multifunction simulator is used. Exchanges with Western partners are on-going either through bilateral or through multilateral co-operation programmes.

21. Technical specifications for operation have been improved, and follow a typical Western approach. Implementation of revised Emergency Operating Procedures is planned for the end of the modernisation process. Improvements are also being made to procedures for normal operation.

Safety culture and management, quality assurance

22. Two safety committees are in place, one at the plant level, the second at the company level. Contacts with Western experts have helped to promote a safety culture. A QA system is in place, covering all the main activities, including the V1 improvement activities.

Operational experience

23. Systematic investigations of plant events and operational feedback are conducted by a

dedicated plant department. At the national level, investigations are also conducted independently by the VUJE institute and in some cases by the ÚJD.

Emergency preparedness

24. The on-site emergency plan is regularly updated and exercises are carried out periodically (quarterly and yearly, depending on the type of exercise). The level achieved is adequate.

(ii) Bohunice V2, units 3-4

25. The statements presented in this chapter regarding the safety of Bohunice V2 plant are based on the general knowledge of VVER-440/213 plants summarised in Annex 2, on the joint international projects (including EU TSOs focusing at specific technical features of the VVER-440 plants), and on information provided by Slovak organisations (utility and safety authority).

Basic technical characteristics

Design basis aspects

26. Units 3-4 at Bohunice are VVER-440/213 type nuclear power plants. General safety characteristics of such plants are presented in Annex 2. Since 1990, significant improvements have been implemented at Bohunice V2. These include for example the installation of in-service diagnostic systems, the renovation of instrumentation and control systems, the improvement of electrical systems, fire and seismic upgrading, and some improvements in operational safety such as the introduction of symptom based emergency operating procedures and a new generation of normal operation procedures.

Reactor pressure vessel and primary pressure boundary

27. The current condition and the surveillance programme of the reactor pressure vessels appear to be adequate. Annealing will not be necessary up to the end of the expected lifetimes of the plants. The leak-before-break (LBB) concept for 500-mm and 200-mm primary circuit pipework has been demonstrated. The LBB is supported by an in-service inspection programme and by suitable instrumentation to detect incipient leaks. Regarding primary to secondary leakage through the steam generator collector head, the same measures as for the Bohunice V1 units are either implemented or planned. Additionally a system for monitoring primary circuit leakage in the steam generators by nitrogen 16 activity in the steam has been installed. The integrity of the primary pressure boundary is therefore, considered safeguarded to an adequate level.

Bubbler condenser containment

28. The performance of the bubbler condenser system in case of Large Break LOCA has been verified in full-scope tests in the framework of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test results for Large Break LOCAs were reported in early 2000. There is still a need for detailed analysis of the experimental project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA). The leak-tightness of Bohunice V2 containments has been improved. However, the leak rate is still somewhat higher than those achieved at Western plants. Nevertheless, the containment internal pressure driving the leak would be effectively limited by the bubbler condenser function in case of design basis accidents, and the overall radioactive releases would not be higher than what is accepted within the EU.

Safety systems and hazards

29. In terms of capacity and redundancy, the safety systems are comparable to Western ones.

However, some shortcomings have been identified and are being addressed in order to achieve adequate reliability of safety systems in all operating situations. For instance, measures against sump filters clogging and measures for fire protection are being implemented in the year 2000. Necessary improvements (e.g., modification of steam generators feed water system) are planned to be implemented by about the year 2002. The seismicity of Bohunice was once again reconsidered in 1998. The related assessment and the proposed values have been assessed by the IAEA. Also, a seismic monitoring network is permanently in operation.

Beyond design basis accidents and severe accidents

30. The regional Phare assistance project PHARE 4.2.7.a/93 considered the issue of beyond design basis accidents and severe accidents in the VVER-440/213 units. Different accident sequences have been considered with different failures. This includes accidents such as ATWS, primary breaks with partial or total failure of emergency core cooling system, total loss of feed water, station blackout. Some preventive measures have been taken into consideration and mitigation measures are being implemented. A new regional Phare project started in June 2000 in order to assist the Czech, Slovak and Hungarian safety authorities to assess the proposed preventive and mitigation measures.

Safety assessments and programmes for further improvements

Safety assessment

31. In 1993, the ÚJD approved the use of US NRC RG 1.70, adopted on country specific conditions for the elaboration of innovated Safety Analysis Report for Bohunice V2 units. This innovated Safety Analysis Report was presented to the regulator in 1994, after 10 years of NPP operation. Following comments from the ÚJD, a new version was produced in 1997. Its content corresponds to what is generally expected in Periodic Safety Reviews in Western Europe, and it has been reviewed and accepted by the ÚJD as part of a rigorous licensing review. The chapter in the report related to accident analysis was reviewed by the IAEA. In addition, a Probabilistic Safety Assessment has been carried out for the plant, within the process of updating of the safety analysis report after ten years of operation. As for Bohunice V1, the scope of this PSA has covered only full power mode, considering all important initiating events, including internal fires and floods. In addition, low power and shutdown PSA study level-1 has been completed and reviewed by IAEA in 1999. Work on PSA study level-2 started in 1999.

Programme for safety improvements

32. A further extensive modernisation programme is planned for implementation between 1999 and 2006, with the major upgrades relating to safety being completed by 2002 (§ 29).

Operational safety

33. Information and conclusions presented above for the V1 units are generally applicable for the V2 units.

(iii) Mochovce units 1-2

34. The statements presented on the Mochovce plant are based on the preliminary results of an independent safety evaluation, which has been carried out by a consortium of Western European Technical Safety Organisations.

Basic technical characteristics

Design basis aspects

35. The Mochovce units 1-2 are the latest ones of the VVER-440/213 type nuclear power plants (see Annex 2). Compared to their predecessors, several modifications were included during the design phase. The most important of these are the use of higher quality equipment (e.g. a modern reactor control system, a new type of pressurizer safety valves, an upgraded feed water control system), and the improvement of systems used in accident situations (e.g. a new design for the steam dump system, an improved emergency feed-water system located outside the turbine hall, upgraded fire fighting water system, a primary circuit venting system).

36. However, some design weaknesses remained, and these were addressed in a nuclear safety improvement programme developed in 1995 for the Mochovce NPP. The programme, comprising 87 safety measures, was reviewed by Western European Technical Safety Organisations and has almost been completed. The remaining measures (e.g. completion of equipment qualification, site seismicity characterisation) are underway.

37. Compared to the original VVER-440/213 design, the safety level of the Mochovce plant has been significantly improved.

Reactor pressure vessel and primary pressure boundary

38. The condition and surveillance programmes of the reactor pressure vessels appear to be adequate. In addition to the measures implemented for the V1 and V2 steam generators, the collector heads have been replaced to reduce the size of possible leaks. The integrity of the primary pressure boundary is thus considered to be safeguarded to a level comparable to Western practices.

Bubbler condenser containment

39. The performance of the bubbler condenser system used at Mochovce containment has been studied in full-scope tests performed in the framework of the Bubbler Condenser Experimental Qualification project sponsored by the EU. The test and analysis results were reported in early 2000, and, together with the investigations financed by the plant, they demonstrate structural strength and adequate confinement of radioactive fission products in case of Large Break LOCA. There is still a need for detailed analysis of the EU project results and for complementary tests for other design basis accidents (Steam Line Break, Small Break LOCA). The leak-tightness of Mochovce units 1 and 2 containments in case of large LOCA is comparable to those of Western European containments. Due to the bubbler condenser function, the calculated radioactive releases after design basis accidents would not be higher than at many Western plants for similar accidents.

Safety systems and hazards

40. In terms of capacity, redundancy and separation, the situation is comparable to Western European practice. The ongoing programme of equipment qualification for accident situations is in an advanced stage. Concerning site seismicity, the final value of the peak ground acceleration has not yet been defined. The utility has already undertaken actions aiming at the evaluation of the site seismic characteristics. The results are expected during 2001. Depending on results, further upgrading could be necessary.

Beyond design basis accidents and severe accidents

41. The utility has already started to analyse Beyond Design Basis Accidents on the basis of a preliminary list including ATWS, total loss of steam generators feed water, small break

LOCA in coincidence with a total loss of high pressure safety injection, steam generator tube rupture in coincidence with steam line break. This list will be reassessed in the light of the PSA results. Concerning severe accidents, the utility has the intention to use the applicable results obtained from the regional Phare project PHARE 4.2.7.a/93.

Safety assessments and programmes for further improvements

Safety assessment

42. A Safety Analysis Report (SAR) was prepared prior to the start-up of unit 1 in 1998. Its content is consistent with the content of safety reports in Western Europe, and it has been reviewed and assessed by the ÚJD as part of a rigorous licensing review. Furthermore, an independent review of the SAR by a consortium of Western European Technical Safety Organisations has been performed. A Probabilistic Safety Assessment level-1 has been developed in two phases where the pre-modifications and post-modifications states of the plant are evaluated. The pre-modification state is the plant state before implementation, and the post-modification state is the plant state after implementation of the safety measures specified in the safety improvement programme. The pre-modification PSA has been assessed by a consortium of Western European Technical Safety Organisations. The post-modifications PSA will be completed in the year 2000. The scopes of these PSAs cover full power mode and consider all important initiating events, including internal fires and floods. Moreover, first results of probabilistic seismic assessment which includes curves of seismic risk will be available as part of PSA level-1 in 2000. These evaluations will be complemented by a shutdown state PSA planned to start this year.

Programme for safety improvement

43. The safety improvement programme presented by the utility and agreed by the ÚJD is almost completed (see § 36).

Operational safety

44. The organisation of Mochovce plant is similar to that one of Bohunice plant. In addition, the preparation for plant operation has benefited from extensive national and worldwide experience, and advanced methods were brought into use prior to the first start-up of unit 1. These include, among other things, the availability of a full-scope simulator for initial training of control room operators.

National industry infrastructure for technical support

45. Slovakia has quite a strong national infrastructure in the nuclear field because some important research institutes were allocated to Slovakia at the division of Czechoslovakia. The Power Equipment Research Institute (VUEZ) supports Slovakian plants in tests of containment sealing, condensation systems, safety system design, filtration and ventilation. The Nuclear Research Institute (VUJE) provides technical support in the areas of training and safety analysis. As part of the infrastructure, one could also include the links with the Czech Republic that has a long tradition of mechanical equipment manufacturing.

On-site spent fuel and waste management

46. Up to 1986, spent fuel was sent back to Russia for reprocessing and final disposal. From 1987, Bohunice V1 and V2 spent fuel has been stored in an interim storage facility. An extension is being built which will allow the storage of V1 and V2 fuel up to the end of their expected operating lifetimes. This extension is expected to be complete during the

first half of the year 2000. The storage at the reactor pools at Mochovce NPP is designed to allow storage of spent fuel for a period of six years. The construction of interim spent fuel storage is foreseen also at the Mochovce site.

47. Currently, all NPP waste is treated on the Bohunice site. Three bitumenization facilities, as well as fragmentation and decontamination units, are under operation. The Bohunice Radwaste Treatment Facility, including cementation, incineration and compacting installations, is now under commissioning.

Conclusions

48. Operational practices at all Slovakian nuclear power plants are consistent with those in Western Europe.

<u>(i) Bohunice V1, units 1-2</u>

49. The following conclusions can be made:

- The revised design requirements provide a coherent target for safety improvement of the plant. The utility has made significant progress towards establishing a new design basis and implementing the relevant measures. A Safety Analysis Report, similar to those of Western plants, was presented by the utility to the ÚJD in June 2000 for review. Some work remains to be done but no technical obstacles are foreseen and completion is expected in 2000,
- In order to achieve and demonstrate adequate protection against possible loss of coolant accidents, several measures have been taken. The engineering safety features have been extended to cope with a leak that is equivalent to the leak from a double-ended guillotine break of a 200-mm pipe. A 500-mm double ended guillotine break LOCA is however evaluated with best estimate assumptions aiming to demonstrate core melt prevention and adequate confinement performance,
- Compared to the original design, the confinement capability has been upgraded. It is considered that a consistent approach has been followed to demonstrate the modified confinement capabilities against postulated events. However, there are smaller margins with respect to the radiological confinement function compared to those of Western reactors,
- Very extensive measures have been taken against fire risk. Regarding the site seismicity, the final value of the peak ground acceleration has not been defined but a conservatively high value has been assumed as a basis for the upgrading programme. Also, improvements concerning post-LOCA coolant re-circulation (sump filter clogging) have been implemented,
- There are isolated deviations from Western practices regarding steam line break postulation and dynamic effects from broken high-energy pipelines. Consideration of this type of break downstream the isolating valves may call for installation of anti-whipping devices. The utility is aware of this problem and considers its solution firmly linked to the residual lifetime of the Bohunice V1 units,
- If a solution can be found to the concerns related to the confinement ability to cope with the double-ended guillotine break LOCA, the Bohunice V1 plant should achieve a safety level comparable to that of plants of the same vintage in Western Europe.

<u>(ii) Bohunice V2, units 3-4</u>

50. The following conclusions can be drawn:

- Since 1990, significant improvements have been implemented at Bohunice V2. However, in order to achieve adequate reliability of safety systems in all operating situations, necessary improvements have been identified and are planned for implementation in 2000 for the measures against sump filters clogging and for fire protection, and in 2002 for the modifications to the steam generators feed water system,

- The integrity of the primary pressure boundary is considered to be safeguarded to an adequate level,
- The performance of the bubbler condenser system used at Bohunice V2 containments has been studied in full-scope tests performed in the frame of the Bubbler Condenser Experimental Qualification project sponsored by the EU. However, as for all the bubbler condensers, there is still need for detailed analysis of the experimental project results and for complementary tests for Steam Line Break and Small Break LOCA. Although, the leak rate of Bohunice V2 containments is somewhat higher than those achieved by those of Western plants, the radiological consequences in case of design basis accidents would not be higher than those accepted for Western European reactors,
- Concerning safety assessment, the content of the Safety Analysis Report is consistent with what is generally expected in Periodic Safety Reviews in Western Europe. It is complemented by a Probabilistic Safety Assessment. With regard to beyond design basis accidents and severe accidents, some preventive measures have been taken into consideration and mitigation measures are being implemented,
- An extensive modernisation programme is planned for implementation between 1999 and 2006, with the major upgrades relating to safety being completed by 2002,
- Consequently, the safety of Bohunice V2 units seems generally adequate. Once the ongoing safety upgrades have been implemented (by about year 2002), the safety level of these units is expected to be comparable to what is commonly found at units of the same vintage in Western European countries.

<u>(iii) Mochovce units 1-2</u>

51. The following conclusions can be drawn:

- Compared to their VVER-440/213 predecessors, units 1 and 2 of Mochovce included several modifications during the design phase. However, some design weaknesses remained, and a dedicated nuclear safety improvement programme, including 87 safety measures, was developed for the Mochovce NPP in 1995. This programme, which is almost complete, was reviewed by Western European Technical Safety Organisations,
- The integrity of the primary pressure boundary is considered to be safeguarded to an adequate level,
- Considering the mechanical reinforcements which improve the structural behaviour of the bubbler condenser and all the analytical and experimental work that has been performed, Mochovce bubbler condenser is presently the one that has undergone the most scrutiny. The leak rates of Mochovce units 1 and 2 containments in case of large LOCA are comparable to those achieved in Western plants. Because of the bubbler condenser function, the calculated radioactive releases after design basis accidents would not be higher than those at many Western plants under similar accident conditions,
- The content of the Safety Analysis Report, prepared prior to the start-up of unit 1, is consistent with the content of safety reports in Western Europe. It has been complemented by a Probabilistic Safety Assessment which will be extended to take account of the plant modifications and initiating events during reactor shutdown states. With regard to beyond design basis accidents and severe accidents, preventive measures have been taken into consideration and mitigation measures are being implemented,
- Although some residual work (e.g. bubbler condenser qualification, Mochovce site

seismicity characterisation) is still needed to confirm all parts of the safety analysis, the safety level of Mochovce units is comparable to the safety level of the nuclear power plants being operated in Western Europe.

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SLOVENIA

Chapter 1: Status of the regulatory regime and regulatory body

The information given here is based on experience gained through bilateral and multilateral assistance and co-operation programmes such as RAMG and CONCERT, IAEA missions and programmes and other open sources. An expert meeting took place in Ljubljana in January 2000 attended by WENRA members, SNSA and the operator of Krško NPP.

Status of the legislative framework

1. When Slovenia became an independent state in 1991, the continuity of the legal system was ensured by adopting all relevant laws from the former Federation of Yugoslavia. The main nuclear Act, the 1984 Act on radiation protection and the safe use of nuclear energy is currently under review. A new law has been under development for several years but without much progress. The 1984 Act is supported by a set of second level regulations related to specific aspects of nuclear, radiation, waste and transport safety.

2. The Slovenian Nuclear Safety Administration (SNSA) was established at the end of 1987 as an independent body dealing with all matters concerning nuclear safety. The SNSA reported directly to the Government and to Parliament until a change in legislation in 1991, since when the SNSA has been reporting to the Ministry of Environment and Spatial Planning.

3. Several deficiencies have been identified in the 1984 Act and improvements are expected in the future law. The SNSA is involved in its preparation. Firstly, the responsibility for safety needs to be clearly assigned to the licence holder. In addition, the roles and responsibilities of all governmental bodies involved in the regulatory process need to be clearly defined. Also adequate provisions need to be established to give the SNSA the authority and resources to manage the independent safety assessment required in the licensing process.

4. It is noted that the use being made of the appeal process, which allows the operator to appeal to the minister on both administrative and technical decisions, may constrain the regulator and undermine his credibility, authority and independence. Such technical appeals, which could have safety implications, need to be reconsidered in the review process of the law.

5. The licensing procedure is generally defined in the 1984 Act, and further developed, including licensing requirements and conditions, in second level regulation. The SNSA has overall control of the licensing procedure, subject to the appeal process above.

6. The Krško plant was built based on equal investment by utilities from Slovenia and Croatia. The responsibility for nuclear safety remains in Slovenia. Since the breakdown of the former Federation of Yugoslavia, a proposed agreement is under discussion among the two countries. In the mean time the Slovenian Government has issued a decree to support the necessary investment programme for the plant modernisation. According to this Decree the Nuclearna Elektrarna Krško (NEK), owner of the Krško nuclear power plant, is transformed into a public company. Future operation of Krško without a final resolution of issues related to shared ownership may affect the plant financial situation, which could

have an impact on safety.

7. Slovenia has ratified all key international conventions related to nuclear safety and they are now part of the Slovenian legislation. A peer review of the Slovenian legislation and regulatory framework has been carried out by an IRRT mission of the IAEA.

Status of the regulatory body and technical support infrastructure

8. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The Director of SNSA is appointed by the Government.

   A Nuclear Safety Expert Commission (NSEC) was established by the 1980 Act on Performing protection against ionising radiation and on measures for the safety of nuclear plants and installations. This Commission has an advisory role to the Ministries, advising on important licensing issues and reviewing the SNSA annual report. The Commission is chaired by the SNSA Director and has representatives of different ministries, experts in nuclear and radiation safety from Slovenia, one representative from the NPP Krško and one expert representing the Croatian Administration.

9. The Ministry of Defence plays, through the Administration for Civil Protection and Disaster Relief (ACPDR), the co-ordinating role in the national emergency system. The Health Inspectorate plays an active role in civil emergency plans for responding to nuclear and radiation accidents.

10. The SNSA does not have a separate and independent budget from the Ministry. No specific provisions are provided in the law to define the mechanism to fund the SNSA from the State budget. The Act on Administrative procedure enables the SNSA to charge to the licensee for the expenses related to a certain licensing process with an appropriate justification. Fees charged to the licensee do not represent a significant addition to the budget. Following the recommendations of the RAMG exploratory mission, the salaries of SNSA inspectors were raised to the similar level as the utility personnel, but this did not apply to the rest of the SNSA staff. This has resulted in some SNSA staff, in particular young engineers, leaving after their initial training. The budget and financial situation of SNSA therefore needs to be improved.

11. The staffing level of the SNSA has evolved from 5 in 1988 to the present level of 37. Therefore, 11 of 48 permanent posts are currently vacant. The SNSA has an active training programme to increase staff skills. There is still a lack of qualified and experienced personnel in most critical areas. It seems that the SNSA needs to further develop capability to perform a thorough safety review and assessment.

12. The Slovenian legislation requires the applicants to submit, in addition to the safety case, an independent assessment, with positive results, performed by an authorised organisation. The licensee contracts and manages such activity. Due to its limited resources and capabilities, the SNSA widely bases its assessment and decision making on the independent reviews by the authorised organisation. To guarantee a genuinely independent assessment, the SNSA needs to be provided with the authority and the resources to contract external organisations. Procedures need to be developed to ensure that such organisations are independent from licensee in the activities contracted.

13. The SNSA is given by law enforcement powers, including the power to stop the operation of a nuclear facility, in case of non-compliance with regulations.

14. There is no unique technical support organisation and several organisations act as TSO depending on the issues being assessed. The main national TSOs are the Jozef Stefan Institute (JSI) and the Milan Vidmar Institute (MVI), which also provide technical support to the utility. The funding for work carried out by JSI and MVI for the SNSA, is provided directly by the utility. The relationship between the TSOs, SNSA and the utility needs to be clarified to ensure there are no conflicts of interest.

15. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, the budget and financial situation of SNSA should be improved in order to ensure a complete independent safety assessment capability.

Status of regulatory activities

16. The SNSA has the powers to propose new legislation and is responsible for preparing new laws and regulations. The SNSA is also responsible for issuing and amending licences for all nuclear facilities and performs regular inspections at those facilities.

17. Though the main responsibilities and functions of the SNSA are well understood by the staff and the licensee, the policy and criteria based on which regulatory decisions are taken are not always defined. The SNSA needs to define its regulatory requirements to allow it to make the licensing decisions. The SNSA strategy needs to include, when feasible, a predefined licensing process for major safety improvements.

18. The SNSA licensing and assessment is based on US NRC practice, and is performed by competent staff in their areas of responsibility. However, due to high workload combined with limited resources, the SNSA needs to develop further its technical capability to perform a complete independent assessment. In addition, an adequate system of internal quality assurance is needed. Further efforts to establish a better co-ordination of the licensing steps and the related assessment and inspection activities are needed.

19. The SNSA inspection programme needs to be strengthened through adequate resources and further development and implementation of a systematic process, based on licensee performance and more focused on safety relevant aspects. Some of the interfaces between the different Slovenian Inspectorates are not very clearly defined, in particular with regard to the Health Inspectorate, fire protection issues, emergency preparedness and physical protection.

20. There is no formal requirement for a periodic safety review of Krško NPP and this needs to be addressed in the revised legislation, regulations and technical guidance.

21. The SNSA has established a system for the feedback of operating experience, including event reporting and assessment of lessons learned form similar operating plants. The SNSA takes part regularly in international regulatory activities aimed at fostering regulatory co-operation. In particular, Slovenia has concluded bilateral agreements with countries operating similar types of reactors.

22. In general the SNSA operates according to Western practice and methodologies. However, the management of regulatory duties could be improved and the inspection programme needs further implementation of a systematic process.

Emergency preparedness on governmental side

23. Under the present legislation, the Administration for Civil Protection and Disaster Relief (ACPDR) regulates and supervises emergency preparedness regarding the protection of the public off-site. The new updated Slovenian national plan has not been finalised. No national full scope exercise has been conducted for six years. A comprehensive exercise is needed to confirm the co-ordination among all authorities involved in responding nuclear emergencies.

24. The SNSA acts as the independent expert governmental organisation, providing advice to the National Civil Protection Headquarters, and acts as the co-ordinator with neighbouring countries and with the IAEA. The SNSA approves the on-site emergency plan. The SNSA has no emergency response centre to perform its emergency function.

25. Attention has to be given to the integration of all interfaces and relationships between the various involved organisations. This integration needs to take into account the core competence of each institution on radiation protection and nuclear safety, to be consistent in the assignment of responsibilities.

26. The Krško plant is close to the borders of Croatia and two bilateral agreements are in place to ensure adequate co-ordination and co-operation by the authorities of both countries, the most recent ratified in 1999. The interface with the Croatian authorities should be reflected in the emergency plan and tested in an integrated exercise.

27. The SNSA needs to have the capability of providing independent information to the public in the case of an emergency. The distribution of information needs to be co-ordinated with other involved parties.

Conclusions

28. The SNSA operates, in general, according to Western practice and methodologies. Since 1987, when the SNSA was established, it has evolved and matured as a regulator, with a clear separation between regulation and promotion of nuclear energy. The SNSA has a staff of motivated and dedicated persons with competence in their areas of responsibility. The SNSA has been assigned most of the roles and responsibilities normally allocated to a regulatory body. However, there are some issues that need to be addressed.

29. It is recommended that the Government of the Republic of Slovenia consider the following suggestions:

- The existing legislation on nuclear and radiation safety is not fully in line with current Western European practice. The review of the existing legislation needs to be completed as soon as possible,
- The existing appeal process on administrative and technical decisions could have safety implications and may constrain the regulator and undermine his independence,
- The lack of a final resolution of issues related to shared ownership of Krško NPP may affect the plant long-term financial situation and have an impact on safety,

- The SNSA needs to be provided with the authority and the resources to contract external organisations providing support for independent safety assessment,
- The budget and financial situation of SNSA needs to be improved. An increase in SNSA salaries and an improvement in the financial stability of the organisation would help to retain staff and increase independent safety assessment capability,
- The national response to nuclear and radiological emergencies needs to be improved by implementing an integrated national emergency plan. Special attention needs to be paid to the interface with the Croatian authorities.

30. It is recommended that the SNSA consider the following suggestions:

- The SNSA needs to define its regulatory requirements to allow it to make the licensing decisions, in particular by establishing a predefined licensing system for major safety activities like the periodical safety review,
- The SNSA needs to develop further its own technical capabilities in order to be able to make better independent decisions.

Chapter 2: Nuclear power plant safety status

Data

1. Slovenia has one nuclear power plant located at Krško, which was built as a joint investment by the electricity utilities of Slovenia and Croatia:

NPP	Reactor type	Electrical power (MW)		Start of construction	First grid connection	End of design life
		Gross	Net			
Krško	Westinghouse 2-loop PWR	*707	*676	1974	1981	2023

*Values after steam-generator replacement and power uprating. Original design values were 664 and 632 MWe respectively.

2. Since the breakdown of the Former Federation of Yugoslavia, a proposed agreement is under discussion between the two independent Republics of Slovenia and Croatia to define the legal provisions relating to the ownership of the plant. Pending this agreement, the Slovenian Government promulgated a Decree in July 1998 on transforming NPP Krško into a public company named "Nuklearna Elektrarna Krško" (NEK), to assure necessary financial resources for safe plant operation. The founder of the company is the Republic of Slovenia.

3. The future operation of Krško NPP is also affected by an Energy Policy resolution accepted by the Parliament. According to the Energy Policy the energy sector will be privatised and an open market established starting from 2001. NEK will, however, remain a public company. The implication for NEK to operate in the market is however not clear at the moment and will require special attention to avoid any negative impact on the current level of safe operation.

4. The statements presented in this Chapter regarding the safety of the Krško plant are based on general knowledge of the Westinghouse 2-loop PWR, on information provided by the Slovenian organisations (Regulatory Body and Utility) and on the reports of IAEA and other international missions, in particular ICISA. The information was verified and complemented by the results of a visit of WENRA experts to Slovenia from January 25 to January 28, 2000.

Basic technical characteristics

<u>Design basis aspects</u>

5. The design of Krško is similar to other Westinghouse PWRs of the same type in the USA, Belgium, Switzerland, Korea and Brazil. The Angra 1 plant in Brazil was the reference plant for Krško.

6. The reactor coolant system comprises two parallel loops connected to the reactor pressure vessel. Each loop contains a vertical single-stage centrifugal coolant pump and one vertical U-tube steam generator.

<u>Reactor pressure vessel and primary pressure boundary</u>

7. RPV surveillance is performed regularly and no operational problems have been identified.

No RPV embrittlement problems are foreseen at the moment. The reactor vessel was manufactured by Combustion Engineering using a low-copper base material. The good characteristics of the steel and the results of the regular monitoring programme (tests on samples of irradiated steel initially placed inside the RPV) have led the Krško RPV to be classified in the "no problem" area, according to US NRC R.G. 1.99 rev.2. The next sample will be tested in 2001 and is expected to confirm the positive NDT trend observed so far. Leaks have developed during past years in the original steam generator (SG) tubes due to stress corrosion cracking, and consequently more and more tubes were plugged. In 1999 the plugging level almost reached 18%. The leakage of SG tubes has caused an increase in the quantity and activity of Low and Intermediate Level waste. The SGs were replaced with ones of a new design in the regular outage of spring 2000. This made possible a 6.3% power up-rate. As part of the process of SGs replacement and power up-rate, complete mechanical and structural re-analyses and evaluation of the Reactor Cooling System (RCS) design were performed by the utility and approved by the regulator. The utility performed an analysis, according to the applicable US guides, for the application of LBB to the reactor coolant system, aimed at reducing the number of pipe restraints. The application of this is still under discussion between NEK and SNSA.

Confinement

8. The reactor and primary coolant system, including the steam generators, are located inside a double containment, which consists of a cylindrical steel shell with a hemispherical dome, an annulus and a surrounding reinforced concrete shield building. A negative pressure is established in the annulus between the primary and the secondary containment. The measured leak-tightness is similar to that of NPPs in Western European countries.

   The containment design presents margins against BDBA. conservative assumptions are used in the analysis of radiological effects and consequences of LOCA. It can be concluded that the containment function is comparable to that of Western European reactors of the same vintage.

Safety systems and hazards

9. In general, the safety systems are based on two redundant trains. Emergency core cooling is provided by one high and one low-pressure safety injection system and two pressurised accumulators. In case of a LOCA, re-circulation of the safety injection cooling water from the containment sump takes place. An evaluation of the clogging of sump strainers in the case of a LOCA has been made by the operator and is under review by the Regulator. On the secondary side, the auxiliary feed-water system consists of two separate redundant loops based on electrical and steam driven pumps.

10. Heat can be removed from the containment by means of the heat exchangers of the low-pressure injection system during sump re-circulation, and also by the containment fan coolers. The Component Cooling Water System and Essential Service Water System each consist of two redundant loops.

11. The seismic design of Krško is based on a maximum acceleration value of 0.3 g at the level of the foundations and on the design response spectrum recommended by the US NRC R.G. 1.60. Some aspects of the seismic characterisation of the site are under re-evaluation (see § 20).

I&C systems and emergency power supply

12. The reactor protection system is based on solid state logic arranged in two redundant trains

in accordance with US NRC R.G. 1.22 and IEEE 279. Qualification testing has been performed on the various items of protection system equipment. The emergency power supply has been upgraded and properly qualified to meet the requirements of US 10CFR50.63 which relates to station blackout events (see § 13).

Beyond design basis accidents and severe accidents

13. The Anticipated Transients Without Scram (ATWS) rule, US 10CFR50.62, has been implemented. Notably this includes the addition of ATWS Mitigating System Actuation Circuitry. Regarding the "Station Blackout" (SBO) event, a study has been performed by NEK using the US 10 CFR50.63 rule as reference. As result of the study an improvement programme, which included the upgrade of the DC sources and the addition of a nitrogen supply to important valves, was approved by the regulatory body and has been implemented.

14. Regarding severe accidents, a reactor vessel wet cavity strategy was implemented during the spring 2000 outage. In the case of LOCA the water in the containment would flow into and fill the reactor cavity. In addition, Severe Accident Management Guidelines have been developed using as a reference the generic Westinghouse owners' group guidelines but based on plant specific analyses and studies. They will be validated on the full-scope simulator, which is capable of simulating severe accident scenarios. The simulator entered into service before the spring 2000 outage. The issue of H_2 management is under discussion with the Regulatory Body.

Safety assessments and programmes for further improvements

Safety assessment and documentation

15. The licensing basis for Krško was a safety analysis report prepared by Westinghouse, as the main supplier of the plant. The content of SAR followed US practice. In 1991 the need was identified for a systematic review and the incorporation of design changes into the plant safety analysis report. That process led into the USAR (Updated Safety Analysis Report). The first revision was issued in 1992. Currently the USAR is a living document that is updated every year according to a written review procedure and is approved by the regulatory body.

16. A PSA has been performed at levels 1 and 2. The main contributors to core damage frequency are internal events, seismic events and internal fire. A PSA for shutdown conditions has also been performed. The findings from the integrated PSA studies were used in the development of the plant safety improvement and modernisation programme. The most important applications of PSA have been the Fire Protection Action Plan (FPAP) and the safety assessment of steam generator replacement and power up-rate. A model "NEK 2000" has been developed to represent the future risk profile of the plant after the implementation of the planned modernisation programme.

17. Since the start of operation, many safety requirements have been established by the SNSA. In addition, recommendations have been made by IAEA and WANO missions, and by the ICISA International Commission (1992-93). Most of these have been resolved and some are currently being implemented. The applicable post-TMI safety requirements have been implemented.

18. The safety related improvements already implemented can be grouped as follows: Post TMI related modifications, instrumentation improvements, improvements to enhance plant protection against special events like ATWS and Station blackout, PSA driven

improvements, development of a fire protection action plan, and major equipment replacement or upgrade.

Programmes for safety improvements

19. A large improvement programme was completed during the spring 2000 outage which included replacement of both steam generators and implementation of associated modifications, testing and validation of the full-scope simulator, implementation of remaining recommendations suggested by the fire hazard analysis, modifications related to the implementation of severe accident strategy (e.g. wet cavity). In order to license these modifications a complete set of safety analyses has been performed. They were independently reviewed by authorised organisations according to the Slovenian licensing procedure and submitted to the SNSA for their review and approval.

20. The seismic qualification of safety-related components is currently under review. Additional geophysical, geological, and seismological investigations in the area surrounding NPP Krško are being performed. Under a PHARE project, an investigation of the site seismicity is being carried out and a seismic monitoring network will be established around Krško. Based on the new structural-tectonic data, the Probabilistic Seismic Hazard Analysis (PSHA) which has already been carried out may need to be revisited.

21. It can be concluded that Krško is undergoing a continuous process of review and safety assessment. A living programme of verification of the compliance with the relevant US Regulatory requirements is in place.

Operational safety

Organisation, procedures, operation and maintenance

22. The site organisation, staff numbers, qualification and training of the personnel are similar to those of Western European NPPs. All the activities directly related to the plant operation are supported by independent review functions reporting at different levels in the organisational structure. In particular the Krško Safety Committee advises the General Director. The Safety Committee is composed of 11 members, 6 of which are external to the plant. It provides an independent review in several areas connected with plant operation and safety, including hardware and procedure changes.

23. At the moment no major financial problems are reported to exist by the NEK management. The need to pay appropriate attention to the NEK needs is recognised by the Government. The Decree establishing the Krško NPP as a public company has allowed the financing of the modernisation programme in the absence of a partnership agreement with Croatia.

24. The plant operational limits and conditions are provided in the Technical Specifications, changes to which are subject to approval by the SNSA. The content and style of the Technical Specifications follow US practice and are similar to those used in some Western European plants.

25. The operating procedures are reviewed and updated every two years in accordance with written procedures. A full set of Abnormal Operating Procedures and Emergency Operation Procedures have been developed and verified during simulator training. In 1988 symptom oriented emergency procedures were implemented.

26. The plant modification procedure is based on current Western practice that categorises

changes according to their safety relevance. Although the utility has its own engineering and technical support for safety assessment of plant modifications, it also relies on external national and international support. All design modifications are reported in advance to the regulatory body. Once implementation is completed, a documentation update is performed by NEK to reflect the changes.

27. Until now, training and retraining of licensed operators has been performed on simulators in the USA. As result of the availability of a full-scope plant specific simulator on the site the Krško NPP takes full responsibility for operator training in the year 2000.

<u>Safety culture and quality assurance</u>

28. A clear commitment to safety exists at management and staff level. A good management style seems to be in place with motivated and competent managers and staff. The existing stability of the personnel and the average age are good bases to preserve and to improve for the future the way the plant is operated. A proactive approach to safety improvements has been noted and the plant safety status seems to be under control.

29. During the last years the Krško NPP management has shown a policy of openness to international peer review. Relevant efforts have been devoted to accomplish the recommendations formulated by these international missions.

30. The Krško Quality Assurance Programme is implemented according to US requirements (10CFR50, Appendix B) and other international standards.

<u>Operational experience</u>

31. During the last three years, the average number of automatic reactor trips per year was below one.

32. The plant has an Operating Experience Assessment Programme, which analyses events and experiences and provides feedback. There is a specific group in the organisation that addresses these issues. Plant personnel are encouraged to report all in-house deviations and to maintain a correct regard for nuclear safety. The operating experience programme has been reviewed by IAEA and WANO missions. Although the feedback of operating experience programme is sufficiently complete, more contacts by NEK with European utilities would be beneficial.

<u>Emergency preparedness</u>

33. The Krško NPP is responsible for on-site emergency planning and maintaining on-site emergency preparedness. In developing its arrangements, Krško has made reference to IAEA guides and standards, and also to regulations and guides from the US NRC. Slovenia has participated in the INEX exercises organised by the OECD/NEA. In 1999 the on-site emergency plan was upgraded taking into account a new off-site emergency facility and severe accident emergency guidelines.

National industry infrastructure for technical support

34. The utility operating Krško NPP operates only that one unit. The plant engineering department cannot provide all the necessary technical assessment and engineering services, and support is therefore provided by outside organisations. Part of the required technical support can be provided by national organisations such as the Jozef Stefan Institute. Improvements need to continue in the development of plant engineering capabilities.

35. Because of the small size of the utility and the limited availability of national technical support, it is important that the NPP continues to maintain close contact with vendors and utilities associations to keep up with the state of the art and with general improvements in the field of nuclear and radiation safety.

On-site spent fuel and waste management

36. Spent fuel is temporarily stored on-site in a deep pool. At the current annual discharge rate the spent fuel pool capacity will be sufficient up to the year 2003 but NEK has initiated an action plan to increase the pool capacity by partial re-racking and the installation of new racks. This will extend the storage capability until the end of operational life of the plant in 2023 and beyond.

    The low level and intermediate level waste storage at the Krško NPP was designed only for the temporary storage of five years, originally having in mind the provision of designed waste treatment systems. The available storage capacity at the NPP was 90% occupied by the end of 1998. A significant volume reduction of stored solid radioactive has been achieved which will provide enough space for all the waste that is estimated to be produced during the operation life of the plant.

37. For bigger components (steam generators) a new multipurpose building was built for safe storage until the decommissioning of the plant. This building also contains a decontamination area and storage for solid radioactive waste generated during steam generators' replacement.

Conclusions

38. The safety of the Krško plant is comparable with that of NPPs of the same vintage in operation in Western European countries. The safety of the plant has been analysed and is documented in a complete safety analysis report.

39. A continuous improvement programme has been implemented in the past and a large modernisation programme has been recently completed. This includes the replacement of the steam generators, the completion of the fire protection improvements, the on-site verification of the full-scope simulator and the implementation of a severe accident strategy.

40. The utility operates only this single nuclear unit and, being relatively small, needs to continue to maintain contacts with outside organisations in order to receive adequate support.

41. The site organisation and the operational safety practice are comparable to those of Western European NPPs.

42. The following issues needs to be addressed:

- The implications on safety of the long-term ownership and the forthcoming privatisation process of the energy sector need to be carefully evaluated,
- The on-going evaluation of a few issues (e.g. the seismic characterisation of the site, clogging of containment sump, management of hydrogen) needs to be completed,

- Further attention is considered necessary regarding the spent fuel storage, given the residual capability of the pool and the need to license the related modifications,
- Improvements need to continue in the development of plant engineering capabilities to verify the deliveries from design organisations and suppliers. Contacts with vendors and utilities associations are recommended to be maintained and reinforced with particular regard to the relationships with Western European utilities,
- A programme to perform a periodic safety review needs to be finalised.

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ANNEX 1

Generic safety characteristics and safety issues for RBMK plants

Status of safety documentation

1. Western knowledge of the design characteristics of RBMK reactors has increased considerably since the Chernobyl accident in 1986, and especially after 1991 through the IAEA extrabudgetary programme on the safety of RBMK nuclear power plants as well as through other multilateral and bilateral assistance projects. Full in-depth safety analyses, based on independently validated computer codes, according to Western standards, have not yet been completed for any RBMK reactor, although the in-depth safety assessment made for Ignalina NPP, has resulted in much better safety documentation than for the other RBMK reactors. In general, however, Western expert knowledge of these reactors is not as deep as it is for Western designs.

2. The 14 RBMK reactors in operation belong to three different design generations, built to comply with different generations of Soviet safety requirements. There are considerable differences between the different generations of RBMK reactors and even significant differences among reactors within the same generation. It was a conclusion from the 1994 International RBMK Safety Review that, in order to get an accurate assessment of the safety level, it would be essential to perform plant-specific safety studies, including a Probabilistic Safety Assessment, for each reactor using state-of-the-art computer codes and methodologies. However, the basic features of the core design, the reactor cavity and the design of the main circulation circuit are common to all RMBK reactors. This implies that some specific safety issues are common to all units. These issues have been addressed to a varied extent by the RBMK operators.

Confinement issues

3. A most important safety issue regarding mitigation of accidents is the lack of a confinement or the lack of a complete confinement of the main circulation circuit, depending on design generation. A feature, common to all design generations, is that the RMBK reactor core is enclosed in a separate cavity, which is designed to handle serious damage to a very limited number of the 1661 fuel channels. Accident sequences, involving a hypothetical failure of the reactor cavity, would lead to consequential fuel failure and a release of radioactivity through the bypass with possible unacceptable environmental consequences.

4. The more modern RBMK designs have an Accident Localisation System (ALS) consisting of leak-tight compartments enclosing the parts of the main circulation circuit that are considered the most important. These plants have a pressure suppression capacity to deal with a loss of coolant accident within the ALS and well-diversified emergency core cooling systems. However, even in the most modern RBMK designs, the upper sections of the pressure tubes, the steam-water lines, the steam drum separators and the upper parts of the downcomers are not included in the ALS. A rupture in these parts of the main circulation circuit could lead to a loss of coolant that would not be possible to isolate. The first generation designs have no ALS and hence their emergency core cooling systems has fewer lines and fewer pumps, since there is no emergency core cooling re-circulation from a suppression pool. The ALS is tested for leakage every four years according to procedures. Normally the real leakage is much higher than the design leakage, even if there is a

compliance with national regulations on environmental impact. The performance of the ALS has not yet been fully validated in any plant to the extent normally required for Western reactors. It can be concluded that RBMK reactors do not have design features comparable with those required in Western European light water reactors regarding the last barrier for protection of the environment in the case of an accident.

Core characteristics

5. The Chernobyl accident highlighted the original design characteristics of the RBMK core with its positive void coefficient and demonstrated an inefficient performance of the shut down system. After the accident, a series of design changes were agreed by the Soviet authorities aimed at reducing the risk of a reactivity-induced accident. Some of these changes were felt to be so urgent that they were implemented in all plants. It was planned that other changes would wait until the mid-life refurbishment of the reactors. The main design changes involved the reduction of the positive void coefficient, improvements of the reactor protection system and display of the reactivity margin. The decrease in void coefficient was to be achieved by increasing the fuel enrichment from 2% to 2.4% and by the introduction of 80-100 additional absorber rods in the core. In the technical specifications, the number of effective equivalent control rods required in the core was increased from 26 to 48 (operational reactivity margin). These changes have reduced the void effect to less than 1 β. Furthermore the reliability and speed of the shut down system has been improved by modification of the control rod and rod drive design and installation of a new fast acting scram system with 24 rods.

6. These improvements were thoroughly evaluated and monitored in the IAEA Extrabudgetary Programme. It was established during the Programme that the two installed shutdown systems could not be regarded as fully independent and diverse. Furthermore, the fast acting scram system cannot maintain the reactor in a sub-critical state in the event of a loss of coolant accident in the control and protection system channels. Such an accident was considered by the designer as having a too low probability to be considered. However, it was agreed that in order to ensure a satisfactory reliability of the reactor shutdown function, backfitting of an additional completely independent and diverse shut down system was necessary in all RBMK reactors.

7. An important safety issue is related to the operational reactivity margin (ORM) since the ORM has to be controlled in order to maintain the void reactivity coefficient, the effectiveness of the shut down system and the power distribution within the given safety limits. In the original design it was the responsibility of the operator alone to keep the ORM within the safety limits.

8. A new type of fuel with burnable poison has recently been introduced in most RBMK reactors, giving more stable core characteristics, without additional absorbers, and has greatly reduced the need to constantly control the power distribution in the core.

9. The general complexity of the large core, with strong spatially dependent interactions between thermal-hydraulics and neutronics, puts a particular burden on the Instrumentation & Control systems. The need for several localised core control systems requires powerful computing systems to process the necessary operational data for control and protection. Although the situation differs from site to site, a generic safety concern is the status of the main computers in the RBMK plants. At Ignalina NPP, however, the main computer of unit 1 was recently replaced with a new American one and the same replacement is planned for unit 2.

10. Complex 3-D codes are necessary for calculation of core dynamics. Although some efforts have been made to develop 3D tools, further efforts are needed to develop 3D computer codes with adequate thermal-hydraulic feedback to properly take into account the spatial integration of the neutron fields with the fields of temperature and water density in the core.

11. The safety analyses done for the RMBK reactors raise some credibility concerns because the computer programmes used for reactor core transient analysis are not validated as thoroughly as the respective programmes used for light water reactor cores, including VVERs. Also, the calculations are very difficult due to the complicated core structure. Additional validation is necessary and analyses need to be performed of uncertainties in plant data and calculation methods.

Redundancy, diversification and separation of safety systems

12. In the later RBMK designs there is a high redundancy in most of the first line safety systems. This is however not the case to the same extent in the supporting systems, such as the service water and intermediate cooling systems. The high level of redundancy in the safety systems cannot always be given full credit due to potential common cause failures. It has also been found that the differences between the plants are so important that the evaluation has to be done on a specific basis.

    A major generic safety concern, however, has been the segregation of the electronic systems and the level of diversity in the most important systems and equipment. For example, the flux control system shares many common elements with the shut down system. The emergency core cooling system is actuated by a combination of signals and is not sufficiently assured that the system responds promptly or that the actuation equipment has a low probability of common cause failure. Also the lack of separation in electrical supply and of the emergency core cooling pumps has raised concerns about the sustainability to area events like fire and flooding.

Primary circulation circuit characteristics

13. The specific RMBK core design, consisting of graphite bricks penetrated by 1661 fuel channels and a number of CPS channels, raises a number of issues. The large mass of graphite (2 000 tons) provides a good heat absorbing capacity but has a disadvantage in a severe accident regarding its flammability which was clearly demonstrated at Chernobyl.

14. The primary circuit includes a few large pressure vessels distributing the coolant flow to a number of smaller vessels and to a large number of parallel pipes connecting each core channel. The system also includes a large number of valves. The design of the primary circuit creates some problems:

- The possibility of blockages, especially blockage of a group distribution header that distributes the flow to about 40 fuel channels. The operating history of RBMK has shown a few blockage incidents, which fortunately did not develop into serious events. Flow blockage is a large contributor to the risk of a severe accident,
- Material degradation due to the large number of pipes and welds. The RBMK pressure circuit suffers from similar material problems and degradation mechanisms, especially intergranular stress corrosion cracking that have been seen in Western BWRs. A large number of defects have consequently been found in RBMK pipework.

Dynamic effects resulting from double-ended guillotine breaks of large diameter high-energy piping, have not been considered in the design. A successful application of the LBB concept is therefore considered to be of a high priority, where intergranular stress corrosion cracking (IGSCC) can be precluded. Actions to address the IGSCC issue for RMBK plants have been initiated for all operating reactors but are not yet complete. International co-operation to address this issue, which is now under control in Western type BWRs, is regarded as important and urgent. A "break preclusion" programme could be developed, for IGSCC sensitive RBMK piping, which has similar elements as in the LBB concept and such a programme is under way at Ignalina NPP. However a level of safety similar to that provided by LBB could not be reached.

15. The RBMK has certain design advantages over other reactors. For instance, there is about double the water inventory of that in a typical Western BWR, while the fuel ratings are about 75% of those in a BWR and about 60% of those in a Western PWR. These features play a significant role in determining the slow heat-up of fuel in many accident scenarios. On the other hand, the large water inventory means that there is also more stored energy to be handled by the confinement and pressure relief systems.

Gap closure issues

16. A specific RBMK ageing issue is the gas gap closure. The pressure tube in each fuel channel is supported inside the channel in the graphite block by a series of graphite rings. It is arranged so that, at the beginning of plant life, there is a gap of 3 mm between the graphite block and rings. In this gap, a mixture of helium and nitrogen is circulated to improve the heat transfer from the graphite to the coolant and to monitor the tube integrity. Under the influence of irradiation during normal operation, this gap slowly reduces. There is no safety justification for continued reactor operation after the gap has reduced to zero, since this is not allowed by the designer. It is not clear if continued operation will challenge the integrity of the pressure tubes but in any way, it could make re-tubing impossible. The average time to expected gap closure varies upwards from about 15 reactor years, depending on operating conditions and on specific material properties of pressure tubes and graphite blocks at each reactor unit. Mid-life re-tubing was foreseen in the RBMK design and has been carried out at Leningrad units 1 and 2 (partially in units 3 and 4) and in Kursk units 1 and 2.

Operational safety

17. It has been concluded in the international reviews that an upgrade of the operational safety is of utmost importance for the improvement of nuclear safety in the operating RBMK plants. Improvements have been recommended to be implemented in parallel with proposed design related safety improvements. There should be a balanced approach to the allocation of resources to both design and operational safety areas. The main recommendations given in the international reviews are associated with the following:

- Clarification of the management structure including responsibilities, authorities and accountabilities at all levels,
- Development of Quality Management, including independent audits and audits of suppliers. Important issues to improve have been documentation management, plant modification control, investigation of events and experience feedback and improvement of the surveillance and testing of plant functions and components,
- Enhancement of the safety culture including promotion of trust and openness,

qualification improvement, self-evaluation and self-critical attitude. Also improvement of working conditions such as procedures for normal operation and emergencies, equipment labelling, housekeeping, improvement of lighting and the access conditions for operation and maintenance,

- Improvement of the training programmes, facilities and materials, introduction of continuous training and regular training of control room operators on full-scope simulators,
- Improvement of the maintenance planning and control,
- Establishment of an ALARA-programme.

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ANNEX 2

Generic safety characteristics and safety issues for VVER plants

1. The first VVERs were built at Novovoronezh in Russia and at Rheinsberg in Germany. The two units at Novovoronezh were rated at 197 MWe and 336 MWe, and operated between 1964-1988 and 1970-1990, respectively. Rheinsberg was rated at 70 MWe and operated from 1966 to 1990.

2. The first standard series of VVERs has nominal electrical power of 440 MW, and the second standard series has a power of 1000 MW.

3. There are two generations of the VVER-440 MW reactors, which are based on different safety philosophies. Of the older VVER-440/230 generation, there are 11 units still operating, while five have been permanently closed down. Of the second VVER-440/213 generation, there are currently 16 units operating.

4. In addition, two non-standard VVER 440 units have been in operation in Finland since 1977. In the contract for these plants, the Soviet vendor was required to meet Finnish regulations, which were based on US safety rules. The original VVER design was therefore modified by incorporating safety features that provide defence-in-depth against the same type of design basis accidents that are postulated for Western designed plants. The control and protection systems were designed and supplied by Western companies. Many vital mechanical components were also purchased from Western manufacturers. Plant layout, civil structures (including fire protection and ventilation systems), and electrical systems were designed by the engineering staff of the owner utility. Western type QA was applied throughout the construction project, including quality control at the factories within the former USSR.

5. In the VVER 1000 MW series, there is a gradual design development through the five oldest plants, while the rest of the operating plants, the VVER-1000/320s are quite similar to each other. In total there are 20 operating VVER-1000s.

Extent and validation of VVER accident analysis

6. In-depth safety evaluation of VVER-440 plants has been done in a number of countries both in the West and the East. This evaluation includes analysis of postulated transients and accidents with validated computer codes. Accident analysis of the Finnish VVER-440 plants has been carried out since the early 1970s by several Finnish teams and also by a German consultant.

7. The expected behaviour of the VVER-440 reactor core is confirmed by extensive data that has been collected during plant operation. For instance, at the Finnish plants the reactor core instrumentation and monitoring system is among the most comprehensive ones in the world's power reactors and accurate records exist from 43 reactor years of operation. In recent years, advanced monitoring systems with frequent automatic calculation of all important core parameters have been installed at many VVER-440 reactors. Studies of fuel rods irradiated at the Finnish reactors have confirmed the predicted fuel properties. These studies include hot cell investigations conducted in Sweden and tests to study fuel response to fast power transients, conducted both in the OECD Halden reactor in Norway, and in

the Studsvik reactor in Sweden. The ability to calculate the behaviour of the nuclear steam supply system during normal operation and small transients (such as reactor trip, reactor coolant pump trip and loss of feedwater) has been verified in extensive commissioning tests and by analysis of operational events.

8. The validation of accident analysis codes for VVERs has been carried out by several organisations in different countries since the mid-1970s. In addition to the generic data available from international standard tests, integral experiments conducted at VVER-specific thermal-hydraulic test facilities such as REWET and PACTEL in Finland and PMK in Hungary have also provided data for this validation.

9. The most recent comprehensive analysis of the Finnish VVER-440 plants was done in connection with periodic re-licensing in 1997, using a validated state-of-the-art computer code package. Independent calculations for verification of the analysis were done by the Finnish regulator and its consultants. Similar analysis has been done for other VVER-440s by competent teams in particular in Hungary and the Slovak Republic.

10. The transient and accident behaviour of the VVER-1000 reactor has also been investigated quite extensively. For instance, a feasibility study on the licensability of an improved VVER-1000 design was done in Finland in 1992. It included a full scope analysis of postulated design basis events. The analysis was updated by a Finnish team in 1995 to support an application to build a similar plant in China.

11. Other VVER-1000 analysis has been done by Western experts for instance in Germany for a plant, which was never completed, and for the Temelin plant under construction in the Czech Republic.

12. Operational experience analyses done by national regulators and reported widely by the VVER Regulators' Co-operation Forum have further improved the understanding of these plant types.

13. In conclusion, the accident analysis of both VVER-440 and VVER-1000 designs is considered sufficient to provide an adequate understanding of the generic safety characteristics of the plants.

VVER-440/230

14. In the EU candidate countries, there are six nuclear power plant units of this type: four in Bulgaria (Kozloduy 1-4) and two in Slovakia (Bohunice V1, units 1-2).

15. The design of the VVER-440/230 was based on the exclusion of a double-ended guillotine break of the main circulation line or the pressurizer surge line in the reactor cooling system. Instead, the accident assumed as the design basis for the safety systems was a break of a pipe directly connected to the main circulation lines. Following this basic assumption, all pipe joints to the main circulation lines were equipped with throttling devices. These would limit the maximum leak rate from any broken pipe that directly joins the primary circuit to the equivalent of a 32-mm diameter break. This size of leak was the basis for designing VVER-440/230 safety systems, and consequently the capacity of the originally installed emergency core cooling systems was very small. It also meant that the design did not feature a substantial, Western-style, containment around the reactor cooling system to limit potential radioactive releases in a medium or large break loss of coolant accident. The as-built confinement system of VVER-440/230s had little overpressure capability and its leak-

tightness characteristics were very poor. On the other hand, in post-accident conditions, this confinement would operate under sub-atmospheric pressure for a significant period of time, which of course would reduce releases in design basis accidents.

16. Although a large break in a reactor coolant circuit has never occurred at any nuclear power plant, a large break LOCA is generally postulated as a design basis for safety systems in existing Western nuclear power plants.

17. In addition to the limitations in the core cooling and confinement capability, the VVER-440/230 plants had two other major safety concerns:

- Internal hazards such as fires or floods, and external hazards such as seismic events or aeroplane crash, were not adequately considered in the original design. Thus the redundant parts of the safety systems were not adequately separated from each other, and were vulnerable to common cause failures. Some important safety systems were installed close to high-energy systems or in high fire risk areas (e.g. the turbine hall). Consequently, an event in one part of the plant could have resulted in complete loss of vital safety functions,
- The auxiliary systems, such as electrical power supply or cooling systems, which support the safety functions, were designed with inadequate redundancy. Consequently, a single failure in a critical component of an auxiliary system could have resulted in the loss of that support function and thus also a loss of the main safety functions.

18. Some additional safety concerns are common to all VVER plants being operated in the EU candidate countries:

- The original quality of electrical equipment and instrumentation & control equipment was inadequate, and the equipment was not qualified to function in accident conditions,
- The reactor pressure vessel wall is exposed to higher irradiation by fast neutrons than most Western designed reactor pressure vessels, and therefore the embrittlement of the vessel material proceeds more quickly,
- The design of the main barrier between primary and secondary coolant inside the steam generators (primary collector) is less robust than the tube sheet in Western PWRs, and the possibility of large primary to secondary circuit leak therefore needs to be taken into account in the design of the safety systems. For instance, primary to secondary leaks occurred in several steam generators during an event at Rovno NPP unit 1 in 1982. Steam generator primary collector covers broke off one after another as a consequence of careless operation and negligent maintenance. Despite very large leaks the Rovno accident developed slowly enough to allow operator intervention to prevent any core damage.

19. The safety concerns with VVER-440/230 plants are discussed in detail in an IAEA report [1]. All the plants have addressed these concerns to various degrees by backfitting and design changes.

20. When assessing the overall safety of the VVER-440/230 plants, it should be noted that, like all VVER-440s, they have certain inherent safety characteristics that are superior to most modern LWR plants. The principal safety characteristic of all VVER-440 plants is the large volume of coolant both in the primary and the secondary side. These reactors have more than twice as much coolant per megawatt as any Western designed NPP. These large coolant volumes and low core power density mitigate any anticipated transients so that the plant response to transients is very smooth. For instance, in all anticipated transients, the primary pressure stays well below the opening set point of the safety valves, and large

safety margins to heat transfer crisis in the reactor core remain. Large coolant inventories also allow multiple failures to occur without damage to the reactor core, e.g. interruption of all AC power supply to plant equipment for several hours, or a complete loss of heat sink for a similar time. This robustness is demonstrated by operational experience from two major blackout incidents, the Greifswald 1 fire in 1975 and Armenia 1 fire in 1982. These safety features provide an inherent protection, far more extensive than typical for Western LWRs, against the possible escalation of transients to more severe events.

21. Other significant inherent safety features are:

- Small and robust reactor core: any oscillations in spatial power distribution quickly die away, and do not require active control as in larger reactor cores,
- Low peak fuel temperatures with good retention of fission gases within the ceramic fuel pellets,
- Low heat flux from the fuel to the coolant giving a very large margin to critical heat flux in normal operation and during abnormal transients, and slow temperature increase in loss of coolant accidents,
- Robust design of main components and piping, including the main circulation lines of the reactor cooling system, which are made of austenitic stainless steel,
- The ability to isolate any failed loop of the reactor cooling system, and after isolation to bring the plant to safe shutdown using normal operating procedures,
- Risks concerning most of the Western PWRs during outages, caused by a temporarily reduced coolant inventory, are excluded in a VVER-440 because there is no need to decrease the water level in the primary circuit during a refuelling or maintenance outage. As the residual heat removal system is connected to the secondary side only, the likelihood of leaks in general and interfacing LOCA in particular are reduced compared to Western designs.

22. The following conclusions can be made regarding the safety of VVER-440/230 plants:

- The original plant design had inadequate systems to cope with accidents that are postulated as design basis of the Westerns PWRs. Due to this reason and also due to other concerns explained above, its safety was not acceptable to Western European standards,
- However, all VVER-440/230s being currently operated in candidate states have been significantly modified to varying degrees as compared to the original design. Where the modifications have been pursued most vigorously, a new design basis accident set has been defined to include up to 200-mm breaks, the emergency core cooling systems and confinement capability have been improved to deal with it, breaks beyond the new DBA have been ruled out by implementation of leak-before-break arguments, and confinement leak tightness has been improved by up to two orders of magnitude. The generic safety issues identified by the IAEA have been addressed to varying degrees at all the plants,
- It has been shown that it is possible to remove most of the safety concerns by refurbishment and backfitting, but it requires a major investment. However it does not appear feasible to backfit the plants with a reactor containment that could provide similar protection to the containments of modern Western PWRs,
- As to the original design requirement of Western PWRs for containment to keep the radioactive releases in connection with a large break LOCA and other design accidents below a specific limit, it seems possible to also meet this target at VVER-440/230s by a combination of upgrading of the existing confinement and installation of other supporting systems,

- The difference between the VVER-440/230 confinement and the Western style containment becomes more evident when analysing the capability to prevent the releases after severe reactor core damage. Although containment buildings of Western PWRs were not designed to cope with severe accidents, they can provide good protection by limiting releases to the environment in many of the investigated accident scenarios. This was well demonstrated in the TMI accident in 1979 in the USA. Conversely the VVER confinement system relies on systems to provide sub-atmospheric conditions to prevent leakage of radioactivity following core damage,
- The proven inherent safety margins and moderate response in connection with potential design basis accidents of the VVER-440 partly compensate the remaining shortcomings of an adequately upgraded confinement structure. There are parallels with the arguments used in the Western Europe to justify the lack of LWR-like containment in gas-cooled reactors. When the transients and accidents caused by equipment failures are less severe, and their rate of progress is relatively slow compared with Western PWRs, the operators have more time to take corrective actions. Although it is difficult to quantify the safety gain associated with this, there is ample operational experience that demonstrates its value. Worth noting are the two very severe fires that resulted in several hours loss of all key safety functions, 1975 in Greifswald and 1982 in Armenia, without resulting in the release of any radioactivity to the environment.

VVER-440/213

23. In the EU candidate countries, there are 12 nuclear power plant units of this type, four in Hungary, four in the Czech Republic, and four in Slovakia.

24. The accidents used as a design basis for the VVER-440/213 safety systems are similar to those postulated in Western plants, including a double-ended guillotine break of the main circulation line in the reactor coolant system. The safety systems are quite similar to those in Western PWRs. Mostly, they consist of three redundant parts, and any one of those parts can provide the intended safety function. This goes beyond many Western designed plants, which have only two redundant parts in their safety systems.

25. VVER-440/213 reactors have bubbler condenser type pressure suppression containments that in principle closely resemble Western boiling water reactor containments. The bubbler condenser is a unique Soviet design. Although its performance during design basis accidents had been studied analytically and with model tests both in the former USSR and in the Eastern European countries, there was a common desire among the international nuclear safety expert community to confirm the results with additional large-scale and separate effects tests. Large Break LOCA tests were conducted with Western support as the Bubbler Condenser Experimental Qualification Project sponsored by the EU. These included full-scale tests at facilities built in Russia, and complementary tests in the Czech and Slovak Republics. The test and analysis results were reported in the early 2000, and provide the necessary experimental evidence that the bubbler condenser is capable of withstanding the imposed loads and maintaining its functionality following a Large Break LOCA. An in-depth assessment of the reported results by independent safety organisations is in preparation, and also performance of large scale experiments for Steam Line Break and Small Break LOCAs, with corresponding pre- and post-test calculations, has been suggested. These would be required to increase confidence in the bubbler condenser performance in all accident conditions.

26. Another concern has been the containment leak-tightness. The leak rates measured in integral tests of containments, in their initially constructed conditions, were clearly higher at

some plants than is allowed for Western containments. Improvements in leak-tightness have now been achieved at all plants, although large variation among the plants is evident. However, it should be noted that comparison with Western containments is not straightforward because, in connection with the design basis accidents, the pressure suppression system tends to cause underpressure rather than overpressure at the time period when the atmosphere of the containment has its highest contents of radioactive aerosols, and when the potential for radioactive releases would thus be the highest. The behaviour of the bubbler condenser containment under severe accident conditions has not been investigated.

27. Compared with other major safety concerns of the older VVER-440/230 plants (cf. § 17), design improvements include:

- Internal and external hazards have been addressed to various degrees on a plant specific basis, and there are major design differences between the plants. There may still be some plant-specific concerns in this area, but to a lesser extent than for the VVER-440/230s,
- Protection against single failures in the auxiliary and safety systems has generally been provided by design, although improvements in detail have been required as a backfitting measure.

28. The safety concerns with VVER-440/213 plants are discussed in detail in an IAEA report and in a German safety evaluation [2], [4]. Most of these concerns have been addressed on a plant specific basis.

29. All the inherent safety characteristics discussed in connection with VVER-440/230 plants (see § 20 and 21) are equally valid for the VVER-440/213 type. Extensive model testing and safety analysis has been done in several countries, including recent analyses with state-of-the-art computer codes. These analyses have confirmed the safe behaviour of the reactor core and its cooling system in all abnormal transients. Furthermore, it has been confirmed that these plants can be brought to safe shutdown in connection with the accidents that are generally assumed as design basis events for modern nuclear power plants.

30. The following conclusions can be made regarding the safety of VVER-440/213 plants:

- The original safety targets set for the plant design were quite similar to Western European standards at the time when most plants in operation to day within the EU were constructed. However, the implementation of the design failed to pay enough attention to details, and several safety deficiencies could be identified in safety analyses done later on. Also the quality of some equipment did not properly correspond their safety importance. At all the plants, most of the safety deficiencies have been addressed by backfitting and plant modifications,
- A general issue that needed specific studies was the performance of the reactor containment during design basis accidents. Large scale Large Break LOCA tests were conducted with Western support as a joint industrial project. The test and analysis results were reported in early 2000 and are being assessed in depth. The performance of large scale experiments for Steam Line Break and Small Break LOCAs, with corresponding pre- and post-test calculations, is still required,
- Due to the robust original design, it is quite straightforward to upgrade the safety of the original VVER-440/213 design to a level comparable with the plants currently operating in Western Europe. The safety issues that need to be addressed have been identified by the

IAEA,

- As concerns protection against severe accidents that were not part of the original design basis of any of the operating Western PWR nor a VVER, the situation is as in the case of VVER-440/230 design discussed in § 22 above: containment capability to limit releases is expected to be somewhat inferior to the Western PWR containments, but much better than in VVER-440/230s. The inherent safety features compensate this shortcoming to a considerable extent.

VVER-1000/320

31. In the EU candidate countries there are two nuclear power plant units of this type in operation, both of them in Bulgaria. In the Czech Republic, two further units are being built that were originally of a similar design but have been extensively upgraded during construction.

32. The VVER-1000 plants were designed to similar safety requirements as Western plants and have equivalent safety systems. However, compared to the VVER-440/213 plants, the overall safety level of the VVER-1000 plants seems to be lower. The reason is that the higher power VVER-1000 plants have lost nearly all the inherent safety features of the smaller VVER-440 plants.

33. The main safety concern regarding the VVER-1000 plants lies with the quality and reliability of individual equipment, especially with the instrumentation and control equipment. Also the embrittlement of the reactor pressure vessel needs continuous attention and action will need to be taken if it approaches a hazardous level.

34. The main barrier between primary and secondary coolant inside the steam generators is a greater safety concern than in the VVER-440 plants, and it has been necessary to replace a number of steam generators when failures have been observed in this barrier. It remains to be demonstrated by further successful operating experience that design and manufacturing method improvements have solved these problems.

35. The plant layout has weaknesses that make the redundant system parts vulnerable to hazardous systems interactions and common cause failures caused by fires, internal floods or external hazards.

36. The safety concerns about the VVER-1000 plants are discussed in detail in an IAEA report [3], [5].

37. The following conclusions can be made regarding the safety of VVER-1000 plants:

- The original plant design had deficiencies, which would not be acceptable by Western European standards. At all plants, many of these deficiencies have been addressed by backfitting and plant modifications,
- It is feasible to upgrade the safety of the VVER-1000 plants to a level comparable with many of the plants being operated in Western Europe. This upgrading should adequately address all the safety issues identified by the IAEA.

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HERCA-WENRA Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident

Stockholm, 22 October 2014

The HERCA-WENRA Approach is an incentive approach that comprises the necessary mechanisms for countries to exchange adequate information and to achieve practical and operational solutions on a voluntary basis during an emergency leading to a uniform way of dealing with any serious radiological emergency situation, regardless of national border line, hence allowing for coherent and coordinated protective actions.

The HERCA-WENRA Approach has the potential to improve the coherence of the response in case of a nuclear accident with impact on territories of other countries and to be used as guidance to implement Article 99.1[1] and 99.2[2] of the Euratom-BSS. It also fulfils recommendation N°12.7.b of the so-known ENCO study and it further addresses some of the other recommendations.

[1] 1 Art. 99.1. Member States shall cooperate with other Member States and with third countries in addressing possible emergencies on its territory which may affect other Member States or third countries, in order to facilitate the organisation of radiological protection in those Member States or third countries.

[2] 2 Art 99.2. Each Member State shall, in the event of an emergency occurring on its territory or likely to have radiological consequences on its territory, promptly establish contact with all other Member States and with third countries which may be involved or are likely to be affected with a view to sharing the assessment of the exposure situation and coordinating protective measures and public information by using, as appropriate, bilateral or international information exchange and coordination systems. These coordination activities shall not prevent or delay any necessary actions to be taken on a national level.

Content

Heads of the European Radiological protection Competent Authorities

Western European Nuclear Regulators' Association

General Presentation of the HERCA-WENRA Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident

Stockholm, 22 October 2014

General presentation of the HERCA-WENRA Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident

The HERCA-WENRA Approach improves the response and cross-border coordination for all types of possible accident scenarios. It contains overarching principles based on radiation safety considerations and provides an incentive for joint actions between neighbouring countries. These principles need to be implemented at national level. Therefore additional conceptual work is needed, some of the principles need additional guidance, and other relevant stakeholders such as civil protection services need to be included.

HERCA and WENRA recognise that in European countries, efficient emergency preparedness and response (EP&R) arrangements have been established since many years and are tested and challenged regularly. They allow authorities to issue recommendations for effective public protective actions. In the development of such arrangements, each European state defines its own priorities and objectives in planning for nuclear emergencies directly affecting its own territory. The international and EU radiological protection frameworks leave large margins of freedom for setting national criteria for intervention. Emergency planning has evolved in all states over many years, mostly without giving great priority to cross-border issues. This has led to sometimes significant differences. Should a nuclear emergency occur in Europe, these differences can potentially have a significant effect, especially if the location of the emergency is close to a national border. Internationally, populations would feel unequally protected, depending on where they live. Agreeing or aligning protective actions along adjacent national borders is therefore highly desirable.

During the very early phase of any accident, the status of the reactor and the estimation of the amount of radioactivity released (the source term) are likely to be poorly understood. Thus, the uncertainties in terms of dose estimation and overall radiological impact are very large. The role of the decision-maker is to arrive at appropriate health protection measures possibly even without any dose estimation. This inevitably leaves room for flexibility in decisions, even where there is a rigid national framework. The HERCA-WENRA Approach makes use of this freedom for coordination between neighbouring countries in order to align early decisions across borders. As a result, the respective national arrangements do not necessarily need to be changed. Instead, the prevailing differences are respected and are taken into account, and the response is based on 'compromise' solutions which are understandable and explainable in each given situation.

HERCA and WENRA additionally consider that the possibility of a severe accident scenario (i.e. Fukushima-like) with no or insufficient information on the plant status cannot be completely ruled out. EP&R arrangements should therefore also cover such cases. However, in such cases, the recommendations of protective actions need to be formulated rapidly, leaving very limited time for cross border coordination during the first phase of the accident. Therefore, the HERCA-WENRA approach contains pre-defined simplified schemes for protective actions that may be applied in these cases, as improbable they might be.

The general mechanism of the HERCA-WENRA approach on EP&R for a better cross-border coordination of protective actions during the early phase of a nuclear accident is independent of the scenario of the accident.

<u>The approach relies on the following principles: shared technical understanding, coordination and mutual trust. It does not propose a uniform cross border framework. The main strategy is to aim at an alignment of the response between neighbouring countries or neighbouring territories.</u> This is supported by early information exchanges using existing dedicated bilateral and international arrangements as far as possible.

The HERCA-WENRA Approach is divided into 3 steps: the preparedness phase, the early phase and the later phase. The approach contains the main principles and leaves necessary margins of freedom for detailed implementation:

Step 1
In preparedness the aim is to achieve and maintain a shared understanding of the existing national emergency arrangements through the improvement of bilateral or multilateral arrangements, the testing of these arrangements and the implementation of improvements.

Step2
In the early phase of an accident, the proposed HERCA-WENRA Approach foresees rapid information exchanges by using existing dedicated bilateral and international arrangements, including the exchange of liaison officers as appropriate. If the response is thought consistent, the neighbouring countries can recommend to their governments to follow these recommendations, i.e. adopt the principle "We do the same as the accident country" in the first hours of the accident.

Step3
In the later phase a common situation report, accepted by all impacted countries, will further support coordinated protective actions.

The HERCA-WENRA Approach has been tested and validated against concrete and realistic accident scenarios in NPP's that are close enough to national borders. A workshop in September 2013 showed that in case of a sufficient information exchange most countries would be able to recommend to their decision-makers that the advice of the accident country should be followed during the very early stages.

For the initial stage of a highly improbable severe accident (i.e. Fukushima-like), requiring rapid decisions for protective actions while very little is known about the situation, simplified schemes for protective actions are needed.

Fukushima has shown again that a severe nuclear accident anywhere in the world, including Europe, cannot be completely excluded. Considering the safety level of European nuclear power plants and their improvements resulting from the lessons learned from various events (including the Fukushima disaster), it is estimated that the probability of such a severe accident is very low. But, as improbable such an accident might be, EP&R arrangements must be prepared for such cases, too.

According to the current studies, international standards and methods used for emergency preparedness and response, an accident comparable to Fukushima would require protective actions such as evacuation to around 20 km and sheltering to around 100 km. These actions would be combined with the intake of stable iodine.

In this framework, HERCA and WENRA propose a methodology for a common European approach allowing to recommend urgent protective actions as well as a minimum common level of preparation for these actions.

HERCA and WENRA consider that in Europe:

- evacuation should be prepared up to 5 km around nuclear power plants, and sheltering and ITB up to 20 km;
- a general strategy should be defined in order to be able to extend evacuation up to 20 km, and sheltering and ITB up to 100 km;
- nuclear and radiation safety authorities in Europe should continue attempts to promote compatible response arrangements and protection strategies amongst the European countries.

The need for rapid decisions using the simplified schemes for protective actions will only apply during an initial phase. As soon as the accident country is in a position to present a more elaborate assessment of the plant status and the expected off-site impact, it shall take the necessary steps to align its decisions and cross-border coordination mechanisms accordingly.

Part I.

HERCA-WENRA Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident.- General Mechanism

Stockholm, 22 October 2014

The HERCA-WENRA Approach (Part I) on emergencies was approved by the Board of HERCA on 12 June 2014 and later approved/endorsed by WENRA on 22 October 2014.

Content

Executive Summary

The Association of the Heads of the European Radiological protection Competent Authorities (HERCA) established its "Working Group on emergencies' (WGE) in June 2011. The mandate of the WGE covered all types of nuclear accidents. With regard to accidents originating from within the boundaries of the HERCA member countries the overall aim of the WGE was to come up with practical and operational solutions leading to a uniform way of dealing with any serious radiological emergency situation, regardless of national border lines. From December 2012 on, 42 experts (from 23 European countries) in nuclear safety, emergency preparedness and radiological protection of the WGE have worked aiming at fulfilling this mandate. As a result, they have proposed a response mechanism for the early phase of an accident for a better cross-border coordination of protective actions, called the "HERCA-Approach" which was approved by the Board of HERCA in June 2014. In developing the approach, the WGE started off by investigating what others had done or were doing, has taken advantage of this work and built on it. It has also acted in a complementary way to these activities, with the aim of reaching maximum mutual benefits. In this context, a representative of the European Commission has participated as observer in these activities and most recently an observer from the IAEA.

The HERCA-WENRA Approach recognises that in European countries, efficient emergency preparedness and response arrangements have been established for many years and are tested and challenged regularly. They allow authorities to issue recommendations for effective public protective actions. In the development of such arrangements, each European state defines its own priorities and objectives in planning for nuclear emergencies directly affecting its own territory, and the international and EU radiological protection frameworks leave large margins of freedom for setting national criteria for intervention. Emergency planning has evolved in all states over many years, mostly without giving great priority to cross-border issues. This has led to differences, sometimes significant. Should a nuclear emergency occur in Europe, these differences can potentially have a significant effect, especially if the location of the emergency is close to a national border. Internationally populations would feel unequally protected, depending on where they live. Aligning protective actions along adjacent national borders is therefore highly desirable.

In the early stages of an accident the uncertainties in terms of dose estimation and overall radiological impact are very large. The status of the reactor, the estimation of the amount and type of radioactivity released (the source term), and the dispersion conditions are very likely to be poorly understood in the first hours. The role of the decision-maker is to arrive at appropriate health protection measures taking into account these uncertainties. This inevitably leaves room for flexibility in decisions, even where there is a rigid national framework. The HERCA-WENRA Approach makes use of this freedom for coordination between neighbouring countries in order to align early decisions across borders. As a result, the respective national arrangements do not necessarily need to be changed. Instead, the prevailing differences are respected and are taken into account, and the response is based on 'compromise' solutions which are understandable and explainable in each given situation.

The HERCA-WENRA Approach on emergencies relies on the following principles: mutual understanding, coordination and mutual trust. It does not aim at proposing a uniform cross border framework to deal with such situations. The main strategy is to aim at an alignment of the response between neighbouring countries, or neighbouring territories. The HERCA-WENRA Approach comprises mechanisms of early information exchanges allowing neighbouring countries to align measures for protective actions by using as far as possible the existing dedicated bilateral and international arrangements. The HERCA-WENRA Approach is divided into 3 steps, the preparedness phase, the early phase and the later phase (development of a common situation report). The approach contains the main principles and leaves necessary margins of freedom for detailed implementation:

- In preparedness the aim is to achieve and maintain a shared understanding of the existing national emergency arrangements through the improvement of bilateral or multilateral arrangements, the testing of these arrangements and the implementation of improvements.
- In the early phase of an accident, the proposed HERCA-WENRA Approach foresees rapid information exchanges by using as far as possible the existing dedicated bilateral and international arrangements, including the exchange of liaison officers as appropriate. If the response is thought consistent, the neighbouring countries can recommend to their governments to follow these recommendations, i.e. adopt the principle that in the first hours, “we do the same as the accident country”.
- In the later phase a common situation report, accepted by all impacted countries, will further support coordinated protective actions.

The HERCA-WENRA Approach has been tested and validated against concrete and realistic accident scenarios in NPP’s that are close enough to national borders. Therefore a workshop was organized in September 2013. The workshop showed that in case of a sufficient information exchange most countries would be able to recommend to their decision-makers during the very early stages that the advice of the accident country should be followed. Other conclusions of the workshop will be taken into account when developing the guidelines for the implementation of the HERCA-WENRA Approach into the national arrangements.

The HERCA-WENRA Approach has the potential to improve the coherence of the response in case of a nuclear accident with impact on territories of other countries and to be used as guidance to implement Article 99.1[3] and 99.2[4] of the BSS Euratom. It also fulfils recommendation N°12.7.b of the ENCO study and it further addresses some of the other recommendations.

The HERCA-WENRA Approach is an incentive approach that comprises the necessary mechanisms for countries to exchange adequate information and to achieve compromise solutions on a voluntary basis during an emergency allowing for coherent and coordinated protective actions.

[3] Art. 99.1. Member States shall cooperate with other Member States and with third countries in addressing possible emergencies on its territory which may affect other Member States or third countries, in order to facilitate the organisation of radiological protection in those Member States or third countries.

[4] Art 99.2. Each Member State shall, in the event of an emergency occurring on its territory or likely to have radiological consequences on its territory, promptly establish contact with all other Member States and with third countries which may be involved or are likely to be affected with a view to sharing the assessment of the exposure situation and coordinating protective measures and public information by using, as appropriate, bilateral or international information exchange and coordination systems. These coordination activities shall not prevent or delay any necessary actions to be taken on a national level.

Part I. - HERCA-WENRA Approach for a better cross-border coordination of protective actions during the early phase of a nuclear accident - General Mechanism

1. Introduction

The Association of the Heads of the European Radiological protection Competent Authorities (HERCA) [1], identified as early as 2007 the need for an improved and harmonized cross-border approach in response to nuclear emergencies. A working group of HERCA was therefore established and issued a document on emergency preparedness "Practical Guidance – Practicability of Early Protective Actions" [2], which included the definition, aim and rationale of three early protective actions (sheltering, evacuation and thyroid prophylaxis), the planning phase, the intervention phase and the lifting of protective actions. Risk/benefit considerations and linked protective actions (e.g. food restrictions that need to be taken at the same time) were also discussed.

Subsequently the HERCA 'working group on emergencies' (WGE) concentrated its activities related to European nuclear emergency situations on mechanisms to improve the response to such an event. Key factors comprise the achieving of a more rapid exchange of information and the improvement of the coherence of national responses, including balancing radiation protection and social issues.

The present document deals with the development and testing of a response mechanism for the early phase of an accident, called the "HERCA-WENRA Approach". The testing of the approach during a dedicated workshop resulted in good progress, but also revealed several difficulties. The second part of this document presents and discusses those difficulties. Solutions for improvements are proposed, including concrete steps to be implemented through national, bilateral and multinational arrangements.

The HERCA-WENRA Approach on emergencies was approved by the Board of HERCA on 12 June 2014 and later approved/endorsed by WENRA on 22 October 2014.

2. Brief analyses of the given situation

Each European state defines its own priorities and objectives in planning for nuclear emergencies directly affecting its own territory, and the international and EU radiological protection frameworks leave large margins of freedom for setting national planning criteria for intervention. Emergency planning has evolved in all states over many years, mostly without giving great priority to cross-border issues. At the same time, the international framework for planning and response has changed.

This has led to differences, sometimes significant, in:

- Criteria for intervention levels for introducing protective actions (defined in terms of projected dose).
- Types of protective actions.
- Operational intervention levels (action levels based on measurements).
- Methods for assessing source terms.
- Methods for radiological impact assessment and dispersion modelling.
- Definitions of emergency planning zones.

In addition, there are differences between the concepts, e.g.:

- The implementation of protective actions can be based on assessments and calculations or triggers.
- The reference level and dose criteria are for planning purposes or also for response.
- The levels are legally binding or guidelines.
- There are national differences in the interpretation of international guidelines.

Should a nuclear emergency occur in Europe, these differences could potentially have a significant effect, especially if the location of the emergency is close to a national border. Figure 1 illustrates schematically how a particular protective action could be implemented when the decision is purely based on national considerations.

In each individual country, the decision is in line with the national plan and the legal framework, and it is well balanced and justified for the situation in the country given the national framework. However, internationally, populations would feel unequally protected, depending on where they live. An unbalanced cross-border implementation of one or more protective actions would lead to distrust in governmental decisions and potentially to panic. Attempts at explanation of the rational for such differences to the affected populations are not likely to be successful during the crisis. Aligning planning for protective actions along adjacent national borders is therefore highly desirable (see figure 1).

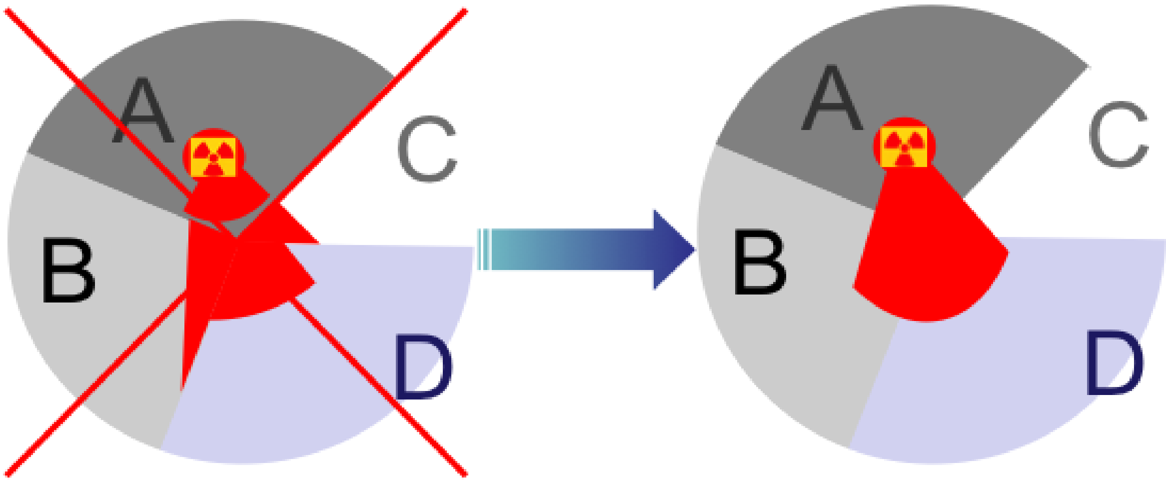

Figure 1: Country A has a nuclear emergency close to the borders of three others countries. All four countries are affected. Each country decides on a particular protective action individually. The protective action applies to the areas marked in red, for each country. The aim is to align protective actions.

3. The HERCA-WENRA Approach for the early phase

From December 2012, the WGE, in cooperation with members of WENRA, developed a new so-called "HERCA-Approach", relying on the following principles: mutual understanding, coordination, mutual trust and alignment of recommendations for decisions between neighbouring countries, or neighbouring territories, as the main strategy. The Board of HERCA had approved these principles on the occasion of its 10th meeting in Paris (30-31 October 2012). The basic aim is to develop mechanisms for implementing protective actions during an emergency in a consistent way along national borderlines but without – necessarily - changing the respective national arrangements. Instead, the prevailing differences are respected and are taken into account, and the response is based on 'compromise' solutions that are understandable and explainable in each given situation.

In the early stages of an accident the uncertainties in terms of the overall radiological impact are very large (see also figure 2).[5] The role of the decision-maker is to arrive at appropriate health protection measures despite this uncertainty. This inevitably leaves room for flexibility in decisions, even where a rigid national framework exists. The WGE believes that this freedom can be used for coordination between neighbouring countries in order to align early decisions across borders.

[5] WENRA and HERCA established the HERCA-WENRA taskforce that started in March 2014 to agree on common principles for advice in the early hours of a severe nuclear emergency. The difference between both approaches is that the general mechanism of the HERCA-WENRA Approach deals with accident scenarios with sufficient information for a technical assessment, whereas the HERCA-WENRA taskforce focuses on severe accidents where either no or not enough reliable information for a technical assessment is available or the rapid development of the accident does not leave sufficient time.

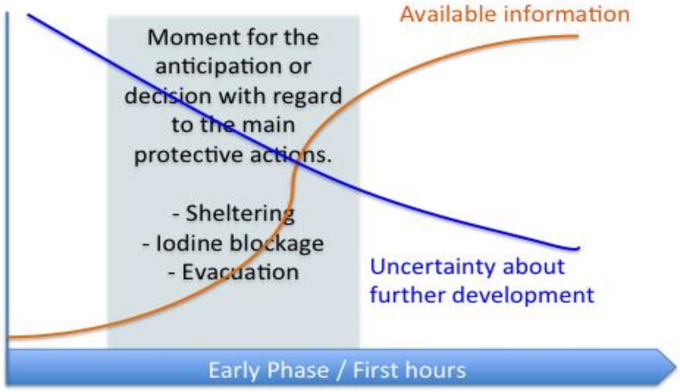

Figure 2: Factors influencing emergency decisions in the early phase of an accident.

For the early phase of an accident, the proposed HERCA- WENRA Approach comprises mechanisms of early information exchanges allowing neighbouring countries to align measures for protective actions by using as far as possible the existing dedicated bilateral and international arrangements.

This objective is in line with article 99 of EU-BSS calling for Member States to cooperate with other Member States and with third countries in addressing possible emergencies on their territory which may affect other Member States or third countries, in order to facilitate the organisation of radiological protection in those Member States or third countries.

The HERCA- WENRA Approach is divided into 3 steps, the preparedness phase, the early phase and the later phase (development of a common situation report). The approach contains the main principles and leaves necessary margins of freedom for detailed implementation. Compared to an earlier version, the approach below has been updated following the discussions at the workshop and the subsequent WGE meetings.

In preparedness

- Develop or improve already existing bilateral or multilateral arrangements, following a graded approach (i.e. the greatest priority is given to arrangements with the closest countries, and less urgent priority is given to countries at greater distances), with the goal of achieving and maintaining a shared understanding, taking into account the following:
  - National organizations:
    - General arrangements and information.
    - Stakeholders are involved in the emergency situations.
    - Facilities or reactors technologies.
  - National Emergency arrangements.
  - National strategies.
  - National expertise:
    - Assessment methods and tools (diagnosis, prognosis, environmental impact, etc.),
    - Information needed to understand correctly the products of national expertise (i.e. this may be available on restricted web pages – see below).
  - Arrangements regarding information exchange (what kind of information, ways to exchange it) during the accident and the deliverables (contents and frequency):

        - Forms, maps, technical and radiological data.
        - Information regarding countermeasures already implemented or planned to be.
        - Media information or press statements.
    - Trans border coordination mechanisms for protective measures in the response phase.
    - Communication policies to explain discrepancies in protection measures.
    - Joint public communication campaigns.
- Test arrangements and implement improvements.

Early phase / First hours

- The accident country should provide and update information required to the understanding of the situation and make available on-site and off-site assessments, using bilateral, multinational and international arrangements.
- Exchange of liaison officers and access to restricted websites may be arranged.
- Based on the information provided by the accident country and knowing the accident country's emergency arrangements, the neighbouring countries should be able to verify quickly if the response in the accident country is consistent with these arrangements.
- If the response is thought consistent, the neighbouring countries can recommend:
    - To their governments: if the accident country provides a recommendation for protective measures affecting part of the territory of the neighbouring country, to follow these recommendations i.e. adopt the principle that in the first hours, **"we do the same as the accident country".**
    - To their embassies: to protect their own nationals living in the accident country, by following the recommendations delivered by the authorities of the accident country.
- If the response is highly inconsistent, the neighbouring countries will urgently try to agree on an alternative position, which, together with their reasons, will be communicated to the accident country. The neighbouring countries should inform the competent authorities in the other European countries of their provisional position and the results of coordination.

Development of a common situation report

- The development of a common situation report is a major step towards more coordinated protective actions.
- Within the framework of the post-Fukushima action plan, the IAEA is currently developing mechanisms for an independent assessment and the production of a common situation report available for all Member States.
- The WGE will continue to analyse in more depth what has been proposed by the IAEA, and will then identify possible synergies and evaluate the potential for cooperation with the IAEA.

During its 12th meeting on 26-27 November 2013, the HERCA Board unanimously confirmed that the Approach fulfils the expectations of the Board in general terms.

4. Cologne WORKSHOP

4.1. Preparation of the workshop

As part of this work, the HERCA- WENRA Approach needs to be tested and validated against concrete examples, such as realistic accident scenarios in NPP's that are close enough to national borders. Therefore a first workshop was organized in September 2013. Some realistic accident scenarios in NPPs that are close to national borders were developed. The main goal was to explore whether the HERCA- WENRA Approach may potentially work, and also to identify difficulties.

Three scenarios with risks of considerable releases were prepared in advance, allowing participants some pre-preparation. The scenarios chosen were:

- NPP-Loviisa-1: The latest 8-countries exercise in Finland was used. STUK made its 15 situation reports available in English (USIE-format). The scenario was a real-time 12 hours exercise with 3 key moments, where decisions needed to be recommended or anticipated. In this scenario, no territory outside Finland was directly concerned with the main protective actions during the duration of the exercise. The main reason for providing this scenario for the Cologne workshop was to indicate how much information an accident country is able to provide in real time and in English.
- NPP Cattenom: IRSN had prepared a core melt scenario with risk of filtered venting. Detailed information was included in 5 IRSN and 3 ASN messages, the latter ones in the USIE format. The scenario contained two key moments, where decisions needed to be recommended or anticipated. In the earlier stage Germany and Luxembourg were concerned with protective sheltering and iodine blockage. The potentially large releases from a filtered venting threatened the territories of Luxembourg and Belgium where additionally evacuations needed to be considered.
- NPP Emsland: The BfS had prepared a short scenario concerning the NPP Emsland. It did not comprise elaborated technical details on the situation of the reactor, but contained key messages to be shared with the participants. The scenario was limited to the threat phase with a potential release along the border between Germany and the Netherlands (protective actions needed to be considered on both sides of the border). The two countries had agreed on the scenario beforehand.

4.2. Conduct of the workshop

The workshop took place on 24 September 2013 in Cologne at the GRS. 18 participants from 10 countries attended the workshop, during which participants responded to a predefined set of questions for each key moment of the 3 scenarios. The questions were:

1. Would we in reality have informed others that early?
2. Do we understand the information received from the accident country? If not, what is the issue?
3. Is the information consistent?
4. Is information missing? If yes, what?
5. As neighbouring country: Would we be able to align our recommendations with the protective action(s) as proposed by the accident country? If not what are the obstacles or concerns?

6. Would we be able to advice our citizens living in or visiting the accident country to follow the advice for protective actions of the accident country?
7. To those countries who are not a neighbouring country: If we were a very close neighbouring country (within the range of protective actions), would we be able to align our recommendations with the protective action(s) as proposed by the accident country?

Participants provided answers to those questions during the workshop as far as they were able to prepare themselves for the workshop and within the limits of a tight timescale. In particular WGE participants stressed the need for technical support from their national experts. Therefore, after the meeting, workshop participants and all other HERCA-WGE members had the opportunity to provide the missing answers in writing. In total, answers were received from 12 countries.

5. Main issues discussed at the workshop

The overall impression from all participants was very positive. The exercise of sitting together and having open and direct dialogues on concrete and relatively realistic accident scenarios was very useful. This gave raise to several interesting discussions.

Most participants thought that in the very early stages they would be able to recommend to their decision-makers that the advice of the accident country should be followed.

More discussion took place with regard to a possible alignment of recommendations for protective actions along national borders. However it has to be noted that the context of the discussion was somehow artificial. To imagine the coordination of protective actions between countries which do not share a common border is indeed difficult (see question 7). One key difficulty in such a discussion is clearly the missing knowledge of the arrangements in the accident country. Discussions concerned issues around the use of definitions and methodologies (for example, in dose calculations), the application of different decision criteria in the accident country and the extension of measures beyond EPZ’s, to name only a few examples. Also societal particularities, such as the density of population, are potentially very different, for example between Nordic countries and West European countries.

The configuration of the workshop also permitted the accident country to provide missing information. With such additional explanations, most participants thought it would be possible to recommend to their own decision makers to ‘do the same as the accident country’. This finding confirms the need for effective pre-emergency arrangements and shared understanding of these between all countries.

In a few countries, it seemed at the workshop, that the possibilities of adjusting recommendations during an emergency to favour coherent protective actions were very limited. These countries are obliged by law to do their own assessments and/or to recommend protective actions based on their own intervention levels. The workshop participants discussed several technical possibilities to overcome these difficulties. However no real solution emerged for these cases.

The WGE has undertaken more in-depth analyses of the workshop after having received all written answers. The main findings of this work are presented and discussed in the following chapter.

6. Main Findings of the workshop

6.1. Findings

The most important findings of the workshop were:

- Mutual knowledge of EPR arrangements is essential to make the HERCA-WENRA Approach work;
- Frequent updates by the accident country on the situation are essential for other countries to be able to verify the response in the accident country and are a pre-condition for them to be able to advise their decision-makers to follow the accident country;
- Most countries (not being neighbouring countries) were able to recommend their own citizens in the affected area to follow the advice given by the accident country based on the available information;
- The success of a harmonized approach between neighbouring countries is strongly influenced by bilateral agreements;
- Different values in decision criteria can result in a different advice than the one given by the accident country.

In the following paragraphs, the main elements of those findings are elaborated in more detail.

6.2. Information to be provided by the accident country through its first messages

A limited amount of information is available to be included in the first message. The type of information available for being included in the first message depends on the accident scenario. The present report does therefore not distinguish between the first message and the subsequent updates, but only refers to those in general terms as "Early Phase Messages (EPM)". As a guiding principle, speed and accuracy are considered more important than quantity or completeness of information. Frequent updates are necessary.

Issue	Rational
(Conservative) evaluation of the potential hazard area.	Favours a common understanding and coherent communication internationally. Helps to give early assurance to populations outside this area.
Reports that contain maps, dispersion calculations and pictures.	Useful internationally for communication and similar purpose. Should be annexed to EMERCON forms.
Use of short message systems.	Essential to be used in case the situation changes rapidly (e.g.: update of emergency classification). It is recommended to the IEC to provide such a tool within USIE.
Include other operational measures, such as traffic restrictions, food restrictions, etc.	Important information for direct neighbouring countries (e.g.: hotlines). Helps deciding on travel advice. This possibility is included in the EMERCON forms, both for food and traffic. One may even add

Issue	Rational
	"other" measures
Clearly distinguish whether a protective action is, planned, recommended, decided (ordered) or implemented (taken).	The EMERCON form only asks to *"Describe protective actions that are planned, ordered, taken or withdrawn".* There is a need for clarity, in particular informing on recommendations for protective actions.
Prognosis for development, including a worsened case scenario (on a case by case basis).	This seems particularly useful in cases of a cliff edge effect (when consequences could dramatically worsen) or when failure of a further barrier is probable. In other situations such a scenario may be misleading and even favouring differences in decisions across national borders.
All responding countries enter information related to their decisions into USIE	This allows all countries, but in particularly the accident country, to reduce the burden on communication, i.e. the accident country may anticipate public discussions that may be stimulated by decisions taken elsewhere.

Table 1: Issues that are presently not considered by the EMERCON forms but that are worth exchanging through the EPM.

The WGE recommends that all countries should make full use of the USIE system and related EMERCON forms to inform the competent authorities of neighbouring countries and the international community. The workshop confirmed that the type of information foreseen to be exchanged by the EMERCON forms is vital in the early phase. In particular, the accident country should enter information on the classification of the emergency and its basis for declaration as early as available. For the provisional INES classification it should be borne in mind that the INES classification refers to the actual situation rather than to a possible degradation.

Besides, the workshop identified a couple of issues that are presently not considered by the EMERCON forms but that are worth for exchanging through the EPM. Those issues and their rational are given in Table 1.

Restricted websites exist in several countries with up-to-date information concerning the development of the accident. In some cases, authorities from neighbouring countries, the IEC and the EC have access to those sites during an emergency. In other countries, similar projects are under discussion. As a result of the workshop, it seems indeed a very good practice since the direct access to such a restricted website allows the technical expertise organizations in neighbouring countries to follow the situation in a timely manner as it develops. This permits anticipation and preparation for necessary developments and decisions. It also fosters a common understanding of the situation. As such, this finding confirms the HERCA-WENRA Approach as given in § 3.

A prerequisite for a successful implementation is a consistent mutual or bilateral agreement with the objective of co-ordinating activities in a wider area. A situation where neighbouring countries use the information provided to implement measures unilaterally has to be avoided. Finally, training will be needed before being able to understand the national expertise products.

6.3. Information to be exchanged in preparedness

The workshop demonstrated again how many differences in emergency preparedness arrangements actually exist. Harmonizing all those differences seems highly unrealistic. Therefore the aim of the HERCA-WENRA Approach to achieve from the beginning alignment of protective actions during the response, taking prevailing differences in preparedness into account, can be seen to be the correct approach.

As a minimum prerequisite, countries need to have sufficient knowledge of each other's arrangements. In this context, the workshop confirmed that good knowledge of the arrangements in the accident country helps to understand and to agree with decisions taken in that country. It further helps other countries in particular on the derivation of the accident country's recommendations for their own population. Indeed, neighbouring countries that, during the workshop, knew the arrangements in the accident country well enough could follow the HERCA-WENRA Approach more easily.

It also became clear that including this type of information in the first messages is not feasible, unless prepared and discussed beforehand. In that sense, although the HERCA- WENRA Approach is purely designed to improve coordination during response, pre-emergency arrangements are vital.

Otherwise the statements of the accident country will not be understood and will not be followed.

One possibility to effectively exchange this type of information would be the establishment of country factsheets. Those sheets should be short (approx. 2 pages), visual (to enable information to be found very quickly), factual and concise (i.e. key words rather than full sentences). The WGE should establish those factsheets and develop an appropriate solution for regular updates and dissemination.

6.4. Bilateral arrangements

The success of a harmonized approach between neighbouring countries is strongly influenced by bilateral arrangements (or multilateral if more than 1 country will be directly influenced by an accident within a certain NPP). This makes it easier to verify the assessment of the accident country and gives more understanding to the neighbouring countr(y)(ies). Understanding the response of the accident country is very important for advising decision-makers to harmonize their response with the accident country.

Additionally the arrangements shall contain provisions for coordinating the media response and for communicating well in advance about decisions and the reasoning behind those decisions, in particular in those cases when different decisions are unavoidable.

The WGE should develop guidelines for the establishment of these arrangements. Where an NPP is close to more than one other country, multilateral arrangements should be envisaged.

6.5. Key differences and potential obstacles for aligning recommendations for protective actions along national borderlines

Since emergency preparedness arrangements differ widely between European countries, it was to be expected that the verification of the HERCA-WENRA Approach against concrete accident scenarios would result in identifying situations where aligning recommendations for protective actions is not realistic. The workshop indeed revealed a few obstacles that may jeopardize the possible success of the HERCA-WENRA Approach. On the other hand, such obstacles cannot be generalized and depend largely on the situation and on the country concerned.

Table 2 gives an overview of the main difficulties that were identified. For most cases, it seems possible, nevertheless, to substantially reduce their impact through adapted solutions, as highlighted in the right column of the table.

Difficulties	Improvements to be implemented
Information not correctly understood (e.g.: definition, classification of the emergency different to IAEA, Basis for decisions, etc).	Increase knowledge of national arrangements and assumptions (country fact sheets, see § 6.3)
Information too late or too slow, not permitting other countries to prepare for necessary protective actions on the necessary timescale.	Improve quality of first messages (see § 6.1); Make use of single short messages to a specific subject or event between the regularly sent EMERCON forms; Grant access to restricted websites with the “national expertise product”.
Different decision criteria, (e.g. action levels or triggers) used by the accident country, not used and/or not known by others.	Increase knowledge of national arrangements (country fact sheets, see § 6.3). The workshop has also shown that a good knowledge of the decision criteria used in the accident country helped most other countries to appreciate related decisions and to adapt their response, even if such criteria were otherwise not used.
Different values in decision criteria (e.g.: intervention levels). In a couple of countries intervention levels need to be used on a mandatory basis to trigger protective actions with only minor room for other considerations in issuing recommendations.	The new EU-BSS [3] has introduced the concept of “reference levels” for emergency and existing exposure situations. It allows for the protection of the individual as well as consideration of other societal criteria in the same way as dose limits and dose constraints for planned exposure situations. The obligation for the Member States to implement the new EU-BSS gives a unique chance for improved understanding and the implementation of adjustable criteria and the reduction of differences across Europe.
While appreciating different criteria, most authorities limit their considerations to their own territory. (the potential hazard	Need for systematic bilateral agreements that allow to look at the potential hazard area as a whole and to coordinate activities effectively during the

Difficulties	Improvements to be implemented
area is not regarded as a whole)	response between the involved bodies.
In a few countries, recommendations need by law to be based on own independent assessments following strict decision triggers.	There seem to be only a few countries concerned. However in those cases a successful implementation of the HERCA-WENRA Approach could be seriously jeopardized. The new EU-BSS may possibly help.
International requirements and recommendations are differently interpreted across Europe. This seems to be a major challenge.	More detailed work needed on listing the basis for each country, e.g. the use of reference levels, the legally binding criteria, the use of triggers. However, the new BSS present possibilities for further harmonisation, as above.
Extendibility of protective measures beyond EPZ's from an operational point of view.	Need for systematic bilateral agreements. It will be important to clearly define operational limitations (how far and how much can be extended) and how the responsibilities are assigned in each country. (In some countries the advisory body considers operational issues when issuing recommendations).
Advices on travel and traffic remain uncoordinated.	Implementation of the recommendations from the HERCA report "Practical proposals for further harmonisation of the reactions in European countries to any distant nuclear or radiological emergency" [4]

Table 2: Overview of the main difficulties that were identified.

7. Summary of the main findings

The workshop has enabled the WGE to clearly identify the following issues that need to be developed and implemented to allow best use of the HERCA-WENRA Approach:

- Implement nationally the guidelines included in chapter 6.1 for improving the effectiveness of the first messages.
- Development of country fact sheets;
- Development of a list of issues to be dealt with in bilateral or multilateral arrangements;
- Develop a common understanding of key elements of the new EU-BSS [3] and aim at reducing differences through a coordinated transposition and a better application of international recommendations. The EU-BSS does support such changes as they give the opportunity to review our basic principles of radiation protection in emergency situations.

Additionally, as a result from the workshop, the WGE makes the following recommendations:

- Make use of short message information exchange during the response;

- Countries should consider granting access to restricted websites and exchanging liaison officers;
- Always consider the whole affected area, independent of a national border, when making decisions.
- Aligning protective measures along borders should be a factor in decision-making.

8. Conclusions and further steps

The workshop has demonstrated that the HERCA-WENRA Approach has the potential to improve the coherence of the response in case of a nuclear accident with impact on territories of other countries. It was also shown that the approach allows for the assurance – sometimes deemed necessary - that things are done properly in the accident country. This permits in particular to recommend one's own citizens, who stay in the affected area, to follow the advice from the accident country.

A further positive result is the confirmation that the HERCA-WENRA Approach, as given in § 3, contains all necessary elements, ideas and principles needed.

While it may remain difficult to completely eliminate the occurrence of differences and inconsistencies in the response, the aim of the approach should be that this becomes the exception rather than the rule. In those cases where two countries take unavoidably different approaches, they shall coordinate their media response and communicate well in advance about their decisions and the reasoning behind those decisions. A systematic implementation of the findings of the present report (see summary in § 7) will certainly help to significantly improve towards the overall objective of the HERCA-WENRA Approach.

References

[1] HERCA the Association of the Heads of the European Radiological protection Competent Authorities; Radiation Regulator, Volume 1, Number 1.

[2] Emergency Preparedness. Practical Guidance – Practicability of Early Protective Actions; www.herca.org

[3] Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.

[4] Practical proposals for further harmonisation of the reactions in European countries to any distant nuclear or radiological emergency; www.herca.org

Definitions

Accident country	The country where the nuclear or radiological emergency has taken place.
Affected countries	The countries that are not the accident country where protective actions need to be considered because of a transborder radiological contamination.
HERCA/WENRA countries	Each country represented through one or more authorities within HERCA/WENRA.
HERCA/WENRA members	Each authority who is a member of HERCA/WENRA.
Impacted countries	Countries that are not necessarily affected countries but which need to issue recommendations for their own citizens in the affected area, including travel arrangements.

List of acronyms

EMERCON: Emergency Convention

EP&R: Emergency Preparedness and Response

EPM: "Early Phase Messages" the first message and the subsequent updates distributed by the accident country in the early phase

EU-BSS: Council Directive 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.

HERCA: Association of the Heads of the European Radiological protection Competent Authorities

IAEA: International Atomic Energy Agency

IEC: Incident and Emergency Centre

MFA: Ministry of Foreign Affairs

NPP: Nuclear Power Plant

TSO: Technical Support Organisation

USIE: Unified System for Information Exchange on Incidents and Emergencies

WENRA: Western European Nuclear Regulators' Association

WGE: HERCA working group on emergencies

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The authors participated actively in the Working Group “Emergencies” (WGE) between December 2011 and May 2014. During this period, the WGE was chaired by Mr. Patrick Majerus (DRP, LU).
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ASN – Autorité de Sûreté Nucléaire; **BABS** - Bundesamt für Bevölkerungsschutz; **BfS** – Bundesamt für Strahlenschutz; **BMLFUW** - Austrian Federal Ministry of Agriculture, Forestry, Environment and Water Management; **BMU** – Bundesumweltministerium; **BNRA** – Nuclear Regulatory Agency; **CSN** - Consejo de Seguridad Nuclear; **DRP** – Division de la Radioprotection; **EC** – European Commission; **ENSI** - Swiss Federal Nuclear Safety Inspectorate; **FANC/AFCN** – Federal agency for nuclear control; **GR** - Icelandic Radiation Safety Authority; **GRS** – Gesellschaft für Reaktorsicherheit **ILT** - Inspectie Leefomgeving en Transport; **IRSN** - Institut de Radioprotection et de Sûreté Nucléaire; **Minez** - Ministerie van Economische Zaken; **NCRRP** - National Center of Radiobiology and Radiation Protection; **NRPA** – Norwegian Radiation Protection Authority; **OFSP** – Federal Office of Public Health; **ONR** - Office for Nuclear Regulation; **OSSKI** - National Research Institute for Radiobiology and Radiohygiene; **PHE** - Public Health England; **RPII** - Radiological Protection Institute of Ireland; **SNSA** - Slovenian Nuclear Safety Administration; **SSM** - Swedish Radiation Safety Authority; **STUK** - Radiation and Nuclear Safety Authority.

Part II.-

HERCA-WENRA Approach in case of a Severe Accident requiring Rapid Decisions for Protective Actions, while very little is known about the Situation

Stockholm, 22 October 2014

The HERCA-WENRA Approach (Part II) on emergencies was approved by HERCA and WENRA on 22 October 2014.

Content

Executive summary

The HERCA-WENRA task force was established jointly by HERCA and WENRA. It included 21 experts in nuclear safety, emergency preparedness and radiological protection, belonging to 14 different countries. The HERCA-WENRA task force operated between March and September 2014. Its mission was to identify a common European approach addressing a severe accident affecting one or more nuclear power plant(s), and requiring rapid decisions for protective actions, while very little information is available. The HERCA-WENRA task force has built on existing European approaches, namely the NERDA, Nordic and HERCA approaches. IAEA developments and recommendations in emergency preparedness and response have also been considered.

HERCA and WENRA recognize that in European countries, efficient emergency preparedness and response arrangements have been established for many years and are tested and challenged regularly. They allow authorities to issue recommendations for effective public protective actions. In many cases, the current arrangements require exchange of a significant amount of information between the plant and the responsible authorities. However, past experience shows that the possibility of severe accidents without the information required by the current arrangements on the plant status in the initial stage cannot be completely ruled out. Such accidents could be as severe as the Fukushima one, affect more than one European country and require rapid protective actions in several of them.

HERCA and WENRA propose a general approach for dealing with the initial stage of such highly improbable cases. This approach, called the “HERCA-WENRA approach”, can serve as a basis to complement, when necessary, existing arrangements in the initial phase of an emergency situation and allow better coordination of protective actions between countries. The HERCA-WENRA approach proposes that, in case of a severe accident with great uncertainty about the situation, protective actions to be recommended to the decision makers be decided on the basis of the plant status and weather conditions. Three so-called Judgement Evaluation Factors (JEFs) are proposed: core melt risk, containment integrity and wind direction. The HERCA-WENRA approach assumes that a General Emergency (or an equivalent emergency class) is declared when a risk of core melt exists.

The JEFs have to be evaluated using expert judgement based on the nature of the event which has initiated the accident (e.g. earthquake, flooding, airplane crash), the information which can be obtained from the plant and the pre-existing knowledge of its behaviour under extreme conditions. As a first step of a general approach, HERCA and WENRA have only considered three direct protective actions: evacuation, sheltering and iodine thyroid blocking (ITB).

If the HERCA-WENRA approach has been initiated and core melt is judged possible, a precautionary approach is applied and the following actions shall be implemented: evacuation over a distance of 5 km, and sheltering and ITB over a distance of 20 km. However, if it is assessed that, additionally, containment integrity is lost, more serious actions would become necessary, such as evacuation up to 20 km, and sheltering and ITB up to 100 km. Depending upon the prevailing and forecast wind conditions, protective actions are implemented either on a circular area around the plant or on a limited number of sectors of this circular area. The precise extension of the areas where protective actions are recommended is also adjusted as a function of demography, wind speed and stability.

It shall be noted that countries may have protection strategies more stringent than the conclusions of the HERCA-WENRA Task Force. Safety authorities in neighbouring countries should agree on consistent recommended protective actions.

Considering the safety level of European nuclear power plants and their improvements resulting from the lessons learned from the Fukushima disaster, HERCA and WENRA recognize that the probability of an accident comparable to Fukushima, which would require evacuation up to around 20 km and sheltering and ITB up to around 100 km, is very low.

Therefore, HERCA and WENRA consider that in Europe:

- **evacuation should be prepared up to 5 km around nuclear power plants, and sheltering and ITB up to 20 km;**
- **a general strategy should be defined in order to be able to extend evacuation up to 20 km and sheltering and ITB up to 100 km;**
- **nuclear and radiation safety authorities in Europe should continue attempts to promote compatible response arrangements and protection strategies amongst the European countries.**

Part II

HERCA-WENRA Approach in case of a Severe Accident requiring Rapid Decisions for Protective Actions, while very little is known about the Situation

1. Introduction

On January 15, 2014, a HERCA-WENRA extraordinary meeting held in Brussels established an Ad hoc High-Level Task force on emergencies. Its assigned task was to identify common European principles addressing nuclear emergency response decisions, in case of a severe accident affecting one or more nuclear power plant(s) and requiring rapid decisions for protective actions, while very little is known about the situation.

Between March and September 2014, the 21 high ranking members representing 14 countries met on four occasions. Coming from the fields of nuclear safety, emergency preparedness and radiological protection, they framed the HERCA-WENRA Approach, using the existing European approaches (NERDA, Nordic and HERCA) as a basis. IAEA developments and recommendations in emergency preparedness and response have also been considered. The following paper is the result of these discussions.

2. General Context

In European countries, emergency preparedness & response (EP&R) arrangements for nuclear power plants have been developed over many years. Nuclear and radiation safety authorities have established mechanisms with the nuclear power plants within their country to enable them, even under very difficult conditions, to have access to sufficient information and plant parameters for independently assessing the situation, both at the plant and concerning possible off-site consequences. This allows the authorities to issue recommendations for effective public protective actions, tailored to the possible or actual radiological exposure situation of the population. Drills are performed regularly through emergency exercises in order to train the teams and provide them with experience, which is in turn used to further improve the current arrangements.

However, in the case of the major nuclear accidents that have occurred in the past, namely Chernobyl and Fukushima, the situation was far less clear. Particularly for Fukushima, information was sparse during the early phase. At best only preliminary assumptions could be made on a possible status of the plant, its evolution, and radiological consequences for the population and environment. In spite of the insufficient and unreliable information, it was necessary for the Japanese authorities to decide rapidly on protective actions for their population.

In Europe, both operators and authorities have sought to learn the main lessons from those accidents in order to strengthen their lines of defence. Should an accident occur, it is particularly vital to maintain functional communication channels between the plant(s) and the responsible authorities at all times, so that relevant and justified decisions can be made to attenuate the consequences of the situation.

Nevertheless, the possibility of a severe accident scenario with no or insufficient information on the plant status cannot be completely ruled out. Furthermore, the immediate effects of such a severe accident in Europe would most probably affect more than one country. This calls for the coordination of the approach to be followed in that case, to allow mutual understanding, consistent and fast decisions of countries affected by an extreme situation of this kind.

Independently of the root cause of the accident, several scenarios can be imagined where being able to use an assessment with projected doses to the populations as a basis for decision making is not realistic. This may be the case when:

- Little or no information from the plant is available (e.g. communication systems have failed such that nothing is known concerning the availability of any safety systems, the status of the reactor core and/or the primary circuit and/or the containment).
- The threat of a large/significant radiological release cannot be reliably estimated.

Since such a severe accident cannot be completely ruled out, the WENRA – HERCA extraordinary meeting of 15 January 2014 in Brussels decided to create an Ad hoc HERCA-WENRA High-Level European Task force in order to identify shared principles on how to address such an extreme situation.

In that context, the HERCA-WENRA Approach focuses explicitly on these extreme cases, producing a framework for European countries as guidance in formulating their detailed EP&R arrangements for an accident involving one or more nuclear power plant(s). Potentially affected countries would benefit from more detailed preparation specifically addressing the issues unique to these types of accidents.

Pre-existing agreement on the approach and actions to be taken in case of such extreme conditions would also significantly enhance the confidence of authorities in charge of protective actions. This is particularly the case for authorities of the countries neighbouring the country where the accident occurred.

It finally needs to be noted that past attempts to reduce differences between national EP&R arrangements were generally not very successful. Any success on agreeing upon common principles for these severe accident scenarios may improve mutual understanding and allow for better coordination of protective actions between countries, including coherence and consistency in early communications to the population across Europe.

Between March and September 2014, the HERCA-WENRA Task Force has developed a shared position for recommendations of generic protective actions for the early phase of those particular nuclear emergency situations at nuclear power plants (NPPs) in operation when only information on the type of event, internal or external, the possibility of core melt and very little other information is available, including cases where the local operator no longer has the means to understand the situation or to communicate about the situation. At the present stage, the focus of the HERCA-WENRA Task Force lies on three direct protective actions: evacuation, sheltering and iodine thyroid blocking (ITB).

The present paper intends in no way to foreclose any national political decision as to whether or not to implement the proposed protective actions. The main intention is to give an outline on protective actions that may need to be considered. The aim of the HERCA-WENRA Task Force is to provide a common approach relevant to specific situations that can serve as a basis to complement, when necessary, existing arrangements in the initial phase of an emergency situation and allow better coordination of protective actions between countries.

It shall be noted that countries may have protection strategies more stringent than the conclusions of the HERCA-WENRA Task Force. Safety authorities in neighbouring countries should agree on consistent recommended protective actions.

3. Mechanisms for triggering decisions during an emergency situation

During the very early phase of any accident, the decision needs to be taken whether available information is sufficient to support a normal national assessment in a timely manner or whether the country faces an accident with great uncertainty about the situation. In the latter case, recommendations of protective actions have to be defined following a simplified scheme, hereafter referred to as the “HERCA-WENRA Approach”. The normal national EP&R approach and the HERCA-WENRA Approach comprise as a whole the overall national approach. The different parts of the overall national approach, the normal approach and the HERCA-WENRA part, have to be compatible to ensure a smooth transition from one part to the other. The country where the accident occurs has the responsibility to judge which approach is appropriate and to organize if necessary the transition between the HERCA-WENRA Approach and the normal national approach.

It is therefore proposed that countries with operating nuclear power plants establish appropriate internal mechanisms, preferably within the competent regulatory authorities for nuclear safety and radiation protection (hereafter referred to as “safety authorities”), for triggering these decisions in a timely manner. The criteria to initiate the HERCA-WENRA Approach are any event, internal or external hazard, including a terrorist attack that might lead to large radioactive release in combination with a lack of the information necessary for applying the normal national EP&R arrangements.

Concerning the necessary decisions for protective actions in neighbouring countries, bilateral arrangements need to contain specific mechanisms for rapid exchange of relevant information between safety authorities. If the notification provided by the safety authority provides elements affecting parts of the territory of the neighbouring country, the neighbouring country should aim at doing the same as the “accident country”[6]. Such a mechanism would be in line with the principles of the general mechanism of HERCA-WENRA Approach.

[6] Country where the affected NPP is located

Generally, the need for simplified decisions using the HERCA-WENRA Approach will only apply during an initial phase. As soon as the accident country is in a position to present a more elaborate assessment of the plant status and the expected off-site impact, it shall take the necessary steps to align its decisions and cross-border coordination mechanisms accordingly.

It is finally emphasized that early information exchanges between neighbouring countries represent an essential element for a successful and coherent cross-border implementation of protective actions. The present paper shall nevertheless in no way provide a justification to neighbouring countries to unilaterally apply the HERCA-WENRA Approach.

4. Basic considerations of the HERCA-WENRA Approach

The HERCA-WENRA Approach is conceived as a general framework. It proposes a methodology for a common European approach allowing to recommend urgent protective actions as well as a minimum common level of preparation for these actions. Detailed implementation aspects will be provided in a second step, provided the methodology is accepted by HERCA and WENRA.

To initiate the HERCA-WENRA Approach (in case of any event, internal or external hazard, including a terrorist attack that might lead to a core melt and for which insufficient information is available to apply the national EP&R arrangements), the protective actions shall be based on the three so called Judgment Evaluation Factors (JEFs):

JEF	Description	Possible values of JEF		
1	Is there a risk of core melt?	Yes	No	Unknown
2	Is the containment integrity maintained?	Yes	No	Unknown
3	Is the wind direction:	Steady	Variable	Unknown

Table 1: Definition of JEFs

Note: If a General Emergency (or an equivalent emergency class) is declared, it is considered that a risk of core melt exists.

Initial containment integrity characterizes the overall structural state of the containment immediately after the initial event. Containment integrity could for example be considered lost after an airplane crash or if the initial event occurs while the containment is open. During the course of the accident, a small increase of containment leakage should not lead to an estimation of loss of containment integrity. However, if a drastic event like a violent internal explosion is expected, then containment integrity should be considered as lost (JEF 2 = No).

Wind direction should be considered, if known, in the period where large releases are expected.

5. Protective actions considered in the HERCA-WENRA Approach

The following protective actions have to be considered in the very early phase:

- **Evacuation**
  The rapid, temporary removal of people from an area to avoid or reduce short-term radiation exposure during emergency.

  It must be noted that there is a certain risk that evacuation will take place under the plume.

  The HERCA-WENRA position on this issue is detailed in § 8.4.
- **Sheltering**
  The use of a structure for protection from an airborne plume and/or deposited radionuclides.
- **Iodine Thyroid Blocking (ITB)**
  The administration of a compound of stable iodine to prevent or reduce the uptake of radioactive isotopes of iodine by the thyroid in the event of an accident involving radioactive iodine releases.

The protective action "Ban on harvesting and grazing" and actions like access control, which are part of the normal EP&R approach, are not considered by the HERCA-WENRA Approach at this stage.

Information to the public and the neighbouring countries as well as notification of and information to international organisations (IAEA and ECURIE) are of primary importance in any case.

6. Protective action zones considered in the HERCA-WENRA Approach

The protective actions should be planned up to distances of 5 km for evacuation and 20 km for sheltering and ITB. An extension to larger distances should also be considered for instance to take into account situations where containment integrity is lost (JEF 2 = No) (e.g. plane crash or large internal explosion) and large releases are expected.

It should be noted that existing planning zones such as the Precautionary Action Zone (PAZ) and the Urgent Protective Action Planning Zone (UPZ) should be used and adapted if already implemented in national emergency response plans. Local specificities (e.g. demographics and geography) should be taken into account to determine the exact shapes of the zones where the protective actions are implemented.

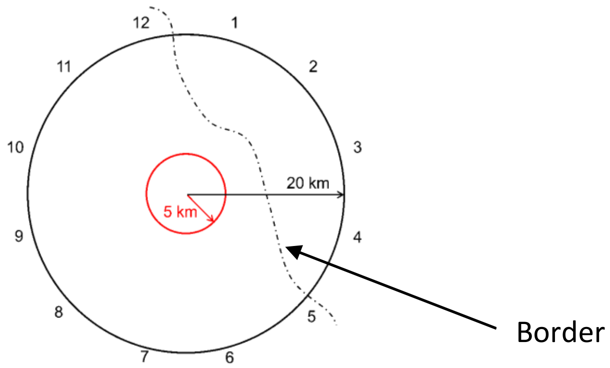

Figure 1: Distances where protective actions have to be planned.

7. Sectors considered in the HERCA-WENRA Approach

Depending on wind conditions forecast during the expected period of large release, these protective actions shall be implemented only in sectors potentially concerned rather than in the whole circular area. On the other hand, when complete circular areas are considered due to lack of wind, the extent of the area could be reduced to a shorter radius. HERCA and WENRA recommend the use of 30° sectors to cover the area concerned. However, other arrangements can also be used if already implemented in national emergency response plans (22.5° etc.). It should be noted that the variation of wind conditions might require adaptation with time of the sectors where protective actions are implemented. It has also been decided to always add an inner circle (keyhole approach) to the concerned sectors of at least 1 km (subject to adaptation to the local/field situation). Outside of the 20 km area, the radial sectors lead to large areas and therefore may not be appropriate and a more regional delimitation following municipality borders, rivers, etc. should be considered when more information on the radiological situation becomes available.

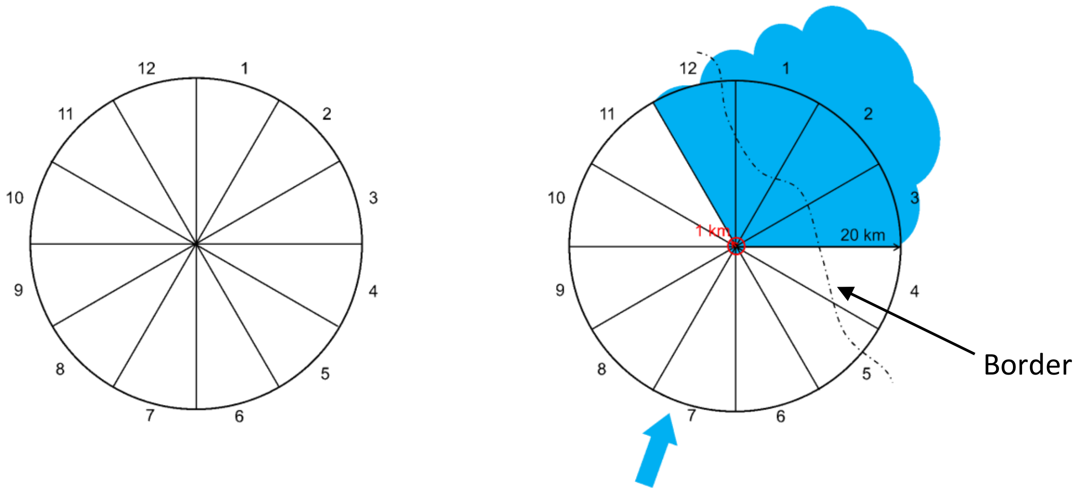

Figure 2: 30° sectors for the implementation of protective actions based on wind/weather conditions

8. Description of the HERCA-WENRA Approach

8.1. Estimation of time to release not available

Using the HERCA-WENRA Approach, a precautionary approach should be applied. Therefore, considering a situation that could lead to a core melt (JEF1 is Yes or unknown) and for which there is no indication of loss of containment integrity (JEF2 is Yes or unknown) and the time to radioactive release is not known, protective actions shall be implemented as follows:

Protective Action	Distance
Evacuation + ITB	up to 5 km
Sheltering + ITB	5 to 20 km

Table 2: Protective actions using the HERCA-WENRA Approach in case JEF1 is Yes or unknown, and for which there is no indication of loss of containment integrity (JEF2 is Yes or unknown) and time of release is unknown.

Note: If a General Emergency (or an equivalent emergency class) is declared it is considered that a risk of core melt exists.

8.2. Estimation of time to release available

Under the same situation and if the time to release ($t_{release}$) can be estimated, the evacuation of the population within a 5 km distance has to be evaluated on the basis of the time needed to evacuate (t_{evac}). If $t_{evac} > t_{release}$, sheltering is preferred over an evacuation under the plume.

The other protective actions are not considered critical because the implementation times for sheltering and ITB are comparably short if iodine tablets are already pre-distributed. Since sheltering cannot be implemented for a very long duration, it should be prepared immediately but only implemented a few hours before the time of release.

Protective Action	Distance	
	tevac > trelease	tevac < trelease
Evacuation + ITB	-	up to 5 km
Sheltering + ITB	up to 20 km	5 to 20 km

Table 3: Protective actions using the HERCA-WENRA Approach in the case JEF1 is yes or unknown, and for which there is no indication of loss of containment integrity (JEF2 is Yes or unknown) and time of release is known.

If containment integrity is lost due to the nature of the initiating event or the evolution of the accident (e.g., plane crash or large internal explosion) and core melt is expected, extended protective actions would become necessary, such as evacuation up to 20 km and sheltering and ITB up to 100 km. Additional ITB actions specific to children could also be necessary.

8.3. Weather Conditions

If the wind direction (JEF3) is known and steady, it can be used to determine in which adjacent basic (30°) sectors (cf. figure 2) protective actions are necessary. If the wind direction is unknown, protective actions have to be implemented in a zone of 360° around the installation and up to the specified distance.

8.4. Risk of evacuation under the plume

In the HERCA-WENRA Approach, sheltering and ITB are preferred to evacuation if it is predicted that the evacuation will actually occur under the plume. In all other cases, including unknown time to release, evacuation is preferred. Given the current state of knowledge, there are many uncertainties on this issue. Therefore, HERCA and WENRA recommend that additional studies be performed regarding the advantages and disadvantages of evacuating under a plume and considering a variety of factors including human behaviour, to provide a sound basis for such recommendations.

8.5. Organisation

The HERCA-WENRA Approach requires rapid assessment of the JEFs by experts, based on the available information. Timely interactions between the operator and the safety authorities will enhance the possibility to quickly assess the JEFs. Therefore, a national framework relying significantly, though not exclusively, on operators would facilitate the HERCA-WENRA Approach. Experts responsible for the necessary assessment should be designated. Robust communication means should also be available between the key players involved in the evaluation of the situation and the production of recommendations for protective actions.

9. Harmonized preparation of protective actions in Europe

As shown by the Fukushima accident, a large nuclear catastrophe anywhere in the world, including in Europe, cannot be completely excluded. Emergency preparedness and response arrangements should therefore be prepared for such cases. According to the current studies, international standards and methods used for emergency preparedness and response, an accident comparable to the Fukushima one would require protective actions such as evacuation to around 20 km and sheltering to around 100 km. These actions would be combined with the intake of stable iodine.

However, considering the safety level of European nuclear power plants and their improvements resulting from the lessons learned from the Fukushima disaster, it is estimated that the probability of such a catastrophic accident is very low.

Therefore, HERCA and WENRA consider that in Europe:

- evacuation should be prepared up to 5 km around nuclear power plants, and sheltering and ITB up to 20 km;
- a general strategy should be defined in order to be able to extend evacuation up to 20 km and sheltering and ITB up to 100 km;
- nuclear and radiation safety authorities in Europe should continue attempts to promote compatible response arrangements and protection strategies amongst the European countries.

List of acronyms

EP&R: Emergency Preparedness & Response
HERCA: Heads of the European Radiation Protection Competent Authorities
ITB: iodine thyroid blocking
JEF: Judgment Evaluation Factor
WENRA: Western European Nuclear Regulators' Association

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ASN – Autorité de sûreté nucléaire; **BMU** – Bundesumweltministerium; **BNRA** – Nuclear Regulatory Agency; **Bel V** – FANC/AFCN TSO; **CSN** - Consejo de Seguridad Nuclear; **DRP** – Division de la Radioprotection; **EC** – European Commission; **ENSI** - Swiss Federal Nuclear Safety Inspectorate; **FANC/AFCN** – Federal agency for nuclear control; **FOCP** - Swiss Federal Office for Civil Protection ; **ILT** - Inspectie Leefomgeving en Transport ; **Minez** - Ministerie van Economische Zaken; **NCRRP** - National Center of Radiobiology and Radiation Protection; **NRPA** – Norwegian Radiation Protection Authority; **OFSP** – Federal Office of Public Health; **ONR** - Office for Nuclear Regulation; **OSSKI** - National Research Institute for Radiobiology and Radiohygiene; **PHE** - Public Health England; **RPII** - Radiological Protection Institute of Ireland; **SNSA** - Slovenian Nuclear Safety Administration; **SSM** - Swedish Radiation Safety Authority; **STUK** - Radiation and Nuclear Safety Authority; **SUJB** - State Office for Nuclear Safety ; **UJD SR** - Urad Jadroveho Dozoru Slovenskey Republiky;

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Western European
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WENRA

REPORT

Safety Reference Levels for Existing Reactors

September 2014

Western European
WENRA RHWG
Nuclear Regulators Association

Report
WENRA Safety Reference Levels for Existing Reactors

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UPDATE IN RELATION TO LESSONS LEARNED FROM TEPCO FUKUSHIMA DAI-ICHI ACCIDENT

24th September 2014

WENRA RHWG
Western European Nuclear Regulators Association

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Foreword

A principal aim of the Western European Nuclear Regulators' Association (WENRA) is to develop a harmonized approach to nuclear safety within the member countries. One of the first major achievements to this end was the publication in 2006 of a set of safety reference levels (RLs) for operating nuclear power plants (NPPs).

The RLs are agreed by the WENRA members. They reflect expected practices to be implemented in the WENRA countries. As the WENRA members have different responsibilities, the emphasis of the RLs has been on nuclear safety, primarily focussing on safety of the reactor core and spent fuel. The RLs specifically exclude nuclear security and, with a few exceptions, radiation safety.

As RLs have been established for greater harmonization within WENRA countries, the areas and issues they address were selected to cover important aspects of nuclear safety where differences in substance between WENRA countries might be expected. They do not seek to cover everything that could have an impact upon nuclear safety or to form a basis for determining the overall level of nuclear safety in operating NPPs.

Given the various regulatory regimes and range of types of plants (PWR, BWR, CANDU and gas-cooled reactors) in operation in WENRA countries, the RLs do not go into legal and technical details. When needed, a reference to a relevant IAEA publication is inserted.

There are significant interactions between some of the issues and hence each issue should not necessarily be considered self-standing and the RLs need to be considered as a whole set.

WENRA is committed to continuous improvement of nuclear safety. To this end WENRA is committed to regularly revising the RLs when new knowledge and experience are available. In line with this policy the initial RLs were updated in 2007 and 2008. After the TEPCO Fukushima Dai-ichi nuclear accident, they have been further updated to take into account the lessons learned, including the insight from the EU stress tests. As a result a new issue on natural hazards was developed and significant changes made to several existing issues.

By issuing the revised RLs WENRA aims at further convergence of national requirements and safety improvements at NPPs in WENRA member countries, as necessary.

Stakeholders were asked for comments on the revised Reference Levels. All the comments were reviewed during the finalization process.

For further information, several documents on the WENRA website describe the basis used and processes followed to develop and update these RLs. Guidance on specific issues is also available on the WENRA website www.wenra.org.

For the explanation of the current update the accompanying report "Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned" was written and can also be downloaded from the WENRA website.

01
Issue A: Safety Policy

Safety area: Safety Management

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A1. Issuing and communication of a safety

A1.1 A written safety policy[1] shall be issued by the licensee.

A1.2 The safety policy shall be clear about giving safety an overriding priority in all plant activities.

A1.3 The safety policy shall include a commitment to continuously develop safety.

A1.4 The safety policy shall be communicated to all site personnel with tasks important to safety, in such a way that the policy is understood and applied.

A1.5 Key elements of the safety policy shall be communicated to contractors, in such a way that licensee's expectations and requirements are understood and applied in their activities.

A2. Implementation of the safety policy and monitoring safety performance

A2.1 The safety policy shall require directives for implementing the policy and monitoring safety performance.

A2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the plant management.

A2.3 The safety policy shall require continuous improvement of nuclear safety by means of:

- Identifying and analysing any new information with a timeframe commensurate to its safety significance;
- Regular[2] review of the overall safety of the nuclear power plant including the safety demonstration, taking into account operating experience, safety research, and advances in science and technology;
- Timely implementation of the reasonably practicable safety improvements identified.

Continuous improvement applies to all nuclear safety activities and hence it is relevant to all of the issues addressed in this document. Therefore, this requirement is not repeated in the other issues although it is applicable to all of them.

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[1] A safety policy is understood as a documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as a visible part of an integrated organisational policy.

[2] Regular is understood as an ongoing activity to review and analyse the plant design and operation and identify opportunities for improvement. Periodic safety review is a complementary tool to verify and follow up this activity in a longer perspective.

A3. Evaluation of the safety policy

A3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, more frequent than the periodic safety reviews.

02
Issue B: Operating Organisation

Safety area: Safety Management

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B1. Organisational structure

B1.1 The organisational structure for safe and reliable operation of the plant, and for ensuring an appropriate response in emergencies, shall be justified[3] and documented.

B1.2 The adequacy of the organisational structure, for its purposes according to B1.1, shall be assessed when organisational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated[4] after implementation.

B1.3 Responsibilities, authorities, and lines of communication shall be clearly defined and documented for all staff with duties important to safety.

B2. Management of safety and quality

B2.1 The licensee shall ensure that the plant is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

B2.2 The licensee shall ensure that decisions on safety matters are timely and preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

B2.3 The licensee shall ensure that the staff is provided with the necessary facilities and working conditions to carry out work in a safe manner.

B2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

B2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are analysed in a systematic way and continuously used to improve the plant and the licensee's activities.

B2.6 The licensee shall ensure that plant activities and processes are controlled through a documented management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

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3 The arguments shall be provided that the organisational structure supports safety and an appropriate response in emergencies.

4 A verification that the implementation of the organisational change has accomplished its safety objectives.

B3. Sufficiency and competency of staff

B3.1 The required number of staff for safe operation[5], and their competence, shall be analysed in a systematic and documented way.

B3.2 The sufficiency of staff for safe operation, their competence, and suitability for safety work shall be verified on a regular basis and documented.

B3.3 A long-term staffing plan[6] shall exist for activities that are important to safety.

B3.4 Changes to the number of staff, which might be significant for safety, shall be justified in advance, carefully planned and evaluated after implementation.

B3.5 The licensee shall always have in house, sufficient, and competent staff and resources to understand the licensing basis of the plant (e.g. Safety Analysis Report or Safety Case and other documents based thereon), as well as to understand the actual design and operation of the plant in all plant states.

B3.6 The licensee shall maintain, in house, sufficient and competent staff and resources to specify, set standards, manage and evaluate safety work carried out by contractors.

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5 Operation is defined as all activities performed to achieve the purpose for which a nuclear power plant was constructed (according to the IAEA Glossary).

6 Long term is understood as 3-5 years for detailed planning and at least 10 years for prediction of retirements etc.

03
Issue C: Management System

Safety area: Safety Management

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C1. Objectives

C1.1 An integrated management system shall be established, implemented, assessed and continually improved by the licensee. The main aim of the management system shall be to achieve and enhance nuclear safety by ensuring that other demands[7] on the licensee are not considered separately from nuclear safety requirements, to help preclude their possible negative impact on nuclear safety.

C2. General requirements

C2.1 The application of management system requirements shall be graded so as to deploy appropriate resources, on the basis of the consideration of:

- The significance and complexity of each activity and its products;
- The hazards and the magnitude of the potential impact associated with each activity and its products;
- The possible consequences if an activity is carried out incorrectly or a product fails.

C2.2 The documentation of the management system shall include the following:

- The policy statements of the licensee;
- A description of the management system;
- A description of the organisational structure of the licensee;
- A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
- A description of the interactions with relevant external organisations;
- A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved.

C2.3 The documentation of the management system shall be understandable to those who use it. Documents shall be up to date, readable, readily identifiable and available at the point of use.

C3. Management commitment

C3.1 The licensee shall develop the goals, strategies, plans and objectives of the organization in an integrated manner so that their collective impact on safety is understood and managed.

C3.2 The licensee shall ensure that it is clear when, how and by whom decisions are to be made within the management system.[8]

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[7] Examples of such demands are health, environmental, security, quality and economic requirements.
[8] With respect to operational decisions that impact on nuclear safety.

C3.3 The licensee shall ensure that management at all levels demonstrate its commitment to the establishment, implementation, assessment and continual improvement of the management system and shall allocate adequate resources to carry out these activities.

C3.4 The licensee shall foster the involvement of all staff in the implementation and continual improvement of the management system.

C4. Resources

C4.1 The licensee shall determine the amount of resources[9] necessary and shall provide the resources to carry out the activities of the licensee and to establish, implement, assess and continually improve the management system.

C5. Process implementation

C5.1 The processes[10] that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the licensee organisation shall be identified, and their development shall be planned, implemented, assessed and continually improved. The sequence and interactions of the processes shall be determined.

C5.2 The methods necessary to ensure the effectiveness of both the implementation and the control of the processes shall be determined and implemented.

C5.3 Documents[11] shall be controlled. Changes to documents shall be reviewed and recorded and shall be subject to the same level of approval as the documents themselves. It shall be ensured that document users are aware of and use appropriate and correct documents.

C5.4 Records shall be specified in the management system documentation and shall be controlled. All records shall, for the duration of the retention times specified for each record, be readable, complete, identifiable and easily retrievable.

C5.5 The control of processes, or work performed within a process, contracted to external organizations shall be identified within the management system. The licensee shall retain overall responsibility when contracting any processes or work performed within a process.

C5.6 Suppliers of products and services shall be selected on the basis of specified criteria and their performance shall be evaluated.

C5.7 Purchasing requirements shall be developed and specified in procurement documents. Evidence that products meet these requirements shall be available to the licensee before the product is used.

C5.8 It shall be confirmed[12] that activities and their products meet the specified requirements and shall ensure that products perform satisfactorily in service.

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9 "Resources" includes individuals, infrastructure, the working environment, information and knowledge, and suppliers, as well as material and financial resources.

10 This is not understood as a full process orientation of the management system. Also functional or organisational oriented routines and procedures could be used for certain activities together with cross cutting processes for other activities.

11 Documents may include: policies; procedures; instructions; specifications and drawings (or representations in other media); training materials; and any other texts that describe processes, specify requirements or establish product specifications.

12 Through inspection, testing, verification and validation activities before the acceptance, implementation, or operational use of products.

C6. Measurement, assessment and improvement

C6.1 In order to confirm the ability of the processes to achieve the intended results and to identify opportunities for improvement:

- The effectiveness of the management system shall be monitored and measured;
- The licensee shall ensure that managers carry out self-assessment of the performance of work for which they are responsible;
- Independent[13] assessments shall be conducted regularly on behalf of the licensee.

C6.2 An organizational unit shall be established with the responsibility for conducting independent assessments. This unit shall have sufficient authority to discharge its responsibilities. Individuals conducting independent assessments shall not assess their own work.

C6.3 The licensee shall evaluate the results of the assessments and take any necessary actions, and shall record and communicate inside the organisation the decisions and the reasons for the actions.

C6.4 A management system review shall be conducted at planned intervals to ensure the effectiveness of the management system.

C6.5 The causes of non-conformances shall be determined and remedial actions shall be taken to prevent their recurrence.

C6.6 Improvement plans shall include plans for the provision of adequate resources. Actions for improvement shall be monitored through to their completion and the effectiveness of the improvement shall be checked.

C7. Safety culture

C7.1 Management, at all levels in the licensee organization, shall consistently demonstrate, support, and promote attitudes and behaviours that result in an enduring and strong safety culture. This shall include ensuring that their actions discourage complacency, encourage an open reporting culture as well as a questioning and learning attitude with a readiness to challenge acts or conditions adverse to safety.

C7.2 The management system shall provide the means to systematically develop, support, and promote desired and expected attitudes and behaviours that result in a strong safety culture. The adequacy and effectiveness of these means shall be assessed as part of self-assessments and management system reviews.

C7.3 The licensee shall ensure that its suppliers and contractors whose operations may have a bearing on the safety of the nuclear facility comply with C7.1 and C7.2 to the appropriate extent.

[13] By an external organisation or by an internal independent assessment unit.

04
Issue D: Training and Authorization of NPP Staff (Jobs with Safety Importance)

Safety area: Safety Management

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D1. Policy

D1.1 The licensee shall establish an overall training policy and a comprehensive training plan on the basis of long-term competency needs and training goals that acknowledges the critical role of safety. The plan shall be kept up to date.

D1.2 A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

D2. Competence and qualification

D2.1 Only qualified persons that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The licensee shall ensure that all personnel performing safety-related duties including contractors have been adequately trained and qualified.

D2.2 The Licensee shall define and document the necessary competence requirements for their staff.

D2.3 Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

D2.4 Staff qualifying for positions important to safety shall undergo a medical examination to ensure their fitness depending upon the duties and responsibilities assigned to them. The medical examination shall be repeated at specified intervals.

D3. Training programmes and facilities

D3.1 Performance based training programmes shall be established for all staff with tasks important to safety. The programmes shall cover initial training in order to qualify for a certain position and regular refresher training.

D3.2 All technical staff including on-site contractors shall have a basic understanding of nuclear safety, radiation safety, fire safety, the on-site emergency arrangements and industrial safety.

D3.3 Representative simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures to be used during an accident is effective. The simulator shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions.[14]

D3.4 For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator.[15]

D3.5 Refresher training for control room operators shall include especially the following items as appropriate:

- Plant operation in normal operational states, selected anticipated operational occurrences and accident conditions;
- Shift crew teamwork;
- Operational experiences and modifications of plant and procedures.

D3.6 Maintenance and technical support staff including contractors shall have practical training on the required safety critical activities.

D4. Authorization

D4.1 Staff controlling changes in the operational status of the plant shall be required to hold an authorization valid for a specified time period. The licensee shall establish procedures for their staff to achieve this authorization. In the assessment of an individual's competence and suitability as a basis for the authorization, documented criteria shall be used.

D4.2 If an authorised individual:

- Moves to another position for which an authorization is required;
- Has been absent from the authorised position during an extended time period;

Re-authorisation shall be conducted after necessary individual preparations.

D4.3 Work carried out by contractor personnel on structures, systems, or components that are important to safety shall be approved and monitored by a suitably competent member of licensee's staff.

[14] This type of simulator is known as a full-scope simulator.
[15] Time includes the necessary briefings.

05
Issue E: Design Basis Envelope for Existing Reactors

Safety area: Design

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E1. Objective

E1.1 The design basis[16] shall have as an objective the prevention or, if this fails, the mitigation of consequences resulting from anticipated operational occurrences and design basis accidents. Design provisions shall be made to ensure that potential radiation doses to the public and the site personnel do not exceed prescribed limits and are as low as reasonably achievable.

E2. Safety strategy

E2.1 Defence-in-depth[17] shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases.

E2.2 The defence-in-depth concept shall be applied to provide several levels of defence including a design that provides a series of physical barriers to prevent uncontrolled releases of radioactive material to the environment, as well as a combination of safety features that contribute to the effectiveness of the barriers.

The design shall prevent as far as practicable:

- challenges to the integrity of the barriers;
- failure of a barrier when challenged;
- failure of a barrier as consequence of failure of another barrier.

E3. Safety functions

E3.1 During normal operation[18], anticipated operational occurrences and design basis accidents, the plant shall be able to fulfil the fundamental safety functions[19]:

- control of reactivity,
- removal of heat from the reactor core and from the spent fuel, and
- confinement of radioactive material.

16 The design basis shall be reviewed and updated during the lifetime of the plant (see RL E11.1).
17 For further information see IAEA SSR-2/1 (2012).
18 Normal operation includes start-up, power operation, shutting down, shutdown, maintenance, testing and refuelling.
19 Under the conditions specified in the following paragraphs.

E4. Establishment of the design basis

E4.1 The design basis shall specify the capabilities of the plant to cope with a specified range of plant states[20] within the defined radiation protection requirements. Therefore, the design basis shall include the specification for normal operation, anticipated operational occurrences and design basis accidents from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis.

E4.2 A list of PIEs shall be established to cover all events that could affect the safety of the plant. From this list, a set of anticipated operational occurrences and design basis accidents shall be selected using deterministic or probabilistic methods or a combination of both, as well as engineering judgement.[21] The resulting design basis events shall be used to set the boundary conditions according to which the structures, systems and components important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

E4.3 The design basis shall be systematically defined and documented to reflect the actual plant.

E5. Set of design basis events

E5.1 Internal events such as loss of coolant accidents, equipment failures, maloperation and internal hazards, and their consequential events, shall be taken into account in the design of the plant.[22] The list of events shall be plant specific and take account of relevant experience and analysis from other plants.

E5.2 External hazards shall be taken into account in the design of the plant. In addition to natural hazards[23], human made external hazards – including airplane crash and other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the nuclear power plant – shall as a minimum be taken into account in the design of the plant according to site specific conditions.

E6. Combination of events

E6.1 Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accidents, shall be considered in the design. Deterministic and probabilistic assessment as well as engineering judgement can be used for the selection of the event combinations.

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20 Normal operation, anticipated operational occurrences and design basis accident conditions.
21 Depending on the specific topic being under review, not all types of insights (deterministic, probabilistic or engineering judgement) may be relevant or needed.
22 Additional information on internal hazards is provided in IAEA Safety Standards NS-G-1.7 and NS-G-1.11.
23 See Issue T.

E7. Definition and application of technical acceptance criteria

E7.1 Initiating events shall be grouped into a limited number of categories that correspond to plant states20, according to their probability of occurrence. Radiological and technical acceptance criteria shall be assigned to each plant state such that frequent initiating events shall have only minor or no radiological consequences and that events that may result in severe consequences shall be of very low frequency.

E7.2 Criteria for protection of the fuel rod integrity, including fuel temperature, Departure from Nucleate Boiling (DNB), and cladding temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis accident.

E7.3 Criteria for the protection of the primary coolant pressure boundary shall be specified, including maximum pressure, maximum temperature, thermal- and pressure transients and stresses.

E7.4 If applicable, criteria in E7.3 shall be specified as well for protection of the secondary coolant system.

E7.5 Criteria shall be specified for protection of containment, including temperatures, pressures and leak rates.

E8. Demonstration of reasonable conservatism and safety margins

E8.1 The initial and boundary conditions shall be specified with conservatism.

E8.2 The worst single failure[24] shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE.

E8.3 Only systems that are suitably safety classified can be credited to carry out a safety function. Non safety classified systems shall be assumed to operate only if they aggravate the effect of the initiating event[25].

E8.4 A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis accidents.[26]

E8.5 The safety systems shall be assumed to operate at their performance level that is most penalising for the initiator.

E8.6 Any failure, occurring as a consequence of a postulated initiating event, shall be regarded to be part of the original PIE.

E8.7 The safety analysis shall:

(a) rely on methods, assumptions or arguments which are justified and conservative;

(b) provide assurance that uncertainties and their impact have been given adequate consideration[27];

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[24] A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.

[25] This means that non safety classified systems are either supposed not to function after the initiator, either supposed to continue to function as before the initiator, depending on which of both cases is most penalising.

[26] This assumption is made to ensure the sufficiency of the shutdown margin. The stuck rod selected is the highest worth rod at Hot Zero Power and conservative values of reactor trip reactivity (conservative time delay and reactivity versus control rod position dependence) are used. A stuck rod can be handled as single failure in the analysis of design basis accidents (DBAs) if the stuck rod itself is the worst single failure.

(c) give evidence that adequate margins have been included when defining the design basis to ensure that all the design basis events are covered;

(d) be auditable and reproducible.

E9. Design of safety functions

General

E9.1 The fail-safe principle shall be considered in the design of systems and components important to safety.

E9.2 A failure in a system intended for normal operation shall not affect a safety function.

E9.3 Activations and control of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes of the initiating event. Any operator actions required by the design within 30 minutes of the initiating event shall be justified.[28]

E9.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components[29], redundancy, diversity[30], physical and functional separation and isolation.

E9.5 For sites with multiple units, appropriate independence between them shall be ensured.[31]

Reactor and fuel storage sub-criticality

E9.6 The means for shutting down the reactor shall consist of at least two diverse systems.

E9.7 At least one of the two systems shall, on its own, be capable of quickly[32] rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.

E9.8 Sub-criticality shall be ensured and sustained:

- in the reactor after planned reactor shutdown during normal operation and after anticipated operational occurrences, as long as needed;
- in the reactor, after a transient period (if any) following a design basis accident[33];
- for fuel storage during normal operation, anticipated operational occurrences, and design basis accidents.

[27] Conservative assumptions, safety factors, uncertainty and sensitivity analysis are means to address uncertainties and their impact on safety assessment.

[28] The control room staff has to be given sufficient time to understand the situation and take the correct actions. Operator actions required by the design within 30 min after the initiating event have to be justified and supported by clear documented procedures that are regularly exercised in a full scope simulator.

[29] Proven by experience under similar conditions or adequately tested and qualified.

[30] The potential for common cause failure, including common mode failure, shall be appropriately considered to achieve the necessary reliability.

[31] The possibility of one unit supporting another could be considered as far as this is not detrimental for safety.

[32] Within 4-6 seconds, i.e. scram system.

[33] Technical acceptance criteria have to be fulfilled during a transient period for which sub-criticality is not ensured.

Heat removal functions

E9.9 Means for removing residual heat from the core after shutdown and from spent fuel storage, during and after anticipated operational occurrences and design basis accidents, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Confinement functions

E9.10 A containment system shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. This system shall include:

- leaktight structures covering all essential parts of the primary system;
- associated systems for control of pressures and temperatures;
- features for isolation;
- features for the management and removal of fission products, hydrogen, oxygen and other substances that could be released into the containment atmosphere.

E9.11 Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

E9.12 Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

E10. Instrumentation and control systems

E10.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, the containment, and the state of the spent fuel storage. Instrumentation shall also be provided for obtaining any information on the plant necessary for its reliable and safe operation, and for determining the status of the plant in design basis accidents. Provision shall be made for automatic recording[34] of measurements of any derived parameters that are important to safety.

E10.2 Instrumentation shall be adequate for measuring plant parameters and shall be environmentally qualified for the plant states concerned.

[34] By computer sampling and/or print outs.

Control room

E10.3 A main control room shall be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences and design basis accidents.

E10.4 Devices shall be provided to give in an efficient way visual and, if appropriate, also audible indications of operational states and processes that have deviated from normal and could affect safety. Ergonomic factors shall be taken into account in the design of the main control room. Appropriate information shall be available to the operator to monitor the effects of the automatic actions.

E10.5 Special attention shall be given to identifying those events, both internal and external to the main control room, which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

E10.6 For times when the main control room is not available, there shall be sufficient monitoring and control equipment available, preferably at a single location that is physically, electrically and functionally separate from the main control room, so that, if the main control room is unavailable, the reactor can be placed and maintained in a shut down state, residual heat can be removed from the reactor and spent fuel storage, and the essential plant parameters, including the conditions in the spent fuel storages, can be monitored.

Reactor protection system

E10.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:
- no single failure results in loss of protection function; and
- the removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

E10.8 The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in operation. Exceptions shall be justified.

E10.9 The design of the reactor protection system shall minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal operation and anticipated operational occurrences. Furthermore, the reactor protection system shall not prevent operators from taking correct actions if necessary in design basis accidents.

E10.10 Computer based systems used in a protection system, shall fulfil the following requirements:
- the highest quality of and best practices for hardware and software shall be used;
- the whole development process, including control, testing and commissioning of design changes, shall be systematically documented and reviewed;
- in order to confirm confidence in the reliability of the computer based systems, an assessment of the computer based system by expert personnel independent of the designers and suppliers shall be undertaken; and

- where the necessary integrity of the system cannot be demonstrated with a high level of confidence, a diverse means of ensuring fulfilment of the protection functions shall be provided.

Emergency power

E10.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

E11. Review of the design basis

E11.1 The actual design basis shall regularly[35], and when relevant as a result of operating experience and significant new safety information[36], be reviewed, using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the design basis is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

[35] See RL A2.3.
[36] Significant new safety information is understood as new insights gained from e.g. site evaluation, safety analyses and the development of safety standards and practices.

06
Issue F: Design Extension of Existing Reactors

Safety area: Design

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F1. Objective

F1.1 As part of defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety of the nuclear power plant by:

- enhancing the plant's capability to withstand more challenging events or conditions than those considered in the design basis,
- minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.

F1.2 There are two categories of DEC:

- DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;
- DEC B with postulated severe fuel damage.

The analysis shall identify reasonably practicable provisions that can be implemented for the prevention of severe accidents. Additional efforts to this end shall be implemented for spent fuel storage with the goal that a severe accident in such storage becomes extremely unlikely to occur with a high degree of confidence.

In addition to these provisions, severe accidents shall be postulated for fuel in the core and, if not extremely unlikely to occur with a high degree of confidence, for spent fuel in storage, and the analysis shall identify reasonably practicable provisions to mitigate their consequences.

F2. Selection of design extension conditions

F2.1 A set of DECs shall be derived and justified as representative, based on a combination of deterministic and probabilistic assessments as well as engineering judgement.

F2.2 The selection process for DEC A shall start by considering those events and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover:

- Events occurring during the defined operational states of the plant;
- Events resulting from internal or external hazards;
- Common cause failures.

Where applicable, all reactors and spent fuel storages on the site have to be taken into account. Events potentially affecting all units on the site, potential interactions between units as well as interactions with other sites in the vicinity shall be covered.

F2.3 The set of category DEC B events shall be postulated and justified to cover situations, where the capability of the plant to prevent severe fuel damage is exceeded or where measures provided are assumed not to function as intended, leading to severe fuel damage.

F3. Safety analysis of design extension conditions

F3.1 The DEC analysis shall:

(a) rely on methods, assumptions or arguments which are justified[37], and should not be unduly conservative;

(b) be auditable, paying particular attention where expert opinion is utilized, and take into account uncertainties and their impact;

(c) identify reasonably practicable provisions to prevent severe fuel damage (DEC A) and mitigate severe accidents (DEC B);

(d) evaluate potential on-site and off-site radiological consequences resulting from the DEC (given successful accident management measures);

(e) consider plant layout and location, equipment capabilities, conditions associated with the selected scenarios and feasibility of foreseen accident management actions;

(f) demonstrate, where applicable, sufficient margins to avoid "cliff-edge effects"[38] that would result in unacceptable consequences, i.e. for DEC-A severe fuel damage and for DEC-B a large or early radioactive release.

(g) reflect insights from PSA level 1 and 2;

(h) take into account severe accident phenomena, where relevant;

(i) define an end state, which should where possible be a safe state, and, when applicable, associated mission times for SSCs.

F4. Ensuring safety functions in design extension conditions

General

F4.1 In DEC A, it is the objective that the plant shall be able to fulfil, the fundamental safety functions:

- control of reactivity[39],
- removal of heat from the reactor core and from the spent fuel, and
- confinement of radioactive material.

In DEC B, it is the objective that the plant shall be able to fulfil confinement of radioactive material. To this end removal of heat from the damaged fuel shall be established.[40]

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[37] These methods can be more realistic up to best estimate. Modified acceptance criteria may be used in the analysis.

[38] A cliff edge effect occurs when a small change in a condition (a parameter, a state of a system…) leads to a disproportionate increase in consequences.

[39] Preferably, this safety function shall be fulfilled at all times; if it is lost, it shall be re-established after a transient period.

[40] For the fulfilment (or re-establishment) of the fundamental safety functions in DEC A and DEC B, the use of mobile equipment on-site can be taken into account, as well as support from off-site, with due consideration for the time required for it to be available.

F4.2 It shall be demonstrated that SSCs[41] (including mobile equipment and their connecting points, if applicable) for the prevention of severe fuel damage or mitigation of consequences in DEC have the capacity and capability and are adequately qualified to perform their relevant functions for the appropriate period of time.

F4.3 If accident management relies on the use of mobile equipment, permanent connecting points, accessible (from a physical and radiological point of view) under DEC, shall be installed to enable the use of this equipment. The mobile equipment, and the connecting points and lines shall be maintained, inspected and tested.

F4.4 A systematic process shall be used to review all units relying on common services and supplies (if any), for ensuring that common resources of personnel, equipment and materials expected to be used in accident conditions are still effective and sufficient for each unit at all times. In particular, if support between units at one site is considered in DEC, it shall be demonstrated that it is not detrimental to the safety of any unit.

F4.5 The NPP site shall be autonomous regarding supplies supporting safety functions for a period of time until it can be demonstrated with confidence that adequate supplies can be established from off site.

Long-term sub-criticality

F4.6 In design extension conditions, sub-criticality of the reactor core shall be ensured in the long term[42] and in the fuel storage at any time.

Heat removal functions

F4.7 There shall be sufficient independent and diverse means including necessary power supplies available to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.

Confinement functions

F4.8 Isolation of the containment shall be possible in DEC. For those shutdown states where this cannot be achieved in due time, severe core damage shall be prevented with a high degree of confidence.

If an event leads to bypass of the containment, severe core damage shall be prevented with a high degree of confidence.

F4.9 Pressure and temperature in the containment shall be managed.

F4.10 The threats due to combustible gases shall be managed.

F4.11 The containment shall be protected from overpressure.

If venting is to be used for managing the containment pressure, adequate filtration shall be provided.

F4.12 High pressure core melt scenarios shall be prevented.

[41] SSCs including their support functions and related instrumentation.

[42] It is acknowledged that in case of DEC B, sub-criticality might not be guaranteed during core degradation and later on during some time in a fraction of the corium.

F4.13 Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

F4.14 In DEC A, radioactive releases shall be minimised as far as reasonably practicable.

In DEC B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

(a) allow sufficient time for protective actions (if any) in the vicinity of the plant; and

(b) avoid contamination of large areas in the long term.

Instrumentation and control for the management of DEC

F4.15 Adequately qualified instrumentation shall be available for DEC for determining the status of plant (including spent fuel storage) and safety functions as far as required for making decisions.[43]

F4.16 There shall be an operational and habitable control room (or another suitably equipped location) available during DEC in order to manage such situations.

Emergency power

F4.17 Adequate power supplies during DEC shall be ensured considering the necessary actions and the timeframes defined in the DEC analysis, taking into account external hazards.

F4.18 Batteries shall have adequate capacity to provide the necessary DC power until recharging can be established or other means are in place.

F5. Review of the design extension conditions

F5.1 The design extension conditions shall regularly[44], and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the selection of design extension conditions is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

[43] This refers to decisions concerning measures on-site as well as, in case of DEC B, off-site.
[44] See RL A2.3.

07
Issue G: Safety Classification of Structures, Systems and Components

Safety area: Design

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G1. Objective

G1.1 All SSCs[45] important to safety shall be identified and classified on the basis of their importance for safety.

G2. Classification process

G2.1 The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment.

G2.2 The classification shall identify for each safety class:
- The appropriate codes and standards in design, manufacturing, construction and inspection;
- Need for emergency power supply, qualification to environmental conditions;
- The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;
- The applicable quality requirements

G3. Ensuring reliability

G3.1 SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.

G3.2 The failure of a SSC in one safety class shall not cause the failure of other SSCs in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.

G4. Selection of materials and qualification of equipment

G4.1 The design of SSCs important to safety and the materials used shall take into account the effects of operational conditions over the lifetime of the plant and, when required, the effects of accident conditions on their characteristics and performance.

G4.2 Qualification procedures shall be adopted to confirm that SSCs important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions[46] over the lifetime of the plant and when required in anticipated operational occurrences and accident conditions.

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45 SSCs include software for I&C.
46 Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof.

08

Issue H: Operational Limits and Conditions (OLCs)

Safety area: Operation

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H1. Purpose

H1.1 OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intentions as documented in the Safety Analysis Report (SAR).

H1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

H2. Establishment and review of OLCs

H2.1 Each established OLC shall be justified based on plant design, safety analysis and commissioning tests.

H2.2 OLCs shall be kept updated and reviewed in the light of experience, the current state of science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.

H2.3 The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review.

H3. Use of OLCs

H3.1 The OLCs shall be readily accessible to control room personnel.

H3.2 Control room operators shall be highly knowledgeable of the OLCs and their technical basis. Relevant operational decision makers shall be aware of their significance for the safety of the plant.

H4. Scope of OLCs

H4.1 OLCs shall cover all operational plant states including power operation, shutdown and refuelling, any intermediate conditions between these states and temporary situations arising due to maintenance and testing.

H5. Safety limits, safety systems settings and operational limits

H5.1 Adequate margins shall be ensured between operational limits and the established safety systems settings, to avoid undesirably frequent actuation of safety systems.

H5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.

H6. Unavailability limits

H6.1 Limits and conditions for normal operation shall include limits on operating parameters, stipulation for minimum amount of operable equipment, actions to be taken by the operating staff in the event of deviations from the OLCs and time allowed to complete these actions.

H6.2 Where operability requirements cannot be met, the actions to bring the plant to a safer state shall be specified, and the time allowed to complete the action shall be stated.

H6.3 Operability requirements shall state for the various modes of normal operation the number of systems or components important to safety that should be in operating condition or standby condition.

H7. Unconditional requirements

H7.1 If operating personnel cannot ascertain that the power plant is operating within operating limits, or the plant behaves in an unexpected way, measures shall be taken without delay to bring the plant to a safe and stable state.

H7.2 Plant shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

H8. Staffing levels

H8.1 Minimum staffing levels for shift staff shall be stated in the OLCs.

H9. Surveillance

H9.1 The licensee shall ensure that an appropriate surveillance[47] program is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

H10. Non-compliance

H10.1 In cases of non-compliance with OLC, remedial actions shall be taken immediately to re-establish compliance with OLC requirements.

H10.2 Reports of non-compliance shall be investigated and corrective action shall be implemented in order to help prevent such non-compliance[48] in future.

[47] The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the surveillance programme include not only deficiencies in SSCs and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the plant is deviating from the design intent. (NS-G-2.6 Para 2.11)

[48] If the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, plant shall be deemed to have operated in non-compliance with OLCs.

09
Issue I: Ageing Management

Safety area: Operation

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I1. Objective

I1.1. The operating organisation shall have an Ageing Management Programme[49] (AMP) to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.

I2. Technical requirements, methods and procedures

I2.1 The licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

I2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.

I2.3. The Periodic Safety Reviews shall be used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.

I2.4. In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.

I2.5. The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.

I3. Major structures and components

I3.1. Ageing management of the reactor pressure vessel[50] and its welds shall take all relevant factors including embrittlement, thermal ageing, and fatigue into account to compare their performance with prediction, throughout plant life.

I3.2. Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

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[49] Ageing is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out). An Ageing Management Programme (AMP) should be understood as an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components.

[50] Or its functional equivalent in other designs.

10

Issue J: System for Investigation of Events and Operational Experience Feedback

Safety area: Operation

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J1. Programmes and Responsibilities

J1.1 The licensee shall establish and conduct a programme to collect, screen, analyse, and document operating experience and events at the plant in a systematic way. Relevant operational experience and events reported by other plants shall also be considered.

J1.2 Operating experience at the plant shall be evaluated to identify any latent safety relevant failures or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margin.

J1.3 The licensee shall designate staff for carrying out these programmes, for the dissemination of findings important to safety and – where appropriate – for recommendations on actions to be taken. Significant findings and trends shall be reported to the licensee's top management.

J1.4 Staff responsible for evaluation of operational experience and investigation into events shall receive adequate training, resources, and support from the line management.

J1.5 The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

J2. Collection and storage of information

J2.1 The information relevant to experience from normal and abnormal operation and other important safety-related information shall be organized, documented, and stored in such a way that it can be easily retrieved and systematically searched, screened and assessed by the designated staff.

J3. Reporting and dissemination of safety significant information

J3.1 The licensee shall report events of significance to safety in accordance with established procedures and criteria.

J3.2 Plant personnel shall be required to report abnormal events and be encouraged to report internally near misses relevant to the safety of the plant.

J3.3 Information resulting from the operational experience shall be disseminated to relevant staff and shared with relevant national and international bodies.

J3.4 A process shall be put in place to ensure that operating experience of events at the plant concerned as well as of relevant events at other plants is appropriately considered in the training programme for staff with tasks related to safety.

J4. Assessment and investigation of events

J4.1 An initial assessment of events important to safety shall be performed without delay to determine whether urgent actions are necessary.

J4.2 The licensee shall have procedures specifying appropriate investigation methods, including methods of human performance analysis.

J4.3 Event investigation shall be conducted on a time schedule consistent with the event significance. The investigation shall:

- Establish the complete event sequence;
- Determine the deviation;
- Include direct and root cause analysis;
- Assess the safety significance including potential consequences; and
- Identify corrective actions.

J4.4 The operating organisation shall maintain liaison as appropriate with the organizations (manufacturer, research organization, designer) involved in design and construction, with the aims of feeding back information on operating experience and obtaining advice, if necessary, in case of equipment failures or abnormal events.

J4.5 As a result of the analysis, timely corrective actions shall be taken such as technical modifications, administrative measures or personnel training to restore safety, to avoid event recurrence and where appropriate to improve safety.

J5. Review and continuous improvement of the OEF process

J5.1 Periodic reviews of the effectiveness of the OEF process based on performance criteria shall be undertaken and documented either within a self-assessment programme by the licensee or by a peer review team.

11
Issue K: Maintenance, In-Service Inspection and Functional Testing

Safety area: Operation

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K1. Scope and objectives

K1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of SSCs important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. They shall take into account operational limits and conditions and be re-evaluated in the light of experience.

K1.2 The programmes shall include periodic inspections and tests of SSCs important to safety in order to determine whether they are acceptable for continued safe operation of the plant or whether any remedial measures are necessary.

K2. Programme establishment and review

K2.1 The extent and frequency of preventive maintenance, testing, surveillance and inspection of SSCs shall be determined through a systematic approach on the basis of:

- Their importance to safety;
- Their inherent reliability;
- Their potential for degradation (based on operating experience, research and vendor recommendation);
- Operational and other relevant experience and results of condition monitoring.

K2.2 In-service inspections of nuclear power plants shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

K2.3 Data on maintenance, testing, surveillance, and inspection of SSCs shall be recorded, stored and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

K2.4 The maintenance programme shall be periodically reviewed[51] in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on plant safety, and their conformance with applicable requirements.

K2.5 The potential impact of maintenance upon plant safety shall be assessed.

[51] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

K3. Implementation

K3.1 SSCs important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

K3.2 Procedures shall be established, reviewed, and validated for maintenance, testing, surveillance and inspection tasks.

K3.3 A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

K3.4 Before equipment is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

K3.5 The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined in the procedures.

K3.6 Repairs to SSCs shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

K3.7 Following any event due to which the safety functions and functional integrity of any component or system may have been challenged, the licensee shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

K3.8 The reactor coolant pressure boundary shall be subject to a system leakage test before resuming operation after a reactor outage in the course of which its leak-tightness may been affected.

K3.9 The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.

K3.10 All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with requirements of the management system.

K3.11 Any in-service inspection (ISI) process shall be qualified[52], in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

[52] The ISI system qualification means to demonstrate that the combination of equipment, inspection procedure and personnel is appropriate for testing of a given inspection area according to a technical specification. It is recommended to use as reference documents e.g. the European Regulators Common Position on NDT Qualification, ENIQ methodology and/or IAEA – EBP-VVER-11 documents.

K3.12 When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.

K3.13 Surveillance measures to verify the containment integrity shall include: a) leak rate tests; b) tests of penetration seals and closure devices such as air locks and valves that are part of the boundaries, to demonstrate their leak-tightness and, where appropriate, their operability; c) inspections for structural integrity (such as those performed on liner and pre-stressing tendons).

12
Issue LM: Emergency Operating Procedures and Severe Accident Management Guidelines

Safety area: Operation

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LM1. Objectives

LM1.1 A comprehensive set of procedures and guidelines, including emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs) shall be provided, covering accident conditions initiated during all operational states.

LM2. Scope

LM2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the plant state to a safe condition.

LM2.2 EOPs, with other specific procedures or guidelines when applicable, shall be provided to cover DEC A. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent severe fuel damage in the core or in the spent fuel storage.

LM2.3 SAMGs, with other specific procedures or guidelines when applicable, shall be provided to mitigate the consequences of severe accidents for the cases where the responses to events including the measures provided by EOPs have not been successful in the prevention of severe fuel damage.

LM2.4 EOPs for design basis accidents shall be symptom based or a combination of symptom based and event based[53] procedures. EOPs for DEC A shall be symptom based unless an event based approach can be justified.

LM2.5 The set of procedures and guidelines shall be suitable to manage accident conditions that simultaneously affect the reactor and spent fuel storages, and shall take potential interactions between reactor and spent fuel storages into account.

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[53] Event-based EOPs enable the operator to identify the specific event and encompass:
- Information for determining the status of the plant,
- Automatic actions that will probably be taken as a result of the event,
- Subsequent operator actions directed to returning the reactor to a normal condition or to provide for safe, extended and stable shutdown conditions.

Symptom-based EOPs enable the operator to respond to situations for which there are no procedures to identify accurately the event that has occurred. The decisions for measures to respond to such situations are specified in the procedures with respect to the symptoms and the state of systems of the plant (such as the values of safety parameters and critical safety functions).

LM2.6 Possibilities for one unit, without compromising its safety, supporting another unit on the site shall be covered by the set of procedures and guidelines.

LM2.7 The set of procedures and guidelines shall be such that they are able to be implemented even if all nuclear installations on a site are under accident conditions, taking into account the dependencies between the systems and common resources.

LM3. Format and Content of Procedures and Guidelines

LM3.1 EOPs shall be developed in a systematic way and shall be supported by realistic and plant specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, such as alarm response procedures and severe accident management guidelines.

LM3.2 EOPs shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate among EOPs and to proceed from EOPs to SAMGs.

LM3.3 SAMGs shall be developed in a systematic way using a plant specific approach. SAMGs shall address strategies to cope with scenarios identified by the severe accident analyses.[54]

LM3.4 EOPs for design basis accidents shall rely on adequately qualified equipment and instrumentation. EOPs for DEC and SAMGs shall primarily rely on adequately qualified equipment.

LM3.5 The set of procedures and guidelines shall consider the anticipated on-site conditions, including radiological conditions, associated with the accident conditions they are addressing and the initiating event or hazard that might have caused it.

LM4. Verification and validation

LM4.1 The set of procedures and guidelines shall be verified and validated in the form in which they will be used in the field, as far as practicable, to ensure that they are administratively and technically correct for the plant, are compatible with the environment in which they will be used[55] and with the human resources available.

LM4.2 The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations, using a simulator, where appropriate.

[54] Analysis aimed at identifying the plant vulnerabilities to severe accident phenomena, assessment of plant capabilities and development of accident management measures, including for containment protection as defined in Issue F (Design Extension of Existing Reactors) in RLs F4.8 to F4.14. It is understood that for these accident conditions also SAMGs shall be developed.

[55] In particular, expected manual operation of equipment shall be possible.

LM5. Review and updating

LM5.1 The set of procedures and guidelines shall be kept updated to ensure that they remain fit for their purpose.

LM6. Training and exercises

LM6.1 Control room staff shall be regularly trained and exercised, using full-scope simulators for the EOPs and simulators, where practicable, for the SAMGs.

LM6.2 Licensee emergency response staff shall be regularly trained and exercised, commensurate with their expected role in managing an emergency, for situations and conditions covered by the set of procedures and guidelines.

LM6.3 The transition from EOPs to SAMGs for management of severe accidents shall be regularly exercised.

LM6.4 Interventions called for in the set of procedures and guidelines and needed to restore necessary safety functions, including those which may rely on mobile or off-site equipment, shall be planned for and regularly exercised. The potential unavailability of instruments, lighting and power and the use of protective equipment shall be considered.

13
Issue N: Contents and Updating of Safety Analysis Report (SAR)

Safety area: Safety Verification

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N1. Objective

N1.1 The Licensee shall provide a SAR[56] to demonstrate that the plant fulfils relevant safety requirements and use it as a basis for continuous support of safe operation.

N1.2 The Licensee shall use the SAR as a basis for assessing the safety implications of changes to the plant, or to operating practices.

N2. Content of the SAR

N2.1 The SAR shall describe the site, the plant layout and normal operation and demonstrate how safety is achieved.

N2.2 The SAR shall contain detailed descriptions of the safety functions; all safety systems and safety-related structures, systems and components; their design basis and functioning in all operational states, including shut down and accident conditions.

N2.3 The SAR shall identify applicable regulations codes and standards.

N2.4 The SAR shall describe the relevant aspects of the plant organization and the management of safety.

N2.5 The SAR shall contain the evaluation of the safety aspects related to the site.

N2.6 The SAR shall outline the general design concept and the approach adopted to meet the fundamental safety objectives.

N2.7 The SAR shall include justification that it adequately demonstrates that the plant fulfils relevant safety requirements. The SAR shall describe the safety analyses performed to assess the safety of the plant in response to anticipated operational occurrences, design basis accidents and design extension conditions against safety criteria and radiological release limits. Safety margins shall be described.

N2.8 The SAR shall describe the emergency operation procedures and severe accident management guidelines, the inspection and testing provisions, the qualification, and training of personnel, the operational experience feedback programme, and the management of ageing.

N2.9 The SAR shall contain the technical bases for the operational limits and conditions.

N2.10 The SAR shall describe the policy, strategy, methods, and provisions for radiation protection.

[56] A consistent safety document or integrated set of documents constituting the licensing basis of the plant and updated under supervision of the regulatory body.

N2.11 The SAR shall describe the on-site emergency preparedness arrangements and the liaison and co-ordination with off-site organizations involved in the response to an emergency.

N2.12 The SAR shall describe the on-site radioactive waste management provisions.

N2.13 The SAR shall describe how the relevant decommissioning and end-of-life aspects are taken into account during operation.[57]

N2.14 The descriptions, assessments and arrangements mentioned in the SAR shall consider the site as a whole, to take into account hazards:

- which may challenge all installations within a short period of time;
- which arise from harmful interactions between installations.

N3. Review and update of the SAR

N3.1 The licensee shall update the SAR to reflect modifications, new regulatory requirements, new information relevant for the safety assessment (including those related to characteristics of the site and the site environment), and relevant standards, in a timely manner after the new information is available and applicable.

[57] Guidance on the specific aspects that need to be addressed in the SAR is given in Chapter XV of the IAEA Safety Guide GS-G-4.1.

14
Issue O: Probabilistic Safety Analysis (PSA)

Safety area: Safety Verification

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O1. Scope and content of PSA

O1.1 For each plant design, a specific PSA shall be developed for level 1 and level 2, considering all relevant[58] operational states, covering fuel in the core and in the spent fuel storage and all relevant internal and external initiating events. External hazards shall be included in the PSA for level 1 and level 2 as far as practicable, taking into account the current state of science and technology. If not practicable, other justified methodologies shall be used to evaluate the contribution of external hazards to the overall risk profile of the plant.

O1.2 PSA shall include relevant dependencies.[59]

O1.3 The Level 1 PSA shall contain sensitivity and uncertainty analyses. The Level 2 PSA shall contain sensitivity analyses and, as appropriate, uncertainty analyses.

O1.4 PSA shall be based on a realistic modelling of plant response, using data relevant for the design, and taking into account human action to the extent assumed in operating and accident procedures. The mission times in the PSA shall be justified.

O1.5 Human reliability analysis shall be performed, taking into account the factors which can influence the performance of plant staff in all plant states.

O2. Quality of PSA

O2.1 PSA shall be performed, documented, and maintained according to requirements of the management system of the licensee.

O2.2 PSA shall be performed according to an up to date proven methodology, taking into account international experience currently available.

O3. Use of PSA

O3.1 PSA shall be used to support safety management. The role of PSA in the decision making process shall be defined.

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[58] Relevant means that the considered initiating event (or operational state) is relevant for the risk as determined with the PSA. Adequate screening criteria shall be defined in order to identify the relevant initiating events and operational states.

[59] Such as functional dependencies, area dependencies (based on the physical location of the components, systems and structures) and other common cause failures. Site aspects and interaction with other units could also be relevant.

O3.2 PSA shall be used[60] to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.

O3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects".

O3.4 PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.

O3.5 Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.

O3.6 The results of PSA shall be used to ensure that the items are included in the verification and test programmes if they contribute significantly to risk.

O4. Demands and conditions on the use of PSA

O4.1 The limitations of PSA shall be understood, recognized and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.

O4.2 When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.

O4.3 The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the SAR.

[60] It is intended that such analyses will be done on a continuous basis, not just every ten years during the Periodic Safety Review.

15
Issue P: Periodic Safety Review (PSR)

Safety area: Safety Verification

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P1. Objective of the periodic safety review

P1.1 The licensee shall have the prime responsibility for performing the Periodic Safety Review.

P1.2 The review shall confirm the compliance of the plant with its licensing basis and any deviations shall be resolved.

P1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices taking into account operating experience, relevant research findings, and the current state of technology.

P1.4 All reasonably practicable improvement measures shall be implemented by the licensee as a result of the review, in a timely manner.

P1.5 An overall assessment of the safety of the plant covering the period until the next PSR shall be provided, and adequate confidence in plant safety for continued operation demonstrated, based on the results of the review in each area. This assessment shall highlight any issues that might limit the future safe operation of the plant and explain how they will be managed.

P2. Scope of the periodic safety review

P2.1 The review shall be made periodically, at least every ten years.

P2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant and, as a minimum the following safety factors shall be covered by the review[61]:

(a) Plant design;

(b) Actual condition of structures, systems and components (SSCs) important to safety;

(c) Equipment qualification;

(d) Ageing;

(e) Deterministic safety analysis;

(f) Probabilistic safety assessment;

(g) Hazard analysis;

[61] Radiation protection is not regarded as a separate safety factor since it is related to most of the other safety factors. As far as there are other units at the site, interactions between them should also be covered by the review.

(h) Safety performance;

(i) Use of experience from other plants and research findings;

(j) Organization, the management system and safety culture;

(k) Procedures;

(l) Human factors;

(m) Emergency planning;

(n) Radiological impact on the environment.

P3. Methodology of the periodic safety review

P3.1 The review shall use an up to date, systematic, and documented methodology, taking into account deterministic as well as probabilistic assessments.

P3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices. The safety significance of all findings shall be evaluated using an appropriate approach. A global assessment shall consider all findings (positive and negative) and their cumulative effect on safety, and shall identify what safety improvements are reasonably practicable.

16
Issue Q: Plant Modifications

Safety area: Operation

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Q1. Purpose and scope

Q1.1 The licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely.[62]

Q1.2 The licensee shall control plant modifications using a graded approach with appropriate criteria for categorization according to their safety significance.[63]

Q2. Procedure for dealing with plant modifications

Q2.1 The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met.

Q2.2 For modifications to SSC, this process shall include the following:

- Reason and justification for modification;
- Design;
- Safety assessment;
- Updating plant documentation and training;
- Fabrication, installation and testing; and
- Commissioning the modification.

Q3. Requirements on safety assessment and review of modifications

Q3.1 An initial safety assessment shall be carried out to determine any consequences for safety.[64]

Q3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

Q3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

Q3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by personnel not immediately involved in their design or implementation.

[62] RL Q2.2 specifically addresses modifications to SSCs, all other RLs relate to all type of modifications in the sense of IAEA SSR-2/2, Para 4.39.

[63] Para 4.5 of IAEA Guide NS-G-2.3 contains information about possible categories.

[64] This assessment is performed for the purpose of categorizing the intended modification according to its safety significance.

Q4. Implementation of modifications

Q4.1 Implementation and testing of plant modifications shall be performed in accordance with the applicable work control and plant testing procedures.

Q4.2 The impact upon procedures, training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated.

Q4.3 Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated.

Q5. Temporary modifications[65]

Q5.1 All temporary modifications shall be clearly identified at the point of application and at any relevant control position.[66] Operating personnel shall be clearly informed of these modifications and of their consequences for the operation of the plant.

Q5.2 Temporary modifications shall be managed according to specific plant procedures.

Q5.3 The number of simultaneous temporary modifications shall be kept to a minimum. The duration of a temporary modification shall be limited.

Q5.4 The licensee shall periodically review outstanding temporary modifications to determine whether they are still needed.

[65] Examples of temporary modifications are temporary bypass lines, electrical jumpers, lifted electrical leads, temporary trip point settings, temporary blank flanges and temporary defeats of interlocks. This category of modifications also includes temporary constructions and installations used for maintenance of the design basis configuration of the plant in emergencies or other unanticipated situations. Temporary modifications in some cases may be made as an intermediate stage in making permanent modifications. IAEA Guide NS-G-2.3, Para 6.1.

[66] By relevant control position it is meant any control point important for the modified system and also any administrative aspect related to the system in which the temporary modification has been implemented.

17
Issue R: On-site Emergency Preparedness

Safety area: Emergency Preparedness

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R1. Objective

R1.1 The licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) Controlling an emergency situation arising at their site, following any reasonably foreseeable event, including events related to combinations of hazards as well as events involving all nuclear installations and facilities on the site;

(b) Preventing or mitigating the consequences at the scene of any such emergency; and

(c) Co-operating with external emergency response organizations in preventing adverse health effects to workers and the public.

R2. Emergency Preparedness and Response Plan

R2.1 The licensee shall prepare an on-site emergency plan and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating plant activities and co-operating with external response agencies in a timely manner and throughout all phases of an emergency.

R2.2 The licensee shall provide for:

(a) Prompt recognition and classification of emergencies, consistent with the criteria set for alerting the appropriate authorities;

(b) Timely notification and alerting of response personnel;

(c) Ensuring the safety of all persons present on the site, including the protection of the emergency workers;

(d) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(e) Performing assessments of the current and foreseeable situation on the technical and radiological points of view (on and off site);

(f) Monitoring radioactive releases;

(g) Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons on site; and

(h) Plant management and damage control.[67]

[67] Understood as urgent mitigatory repairs, controls, and other actions that are carried out, primarily at the site, while the emergency is still in progress.

R2.3 The site emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall:

- address long-lasting situations;
- clarify how site (and if applicable corporate) resources (human and material) common to several installations are used;
- be co-ordinated with all other involved bodies;

The plan shall be capable of extension, should more severe events occur.

R3. Organization

R3.1 The licensee shall have people on-site at all times with the authority and responsibilities to classify and declare an emergency and, upon classification, to initiate promptly the appropriate on-site response.[68]

R3.2 Sufficient ~~numbers~~ of qualified personnel shall be available at all times for staffing appropriate positions promptly following the declaration and notification of an emergency. Arrangements shall be established to ensure that sufficiently qualified personnel can staff appropriate emergency positions in long-lasting situations.

R3.3 Arrangements shall be made to provide technical assistance to operational staff. Teams for mitigating the consequences of an emergency (e.g. radiation protection, damage control, fire fighting, etc.) shall be available.

R3.4 Arrangements shall be made to alert off-site responsible authorities promptly.

R3.5 The licensee shall identify those who are authorized to carry out the response functions assigned in the emergency plan.

R3.6 The licensee emergency response shall be functional in cases where infrastructures at the site and around the site are severely disrupted.

R3.7 Arrangements to support on-site actions shall be in place with considerations for large-scale destruction of infrastructure in the vicinity of the site due to external hazards.

R4. Facilities and equipment

R4.1 Appropriate emergency facilities shall be designated for responding to events on site and that will provide co-ordination of off-site monitoring and assessment throughout different phases of an emergency response.

R4.2 An "On-site Emergency Control Centre", which is separated from the main control room, shall be provided for on-site emergency management staff. Important information shall be available in the control centre about the plant and radiological conditions on and around the site. The centre shall have means of communicating with the control room, any supplementary control room, other important points on site, and with the on-site and off-site emergency response organizations.[69]

R4.3 Emergency facilities shall be suitably located, designed and protected to

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[68] The on duty shift supervisor could be among those authorised to declare an emergency and to initiate the appropriate on-site response.

[69] The On-site Emergency Control Centre is the office accommodation and associated office services set aside on or near to the site for staff who are brought together to provide technical support the operations staff during an emergency or where the licensee emergency response is directed. It may have plant information systems available, but is not expected to have any plant controls.

- remain operational for accident conditions to be managed (including design extension conditions) from these facilities;
- allow the protection from radiation as well as control of radiation exposure of emergency workers[70].

Appropriate measures shall be taken to protect those occupying emergency facilities for a protracted time from hazards resulting from accident conditions.[71]

R4.4 Instruments, tools, equipment, documentation, and communication systems for use in emergencies (including necessary mobile equipment and consumables such as fuel, lubrication oil etc.), whether located on-site or off-site, shall be stored, maintained, tested and inspected sufficiently frequently so that they will be available and operational during DBA and DEC. Access to these storage locations shall be possible even in case of extensive infrastructure damage.

R5. Training, drills and exercises

R5.1 Arrangements shall be made to identify the knowledge, skills, and abilities needed for personnel (operating organization staff and, if necessary, contractors) to perform their assigned response functions.

R5.2 Arrangements shall be made to inform all employees and all other persons present on the site of the actions to be taken in the event of an emergency.

R5.3 Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that emergency response personnel (operating organization staff and, if necessary, contractors) meet the training obligations.

R5.4 The site emergency plan shall be regularly exercised at least annually. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned. For sites with multiple nuclear installations, some exercises shall address situations affecting multiple facilities on the site. Exercises shall also include the use and connection of mobile equipment, if any.

R5.5 Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the plan shall be subject to review and updating in the light of experience gained.

70 Emergency workers include workers from the operating organisation and, if necessary, contractors, as well as off-site emergency responders that may be needed on-site.

71 This refers, primarily, to ensuring that the On-site Emergency Control Centre and other locations where staff are expected to spend a significant time are located somewhere that the staff can reach and work throughout an extended emergency with minimum risk to health. This will require location away from areas that are likely to be damaged or affected by radiation fields and, where appropriate, this will include provision of re-circulatory air conditioning and continuous radiation monitoring systems.

18
Issue S: Protection against Internal Fires

Safety area: Emergency Preparedness

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S1. Fire safety objectives

S1.1 The licensee shall implement the defence in depth principle to fire protection, providing measures to prevent fires from starting, to detect and extinguish quickly any fires that do start and to prevent the spread of fires and their effects in or to any area that may affect safety.[72]

S2. Basic design principles

S2.1 SSCs important to safety shall be designed and located so as to minimize the frequency and the effects of fire and to maintain capability for shutdown, residual heat removal, confinement of radioactive material and monitoring of plant state during and after a fire event.

S2.2 Buildings that contain SSCs important to safety shall be suitably[73] fire resistant.

S2.3 Buildings that contain equipment that is important to safety shall be subdivided into compartments that segregate such items from fire loads and segregate redundant or diverse trains of a safety system from each other.[74] When a fire compartment approach is not practicable, fire cells shall be used[75], providing a balance between passive and active means, as justified by fire hazard analysis.

S2.4 Buildings that contain radioactive materials that could cause radioactive releases in case of fire shall be designed to minimize such releases.

S2.5 Access and escape routes for fire fighting and operating personnel shall be available.

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[72] In this context, safety refers to all sources of nuclear safety risk, including radioactive waste facilities.

[73] In accordance with the results of the fire hazard analysis.

[74] A fire compartment is a building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total combustion of the fire load can occur without breaching the barriers (barriers comprise doors, walls, floors and ceilings). The fire resistance rating of the barriers must be sufficiently high so that the total combustion of the fire load in the compartment can occur without breaching the barriers.

[75] In the fire cell approach the spread of fire is avoided by substituting the fire resistant barriers primarily with other passive provisions (e.g. distance, thermal insulation, etc.), that take into account all physical and chemical phenomena that can lead to propagation. Provision of active measures (e.g. fire extinguishing systems) may also be needed in order to achieve a satisfactory level of protection. The achievement of a satisfactory level of protection is demonstrated by the results of the fire hazard analysis.

S3. Fire hazard analysis

S3.1 A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire design principles are satisfied, that the fire protection measures are appropriately designed and that any necessary administrative provisions are properly identified.

S3.2 The fire hazard analysis shall be developed on a deterministic basis, covering at least:

- For all normal operating and shutdown states, a single fire and consequential spread, anywhere that there is fixed or transient combustible material;
- Consideration of credible combination of fire and other PIEs likely to occur independently of a fire.

S3.3 The fire hazard analysis shall demonstrate how the possible consequential effects of fire and extinguishing systems operation have been taken into account.

S3.4 The fire hazard analysis shall be complemented by probabilistic fire analysis. In PSA level 1, the fires shall be assessed in order to evaluate the fire protection arrangements and to identify risks caused by fires.

S4. Fire protection systems

S4.1 Each fire compartment or fire cell shall be equipped with fire detection and alarm features, with detailed annunciation for the control room staff of the location of a fire. These features shall be provided with non-interruptible emergency power supplies and appropriate fire resistant supply cables.

S4.2 Fixed or mobile, automated or manual extinguishing systems shall be installed. They shall be designed and located so that their rupture, spurious or inadvertent operation does not significantly impair the capability of SSCs important to safety to carry out their safety functions.

S4.3 The distribution loop for fire hydrants outside building and the internal standpipes shall provide adequate coverage of areas of the plant relevant to safety. The coverage shall be justified by the fire hazard analysis.

S4.4 Ventilation systems shall be arranged such that each fire compartment fully fulfils its segregation purpose in case of fire.

S4.5 Parts of ventilation systems (such as connecting ducts, fan rooms and filters) that are located outside fire compartments shall have the same fire resistance as the compartment or be capable of isolation from it by appropriately rated fire dampers.

S5. Administrative controls and maintenance

S5.1 In order to prevent fires, procedures shall be established to control and minimize the amount of combustible materials and minimize the potential ignition sources that may affect items important to safety. In order to ensure the operability of the fire protection measures, procedures shall be established and implemented. They shall include inspection, maintenance and testing of fire barriers, fire detection and extinguishing systems.

S6. Fire fighting organization

S6.1 The licensee shall implement adequate arrangements for controlling and ensuring fire safety, as identified by the fire hazard analysis.[76]

S6.2 Written emergency procedures that clearly define the responsibility and actions of staff in responding to any fire in the plant shall be established and kept up to date. A fire fighting strategy shall be developed, kept up-to date, and trained for, to cover each area in which a fire might affect items important to safety and protection of radioactive materials.

S6.3 When reliance for manual fire fighting capability is placed on an offsite resource, there shall be proper coordination between the plant personnel and the off site response group, in order to ensure that the latter is familiar with the hazards of the plant.

S6.4 If plant personnel are required to be involved in fire fighting, their organization, minimum staffing level, equipment, fitness requirements, and training shall be documented and their adequacy shall be confirmed by a competent person.

[76] Such arrangements must include nominating persons to be responsible for or have duties with respect to fire protection. The arrangements must set out the requirements for control of all activities that can have impact on fire safety, e.g. maintenance; control of materials; training; tests and drills; modifications to layouts and systems – such as fire detection, fire extinguishing, ventilation, electrical and control systems.

19
Issue T: Natural Hazards

Safety area: Design

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T1. Objective

T1.1 Natural hazards shall be considered an integral part of the safety demonstration of the plant (including spent fuel storage). Threats from natural hazards shall be removed or minimised as far as reasonably practicable for all operational plant states. The safety demonstration in relation to natural hazards shall include assessments of the design basis and design extension conditions[77] with the aim to identify needs and opportunities for improvement.

T2. Identification of natural hazards

T2.1 All natural hazards that might affect the site shall be identified, including any related hazards (e.g. earthquake and tsunami). Justification shall be provided that the compiled list of natural hazards is complete and relevant to the site.

T2.2 Natural hazards shall include:
- Geological hazards;
- Seismotectonic hazards;
- Meteorological hazards;
- Hydrological hazards;
- Biological phenomena;
- Forest fire.

T3. Site specific natural hazard screening and assessment

T3.1 Natural hazards identified as potentially affecting the site can be screened out on the basis of being incapable of posing a physical threat or being extremely unlikely with a high degree of confidence. Care shall be taken not to exclude hazards which in combination with other hazards[78] have the potential to pose a threat to the facility. The screening process shall be based on conservative assumptions. The arguments in support of the screening process shall be justified.

T3.2 For all natural hazards that have not been screened out, hazard assessments shall be performed using deterministic and, as far as practicable, probabilistic methods taking into account the current state of science and technology. This shall take into account all relevant available data, and produce a relationship between the hazards severity (e.g. magnitude and duration) and exceedance frequency, where practicable. The maximum credible hazard severity shall be determined where this is practicable.

[77] Design extension conditions could result from natural events exceeding the design basis events or from events leading to conditions not included in the design basis accidents.

[78] This could include other natural hazards, internal hazards or human induced hazards. Consequential hazards and causally linked hazards shall be considered, as well as random combinations of relatively frequent hazards.

T3.3 The following shall apply to hazard assessments:

- The hazard assessment shall be based on all relevant site and regional data. Particular attention shall be given to extending the data available to include events beyond recorded and historical data.
- Special consideration shall be given to hazards whose severity changes during the expected lifetime of the plant.
- The methods and assumptions used shall be justified. Uncertainties affecting the results of the hazard assessments shall be evaluated.

T4. Definition of the design basis events

T4.1 Design basis events[79] shall be defined based on the site specific hazard assessment.

T4.2 The exceedance frequencies of design basis events shall be low enough to ensure a high degree of protection with respect to natural hazards. A common target value of frequency, not higher than 10^{-4} per annum, shall be used for each design basis event. Where it is not possible to calculate these probabilities with an acceptable degree of certainty, an event shall be chosen and justified to reach an equivalent level of safety. For the specific case of seismic loading, as a minimum, a horizontal peak ground acceleration value of 0.1g (where 'g' is the acceleration due to gravity) shall be applied, even if its exceedance frequency would be below the common target value.

T4.3 The design basis events shall be compared to relevant historical data to verify that historical extreme events are enveloped by the design basis with a sufficient margin.

T4.4 Design basis parameters shall be defined for each design basis event taking due consideration of the results of the hazard assessments. The design basis parameter values shall be developed on a conservative basis.

T5. Protection against design basis events

T5.1 Protection shall be provided for design basis events.[80] A protection concept[81] shall be established to provide a basis for the design of suitable protection measures.

T5.2 The protection concept shall be of sufficient reliability that the fundamental safety functions are conservatively ensured for any direct and credible indirect effects of the design basis event.

T5.3 The protection concept shall:

(a) apply reasonable conservatism providing safety margins in the design;

(b) rely primarily on passive measures as far as reasonable practicable;

(c) ensure that measures to cope with a design basis accident remain effective during and following a design basis event;

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[79] These design basis events are individual natural hazards or combinations of hazards (causally or non-causally linked). The design basis may either be the original design basis of the plant (when it was commissioned) or a reviewed design basis for example following a PSR.

[80] If the hazard levels of RL T4.2 for seismic hazards were not used for the initial design basis of the plant and if it is not reasonably practicable to ensure a level of protection equivalent to a reviewed design basis, methods such as those mentioned in IAEA NS-G-2.13 may be used. This shall quantify the seismic capacity of the plant, according to its actual condition, and demonstrate the plant is protected against the seismic hazard established in RL T4.2.

[81] A protection concept, as meant here, describes the overall strategy followed to cope with natural hazards. It shall encompass the protection against design basis events, events exceeding the design basis and the links into EOPs and SAMGs.

(d) take into account the predictability and development of the event over time;

(e) ensure that procedures and means are available to verify the plant condition during and following design basis events;

(f) consider that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;

(g) ensure that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

(h) not adversely affect the protection against other design basis events (not originating from natural hazards).

T5.4 For design basis events, SSCs identified as part of the protection concept with respect to natural hazards shall be considered as important to safety.

T5.5 Monitoring and alert processes shall be available to support the protection concept. Where appropriate, thresholds (intervention values) shall be defined to facilitate the timely initiation of protection measures. In addition, thresholds shall be identified to allow the execution of pre-planned post-event actions (e.g. inspections).

T5.6 During long-lasting natural events, arrangements for the replacement of personnel and supplies shall be available.

T6. Considerations for events more severe than the design basis events

T6.1 Events that are more severe than the design basis events shall be identified as part of DEC analysis. Their selection shall be justified.[82] Further detailed analysis of an event will not be necessary, if it is shown that its occurrence can be considered with a high degree of confidence to be extremely unlikely.

T6.2 To support identification of events and assessment of their effects, the hazards severity as a function of exceedance frequency or other parameters related to the event shall be developed, when practicable.

T6.3 When assessing the effects of natural hazards included in the DEC analysis, and identifying reasonably practicable improvements related to such events, analysis shall, as far as practicable, include:

(a) demonstration of sufficient margins to avoid “cliff-edge effects” that would result in loss of a fundamental safety function;

(b) identification and assessment of the most resilient means for ensuring the fundamental safety functions;

(c) consideration that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;

(d) demonstration that sufficient resources remain available at multi-unit sites considering the use of common equipment or services;

(e) on-site verification (typically by walk-down methods).

[82] See issue F section 2.

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REPORT
Updating WENRA Reference Levels for existing reactors

September 2014

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Report

Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned

September 2014

WENRA RHWG
Western European Nuclear Regulators Association

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Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned

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EXECUTIVE SUMMARY

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As requested by WENRA, its Reactor Harmonisation Working Group (RHWG) reviewed the 2008 version of the WENRA Reference Levels (RLs) for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned. This review covered the whole set of RLs, taking into consideration recommendations and suggestions published by ENSREG as a result of the complementary safety assessments performed in Europe following the TEPCO Fukushima Dai-ichi accident as well as IAEA safety requirements being under updating for the same reason and the conclusions of the 2nd Extraordinary Meeting of the Contracting Parties to the Convention on Nuclear Safety. WENRA made the updated reference levels available for stakeholder consultation, prior to their finalization.

As a result of the RHWG review and stakeholder input:

- For about half of the issues, there have been either no or only very limited changes.
- The issues where there have been the most significant changes are:
  - issue A (Safety Policy);
  - issue C (Management System) RLs relevant to safety culture have been introduced;
  - issue E (Design Basis Envelope for Existing Reactors);
  - issue F (Design Extension of Existing Reactors) Design extension conditions have in particular been introduced for consistency with IAEA SSR-2/1 safety standard, as well as the need for independent and diverse heat removal means, one being effective for natural hazards exceeding the design basis;
  - issue LM (Emergency Operating Procedures and Severe Accident Management Guidelines);
  - issue N (Contents and Updating of Safety Analysis Report);
  - issue O (Probabilistic Safety Analysis);
  - issue P (Periodic Safety Review);
  - issue R (On-site Emergency Preparedness);
- A new issue (Issue T) dedicated to natural hazards, has been established. This issue has a strong interface with issues E and F.

In many cases, changes have been introduced to explicitly take into account spent fuel storage, sites with multiple reactors, actual conditions at the site resulting from an accident (including those which may be caused by a natural hazard), conditions more severe than the ones considered in the design basis of the plant or the need to ensure relevant equipment or facilities will remain unaffected so that foreseen actions to respond to an accident can be implemented.

RHWG also provides guidance documents on issues E/F and on issue T.

01
Purpose of the report

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The goals of the report are to summarize:

- the process followed by RHWG to review WENRA Reference Levels (RLs) for existing reactors (January 2008 version) in the light of TEPCO Fukushima lessons learned;
- the main comments received during the stakeholder consultation on the draft updated RLs and their disposition;
- the main conclusions of this process.

The proposed set of RLs, updated accordingly, is not included in this report and is available in a separate document at www.wenra.org.

This report is an update of the RHWG report "Updating WENRA Reference Levels for existing reactors in the light of TEPCO Fukushima Dai-ichi accident lessons learned - November 2013"[1].

[1] Report available on WENRA web site: http://www.wenra.org/media/filer_public/2013/11/21/rhwg_report_on_updated_rls_for_existing_npp_-_november_2013.pdf

02
Historical background and context

02.1 Developing WENRA Reference levels for existing reactors (2000-2008)

With the view to increase harmonisation within WENRA countries, as a result of efforts initiated in 2000, WENRA published in 2006 a set of **Reference Levels (RLs)** for reactors in operation at that time in WENRA member countries. The process for their development and the boundary conditions are described in detail in the January 2006 RHWG report[2]. In particular:

- Given WENRA members' responsibilities, RLs should cover nuclear power reactor safety, excluding radiation protection and physical protection[3];
- RLs should not go into legal and technical details;
- RLs should concentrate on safety requirements that are placed by the regulatory regime upon the licensee;
- The safety areas and issues included were selected to cover important aspects of reactor safety where differences in substance between WENRA countries might be expected. They did not seek to cover everything that could have an impact upon safety or to judge the overall level of safety in existing plants;
- As a basis for the RLs, the most recent publicly available edition of the IAEA safety requirements was used. WENRA countries also had an opportunity to propose additional RLs based on national regulations or regulatory guidance.

Following stakeholder comments, the RLs were updated twice in 2007[4] (for example, issues E "Design Basis Envelope for Existing Reactors" and F "Design Extension of Existing Reactors" were largely modified) and, again, in 2008[5] following the publication of IAEA GS-R-3[6] (issue C "Management System" was significantly modified).

WENRA members committed themselves to reach a harmonised situation for existing nuclear power plants by the end of 2010, using the RLs. In January 2011, WENRA published a status report "Progress towards harmonisation of safety for existing reactors in WENRA countries"[7].

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[2] Report available on WENRA website: http://www.wenra.org/media/filer_public/2012/11/05/rhwg_harmonization__report_final.pdf

[3] Despite of this goal, at the end of the RLs development process, some radiological aspects have been included in some issues (e.g. design, periodic safety review and on-site emergency preparedness) because they relate so closely to nuclear safety

[4] Report available on WENRA website: http://www.wenra.org/media/filer_public/2012/11/05/list_of_reference_levels_january_2007.pdf

[5] Report available on WENRA website: http://www.wenra.org/media/filer_public/2012/11/05/list_of_reference_levels_january_2008.pdf

[6] The Management System for Facilities and Activities Safety Requirements, July 2006.

[7] Report available on WENRA website: http://www.wenra.org/media/filer_public/2012/11/05/rhwg_report_harmonisation_existing_npps_feb2011.pdf

02.2 Taking into account the lessons from the TEPCO Fukushima Dai-ichi accident

In March 2011, a major nuclear accident occurred at TEPCO Fukushima Dai-ichi nuclear power plant in Japan.

In Europe, the European Commission and ENSREG initiated a targeted reassessment of the safety margins of nuclear power plants, so called "stress tests". These stress tests were largely based on a specification developed by WENRA. They included a peer review process, performed in the first half of 2012, which resulted in recommendations and suggestions from ENSREG[8]. At a worldwide level, the 2nd Extraordinary Meeting of the Contracting Parties to the Convention on Nuclear Safety (CNS 2012) was held in August 2012 with one objective, to review and discuss lessons learned at that time from the accident at TEPCO's Fukushima Dai-ichi NPP.

In March 2012, WENRA published a Statement "WENRA Conclusions arising from the Consideration of the Lessons from the TEPCO Fukushima Dai-ichi Nuclear Accident"[9], which mentioned "*WENRA emphasizes institutional (roles and responsibilities of governments, regulators and utilities) and cultural (continuous improvement) aspects of nuclear safety in addition to technical issues. WENRA is ready to tackle further issues as necessary on the basis of the lessons learned from the Fukushima accident.* ***WENRA's commitment is to proceed along the path of defining or revising existing Reference Levels as well as developing guidance documents for practical use by regulators***";

To this end, WENRA also announced in this statement the creation of several working groups to address technical issues, including to review and revise (or develop) as necessary existing RLs:

- T1 Natural hazards;
- T2 Containment in Severe Accident;
- T3 Accident Management.

WENRA mentioned that "*The results from the stress tests and conclusions from the CNS 2012 will be incorporated as soon as they become available.*"

In addition, on natural hazards, WENRA also specifically decided to develop guidance: "*WENRA will produce updated harmonised guidance for the identification of natural hazards, their assessment and the corresponding assessment for "cliff-edge" (margins) effects*."

[8] ENSREG puplished a compilation of stress test peer review recommendations and suggestions (available on ENSREG website: http://www.ensreg.eu/NODE/513)

[9] Report available on WENRA website: http://www.wenra.org/archives/wenra-conclusions-arising-consideration-lessons-te/

03
RHWG process to develop updated RLs

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03.1 Purpose of the review and revision of RLs

The main goals of the review and, where necessary, revision of RLs were to:

- have a full review of all RLs (2008 version) but only in relation to Fukushima lessons learned;
- ensure that RLs were still consistent after the update;
- ensure that RLs were still balanced (high level vs detailed level of expectations).

Consistently with the original scope of the RLs, issues related to malicious acts were not addressed. Similarly radiation protection aspects were not addressed except if they were too closely related to nuclear safety. For example in issue R (on-site emergency preparedness), the TEPCO Fukushima Dai-ichi accident highlighted questions on ability to perform necessary emergency management actions and hence these have been included.

03.2 Inputs for the review of RLs

As a result of the WENRA decision, RHWG was tasked to review and, where necessary, revise or develop new RLs to take into account lessons learned from the TEPCO Fukushima Dai-ichi accident. To perform this task, the inputs were:

- work performed by the IAEA to review and revise Safety Requirements in the light of TEPCO Fukushima Dai-ichi accident (mainly the gap analysis performed to support the definition of IAEA DS462[10] and additional inputs to DS456[11] and DS457[12]) as well as conclusions of the 2nd Extraordinary meeting of the Contracting Parties to the Convention on Nuclear Safety;
- compilation of ENSREG recommendations and suggestions[13] as a result of EU Stress Tests;
- initiatives in WENRA member countries to update national requirements and guidance as a result of TEPCO Fukushima Dai-ichi accident lessons learned;
- WENRA work performed on the safety of new reactors, resulting in the RHWG report "Safety of new NPP designs"[14] and the associated WENRA statement[15;]

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[10] IAEA Safety Standards preparation profile - Revision by amendment of GSR Part 1, NS-R-3, SSR-2/1, SSR-2/2 and GSR Part 4

[11] IAEA Safety Standards preparation profile - Leadership and Management for Safety (revision of GS-R-3)

[12] IAEA Safety Standards preparation profile - Preparedness and Response for a Nuclear or Radiological Emergency (revision of GS-R-2)

[13] Document available on ENSREG website: http://www.ensreg.eu/NODE/513

[14] Report available on WENRA website: http://www.wenra.org/media/filer_public/2013/04/30/rhwg_safety_of_new_npp_designs.pdf

[15] Report available on WENRA website: http://www.wenra.org/media/filer_public/2013/04/05/wenra_statement_newdesigns2.pdf

- WENRA position paper on periodic safety review published in April 2013[16];
- work performed by WENRA T1, T2 and T3 working groups;
- comments provided by stakeholders as a result of WENRA consultation on draft RLs (see **03.4**).

03.3 Developing a draft set of updated RLs ready for stakeholder consultation

New or modified RLs were drafted by the various working groups. In addition, some RHWG members also provided drafts. Four RHWG meetings (September 2012, January 2013, May 2013, September 2013) were mostly devoted to the collective review of new or modified RLs.

At RHWG meetings, where working groups leaders where welcomed, discussions took place on:

- adequacy of the proposed modified or new RL, especially with regard to lessons learned from the TEPCO Fukushima Dai-ichi accident;
- whether a RL or guidance may be more appropriate (when a guidance document was under development);
- management of interfacing RLs or issues;
- level of detail and balance of proposed RLs;
- direction for further development at the working group level.

During the RHWG meetings in May and September 2013, re-drafting was also performed to strive towards consensus within RHWG.

These efforts resulted in a version of the RLs that was presented to WENRA at its November 2013 meeting. The RLs were accompanied by the previous version of this report for explanation of the changes. RHWG view was that this version of the RLs was ready for stakeholder consultation. WENRA agreed for such consultation prior to finalization of the RLs.

03.4 Finalizing the set of updated RLs by considering stakeholder comments

The draft updated RLs were put on the WENRA web-page at the end of November 2013 for a commenting period of three months (i.e until the end of February 2014). In addition, key European stakeholders were informed via E-mail on the possibility for commenting.

Altogether 95 comments were received from stakeholders. The main stakeholder who commented was ENISS[17] with altogether 62 comments. Several comments were also received from the UK's Nuclear Institute (15), the European Commission's Joint Research Centre-Petten as support for DG ENER (13) and the German Reactor Safety Commission (4).

The most commented issues were issue F (22 comments), issue T (21), issue E (19) and issue LM (15). In these issues mostly more than one comment was dedicated to a changed RL.

[16] Position available on WENRA website : http://www.wenra.org/media/filer_public/2013/04/05/rhwg_position_psr_2013-03_final_2.pdf

[17] ENISS: European Nuclear Installations Safety Standards Initiative (www.eniss.eu). ENISS provides the nuclear industry with the platform that it needs to exchange information on new national and European regulatory activities, to express its views and provide expert input on all aspects related to harmonization of safety standards.

Sometimes different aspects of the RL were touched by the comments, but mostly the comments had the same intention.

All comments were compiled and, to the greatest extent, grouped according to the RL they were addressing so that all comments addressing a RL could be considered by RHWG before deciding whether the RLs should be modified or not.

When addressing the comments and developing new wording for some RLs, the need for a consistency check of the wording was recognized, to ensure that the same expression was always used with the same meaning or to express the same idea. This check also took account of the definitions in IAEA Safety Glossary (2007) or more recent IAEA Safety Standards (for example IAEA SSR-2/1 published in 2012).

A new version of the RLs as well as a detailed table listing including all comments and suggested disposition was developed prior to the RHWG May 2014 meeting. Most of this meeting was devoted to review this table and develop the final version of the RLs.

As a result of this meeting, late June 2014, RHWG submitted to WENRA, for approval, the final version of the RLs updated in the light of TEPCO Fukushima Dai-ichi accident.

04
Review and revision conclusions, including main changes incorporated in the RLs

04.1 Issues where no or very limited changes are proposed

Although the review was performed on the 18 issues of the 2008 RLs, no or very limited changes were identified in issues B (Operating organisation), D (Training and Authorisation of NPP staff (jobs with safety importance)), G (Safety Classification of Structures, Systems and Components), H (Operational Limits and Conditions), I (Ageing Management), J (System for Investigation of Events and Operational Experience Feedback), K (Maintenance, In-service Inspection and Functional Testing), Q (Plant Modification) and S (Protection against Internal Fires).

With regard to the stakeholder comments received, some improvements of the wording were made in issues D (on training programmes and facilities) and G (on selection of materials and qualification of equipment). Changes were also introduced due to the consistency check of the wording.

04.2 Changes proposed to issue A (Safety Policy)

RHWG proposes to add a new RL on the timely implementation of reasonably practicable safety measures as well as continuous improvement of NPP safety. This new RL widens the scope of application of continuous improvement which was contained within specific issues.

Based on the stakeholder comments, wording of A2.3 was improved and a sentence was added to express the idea that continuous improvement applies to all nuclear safety activities, thus avoiding repeating it in other issues (for example in E11.1 or F5.1). Timely implementation of reasonably practicable safety improvements is also stressed.

04.3 Changes proposed to issue C (Management System)

RHWG proposes to create a new section “7.” on safety culture. Three RLs (C7.1 to C7.3) have been developed to express the need for licensees and their contractors to develop and sustain a strong safety culture with emphasis on the role of the management in supporting and demonstrating safety culture.

04.4 Changes proposed to issue E (Design Basis Envelope for Existing Reactors)

Changes to issue E are mainly clarifications, with the exception of changes introduced to highlight the need to consider spent fuel storage safety (E3.1, E9.8, E9.9, E10.1, E10.6) and to address the interface with the new issue T on natural hazards (E5.2).

RHWG proposes new RLs on attributes to support a sound safety analysis (E8.7) and to take into account site-wide issues when several reactors are collocated (E9.5).

As a guidance document is being developed on issue F (Design Extension of Existing Reactors), the list of transients/accidents to be considered in the design basis, which appears in the 2008 version of the RLs in an appendix to issue E, have been deleted to be included in the guidance.

The changes in issue E due to stakeholder comments were mainly improvements of the wording and changes due to the consistency check of the wording.

With regard to internal hazards (E5.1), RLs were found to be slightly unbalanced as a result of the creation of issue T concerning natural hazards. Recognizing internal hazards would, in principle, require additional RLs but since this could not be managed in detail in this revision, a footnote was added to refer to related IAEA Safety Standards.

Finally, E11.1 (Review of the design basis) was reworded because the principle of "continuous improvement" is now only addressed in A2.3 but is valid for all issues.

04.5 Changes proposed to issue F (Design Extension of Existing Reactors)

The whole issue F was revisited and its structure was changed. Interfaces with issue E (Design Basis Envelope for Existing Reactors) and the new issue T (Natural Hazards) warranted specific attention, as well as the use of the concept of "Design Extension Conditions" (DEC) as established in IAEA SSR-2/1 safety standard (Safety of Nuclear Power Plants: Design – Safety Requirement 2012).

RHWG proposes RLs which:

- address safety of spent fuel storage;
- clearly express whether they are applicable to DEC involving a severe accident (DEC-B) or to DEC not involving a severe accident (DEC-A);
- clarify how DEC to be addressed in safety analysis will be identified (F2.1 to F2.3);
- explicit goals of DEC analysis as well as attributes of the safety analysis of the selected DEC;
- address adequate qualification and operability of (mobile) equipment used to manage DEC;
- address sites where several reactors are collocated;
- require independent and diverse heat removal means, one of them being effective for natural hazards more severe than the one used for design basis (F4.7);
- address availability of I&C, electric power and control room to manage a DEC (F4.7, F4.16 to F4.18).

These general considerations were not changed due to the stakeholder comments. However, some changes were introduced to better express the objective to be achieved (mostly on the purpose of the DEC analysis and on the identification of DEC A and DEC B to be analysed) and a few changes were made for consistency in the wording. The necessity to ensure sufficient margins to "cliff edge effects" is now only expressed in F3.1 as part of the DEC analysis.

When needed, "core damage" was replaced by "fuel damage" to stress that not only the fuel in the core has to be considered but also the fuel in the spent fuel storage.

Similar to E11.1 (Review of the design basis), F5.1 (Review of the design extension conditions) was reworded because the principle of "continuous improvement" is now only addressed in A2.3 but is valid for all issues.

04.6 Changes proposed to issue LM (Emergency Operating Procedures and Severe Accident Management Guidelines)

RHWG proposes RLs which:

- address spent fuel storage safety as well as accidents compromising safety of fuel both in the reactor and in the spent fuel storage;
- address sites with multiple reactors, both considering that all units may be challenged or that one unit may support another;
- prioritise relying on adequately qualified equipment for the implementation of SAMG;
- stress the need to carefully consider potential site conditions to ensure measures envisaged in EOP or SAMG can actually be implemented if needed;
- extend training to all licensee emergency response personnel and expect drills to reflect realistic conditions as far as practicable.

Stakeholder comments resulted mostly in the possibility to use not only EOP and SAMGs but also other specific procedures and guidelines.

A couple of RLs were slightly reworded due to the consistency check of the wording.

04.7 Changes proposed to issue N (Contents and Updating of Safety Analysis Report)

Changes proposed by RHWG are mostly aimed at improving consistency with issue F by referring to DEC, insisting on identification of safety margins (N2.7) and update of information related to site characteristics (N3.1).

In addition, a new RL is proposed (N2.14) to address safety of sites with multiple units.

04.8 Changes proposed to issue O (Probabilistic Safety Analysis)

RHWG proposes changes to O1.1 with respect to how natural hazards are (or not) included in level 1 PSA and to the need to include spent fuel storage in level 1 and 2 PSA.

Furthermore, RHWG proposes to stress the need to consider:

- appropriate mission time for equipment in PSA (O1.4);
- all plant staff and not only control room operators in human reliability analysis (O1.5).

As a result of stakeholder comments, O1.1 was reworded on the operational states and internal and external initiating events to consider in the PSA.

04.9 Changes proposed to issue P (Periodic Safety Review)

RHWG proposes changes to insist on:

- the determination of safety significance of each PSR findings (P3.2);
- the timely implementation of reasonably practicable safety improvements (P1.4);
- the need to identify safety issues which may limit the future safe operation of the plant and measures taken by the licensee to address them (P1.5).

RHWG proposes to clarify the scope of PSR (P2.2) to increase consistency with IAEA SSG-25 safety standard (Periodic Safety Review for Nuclear Power Plants, Safety Guide 2013) and to clarify that interaction between reactors at the same site have to be considered.

Following stakeholder comments highlighting some discrepancies with recent IAEA SSG-25, the listing of areas which have to be considered in the PSR (P2.2) was replaced by the 14 safety factors used in IAEA SSG-25.

04.10 Changes proposed to issue R (On-site Emergency Preparedness)

RHWG proposes changes to improve the consideration of accidents affecting several reactors at the same site (R1.1, R2.3, R5.4), long lasting accidents (R2.3, R3.2) or events where regional infrastructure might be severely disturbed (R3.6, R3.7).

RHWG proposes also to emphasise the need for effective measures for emergency management. This covers the need for:

- adequate emergency facilities, designed to ensure workers radiation safety and enable emergency management (R4.3);
- appropriate procedures and (mobile) equipment to manage the emergency (R4.4);
- sufficient staff (R3.2) who have been appropriately trained (R5.1, R5.3), including through drills and exercises (R5.4). Where contractors are expected to contribute to emergency management, training requirements are applicable (R5.3);
- measures to accommodate long lasting situations (R2.3, R3.2) as well as the situation where site or regional infrastructure would be severely disrupted (R3.6, R.3.7, R4.4).

The potential use of mobile equipment, its associated storage and its use in drills/exercises are also explicitly covered (R4.4, R5.4).

Following stakeholder comments and the consistency check of the wording, some changes were made in the RLs addressing facilities and equipment.

04.11 New issue on natural hazards (issue T) and new associated RLs

RHWG proposes to create a new issue to address natural hazards[18].

After stating the objective of removing or minimizing the threats from natural hazards to the plant, the proposed RLs cover the screening and assessment of natural hazards which might challenge the safety of the reactor. The RLs within this issue address:

- screening of hazards relevant to the site;
- how design basis events shall be identified. A target of 10^{-4}/y for event selection, as well as a 0.1g minimum PGA, are set;
- the need to develop a protection concept to minimize threats to the plant, relying preferably on passive features;
- the consideration of events that may exceed the design basis, to ensure that the design basis chosen is sound and that sufficient margins exist before cliff edge effects may occur.

Most of the changes in issue T due to the stakeholder comments are clarifications.

As for earthquake resistance, T4.2 was discussed extensively in the commenting process because of the 0.1g horizontal peak ground acceleration value and the potential consequences for existing reactors. In a related footnote, it is now stated that “methods such as those mentioned in IAEA NS-G-2.13 may be used” to demonstrate the plant is protected against such an earthquake. More explanation will also be given in the guidance being developed on issue T.

[18] In current RLs, there is already one issue dedicated to a specific internal hazard: Fire (issue S).

05 Conclusion

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In response to the WENRA request, RHWG performed a thorough review of existing RLs in the light of TEPCO Fukushima Dai-ichi lessons learned and proposed a new set of reference levels with an additional issue on natural hazards.

Following WENRA's agreement, the draft revised WENRA RLs were subject to a 3 month consultation period (1 December 2013 to 28 February 2014) to allow stakeholders to comment on the proposals before the RLs were finalized.

All the comments were considered within the review process and a final version of the updated RLs was developed by RHWG to address these comments. This new version - a list of 342 RLs compared to 295 in the 2008 list – has been endorsed by WENRA accompanied by a related WENRA statement.

In addition to the updated RLs, RHWG provides guidance documents on issues E/F and T.

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REPORT

January 2011

Western European

WENRA

Nuclear Regulator's Association

Progress towards harmonisation of safety for existing reactors in WENRA countries

Study by

WENRA Reactor Harmonization Working Group

January 2011

RHWG

Reactor Harmonization Working Group

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I INTRODUCTION

One of the aims of the Western European Nuclear Regulators' Association (WENRA) is to develop a harmonized approach to nuclear safety and regulations[1].

To achieve this objective as far as power reactors are concerned, the Reactor Harmonization Working Group (RHWG) was set up. Following a pilot study, the RHWG performed, from 2003 to 2006, a study on the harmonisation of reactor safety in WENRA countries, addressing nuclear power reactors that were in operation in 2006. Harmonisation matters for new projects were not in the scope of this study, they are now being addressed in a separate study.[2]

The following definition for harmonization was used: "*No substantial differences between countries from the safety point of view in generic, formally issued, national safety requirements, and in their resulting implementation on nuclear power plants*".

The safety areas and issues included in the study were selected to cover important aspects of reactor safety where differences in substance between WENRA countries might be expected. They did not seek to cover all topics that could have an impact upon safety or to judge the overall level of safety in existing plants.

A methodology was developed in five main steps:

1. A set of Reference Levels (RLs) identifying the main relevant requirements on reactor safety was developed for 18 safety issues. These Reference Levels were primarily based on IAEA safety standards;
2. Countries assessed themselves against the Reference Levels on both the legal and implementation side and documented their national position;
3. The national positions were scrutinized in peer review panel sessions to validate the self-assessments;
4. Where judged necessary, changes were made to national assessments and, in some cases, Reference Levels were modified;
5. Areas where harmonization was considered necessary on the implementation and/or legal side in each country were identified.

As part of step 2, national self-assessments against the set of RLs were performed. The purpose of these self-assessments was, for each RL, to answer two questions:

i. Is there an equivalent national requirement[3] that meets the substance of the RL?

ii. Have all operating nuclear power plants in the country implemented the RL?

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[1] WENRA terms of reference (26 March 2010)

[2] Safety Objectives for New Power Reactors – Study by WENRA RHWG – December 2009

[3] A national requirement is to be understood as a documented statement in an official, open document/publication that is part of the legal regulatory system and has been formally issued. These requirements are of two types, both of which provide a basis for regulators to exercise their powers and duties, but at different levels:

- a legally binding requirement, such as a law, ordinance or regulation that is mandated and enforced, if necessary with the use of legal sanctions. These requirements are issued by the parliament, government, or regulatory body as authorized; and
- a general recommendation (rule, condition, guideline, principle, standard, etc) that the regulatory body issues formally with reference to a legally binding document, decision, permission, or other formal authorization. These are not legally binding and enforced like regulations; however, they are used for granting licences and regulating licensees' activities.

There were three possible coded results for each question[4] :

A: Yes – already harmonized in substance;

B: No – a difference exists, but can be justified from a safety point of view; or

C: No – a difference exists, and should be addressed for harmonization.

Reference levels codified as 'C' form the basis of national action plans.

The methodology and results of the study are presented in a report published by WENRA in January 2006 "Harmonization of Reactor Safety in WENRA Countries"[5]. The results of the study were also presented during a seminar in Brussels in February 2006. It appeared that the majority of the RLs are implemented in nuclear power plants in WENRA countries, but that a significant amount of work had to be done to align the national requirements with the Reference Levels.

Stakeholders were invited by WENRA to provide comments on this report[6]. The comments received affected each of the 18 safety issues. As a result, the RLs were updated in March 2007. The RLs were again updated in January 2008[7], mainly to take into account the publication of the IAEA document GS-R-3[8]. This constitutes today the latest revision of the RLs.

On the basis of the above described self-assessment validated by the peer review panel sessions, each country set up a national action plan to reach a harmonized situation by 2010, which was made public. WENRA monitored progress with national action plans on an annual basis.

II OBJECTIVE AND CONTENT

WENRA members committed themselves to reach a harmonised situation for existing nuclear power plants by the year of 2010, using as a minimum the RLs. The aim of this document is to report progress on this way and on the further steps envisaged to fully achieve this commitment.

Sections III.1 and III.2 present an overview of the situation in WENRA countries regarding harmonization of safety of existing reactors as of September 2010 and a perspective for the near future on both the regulatory side – questions (i) – and the implementation side – questions (ii). However, it was not possible to compare directly the current situation with the situation in 2006. The reason for this is that the RLs have undergone substantial modification in 2007 and 2008. For instance, issues E and F have been fully reconsidered in 2007 following stakeholders' comments and issue C has been largely modified in 2008 to take into account a new IAEA publication (GS-R-3).

4 It should be noted that an 'A' assessment could be achieved for implementation on NPPs, even when there were no formally issued, public, generic, national requirements.

5 Report is available on WENRA web site at the following link : http://www.wenra.org/dynamaster/file_archive/060116/b8c660648ecc1fd66a0280b7d0ccd05b/RHWG%20Harmonization%20%20Report%20Final.pdf

6 As a result of stakeholder consultation, WENRA received 177 specific comments on the reference levels and 65 comments of a more general nature. Most of the technical comments were received from the European Nuclear Installations Safety Standards Initiative (ENISS), a consortium of European nuclear utilities.

7 Current reference levels for existing reactor are available on WENRA web site at the following link : http://www.wenra.org/dynamaster/file_archive/080121/1c826cfa42946d3a01f5ee027825eed6/List_of_reference_levels_January_2008.pdf

8 The management system for facilities and activities, July 2006.

Section III.3 presents the lessons learned from the associated national regulatory efforts.

The current situation of each country is described in appendixes:
- appendix 1 summarizes the status of each country towards harmonisation;
- appendix 2 deals with update of the national requirements to take into account the RLs;
- appendix 3 deals with the implementation of the RLs on the nuclear power plants.

III EUROPEAN OVERVIEW

III.1 European overview: regulatory side

In 2006, across WENRA countries, approximately only half of the RLs were formally required in the various national requirements ('A' code). The conclusion of the review was that there was a need for a significant number of additional, formally issued, generic national requirements or recommendations.

Since 2006, each country has made a great effort to develop or revise national requirements in order to fill the gap on the regulatory side. In each country, the regulatory body has set up an action plan with specific projects to incorporate the RLs within the national regulatory framework. As this was not a "translation" process but more a "transposition" process, this required large efforts. In many countries, this transposition process lead to revisit existing regulations or to create a new set of regulations. Some countries also extended the applicability of relevant RLs to other nuclear installations.

All countries have reported significant progress in their action plans and although not completed in some cases, the aforementioned gap is now smaller. New or updated regulations as well as guidance documents have been published or draft documents are under review by stakeholders. In some countries however, it could take a few more years to finalise harmonisation on the regulatory side.

The development of national requirements is indeed a complicated process, involving stakeholders' consultation. According to national legal framework and practices, the final approval of the requirements may depend not only on the regulator but also on the national government or parliament. This introduces uncertainties on when the new requirements will come into force.

III.2 European overview: implementation side

In 2006, it was concluded that the RLs were implemented to a large extent in all WENRA countries, even in the absence, in some cases, of corresponding legal requirements or formally issued recommendations. The reason for this was that, in many countries, the licensees would respond to expectations from the regulatory body even if they were not in legally binding documents or formally issued recommendations, or act on their own under their prime responsibility for safety.

In a majority of countries, the licensees were requested to perform self-assessments against the RLs. In several countries, the regulatory bodies also verify compliance with the RLs as part of their inspection and control process, sometimes through dedicated inspections.

The implementation of the reference levels has made progress in all countries since 2006. However, it will take some additional years for full implementation. For the few RLs not yet implemented:

- if these RLs are incorporated in legally binding documents, their implementation will be required by the end of the transitory period allowed by these documents (usually one to three years or less);
- if these RLs are incorporated in formally issued public recommendations, the licensees will have to justify either that they comply or that they have implemented equivalent safety provisions. In many cases, this is checked through the periodic safety review process.

III.3 Lessons learned

This harmonisation project included establishment of common Reference Levels, benchmarking of national positions, setting up national action plans, and where necessary revising national requirements and implementing the Reference Levels on nuclear power plants. It has been a unique international voluntary effort and is a step towards harmonisation of nuclear safety in Europe. The study is also believed to be the most extensive joint international use of the IAEA safety standards.

This project has been possible due to:

- the commitment to harmonisation of each WENRA member;
- the human and technical involvement of each national regulatory body;
- the framework based on voluntary cooperation;
- the atmosphere of openness and mutual trust throughout the project.

The project has required much more resources and working time than foreseen at its beginning. Reasons for this are that the benchmarking process was a pioneering effort and that the revision of the regulations was a huge effort given the number of documents to develop or update.

Beyond the documents and reports produced as part of this project, RHWG meetings have enabled a greater understanding of the various national regulations and practices. They also were opportunities to discuss key safety issues. Additional benefits from the participants' point of view have been the building of a strong informal network.

The seminar held in 2006 at the institutional level has made possible a transparent dialogue with stakeholders, not least the European utilities. This dialogue has further improved the quality of the project. The utilities were receptive to the project and this encouraged them to increase the cooperation among them on the technical topics covered by the project. On a national basis, licensees sometimes anticipated future national requirements for example by modifying plant or operating practices.

Some reference levels have generated in-depth discussions about their interpretation, both at an international level during stakeholders' consultation and at a national level between regulators and licensees when drafting requirements or discussing implementation. Some clarifications have already been given and additional work on this topic is ongoing inside WENRA.

The working methodology which has been developed for this project has proven to be fit for purpose and could be used in other domains.

IV CONCLUSION

Considerable progress has been made since 2006 towards the objective of harmonisation of reactor safety for existing nuclear power plants in WENRA countries. Some work is still going on, with clear steps to complete the action plans for harmonisation. Ensuring completion of national action plan is the responsibility of each national regulator. Each WENRA country will report publicly on it.

The project has already resulted in convergence of national requirements and in safety improvements on some nuclear power plants in WENRA countries.

There has already been discussion on the common understanding of some reference levels and it is now envisaged to further discuss the implementation of some reference level to ensure consistency between WENRA countries in the long term. It is also envisaged to revise the reference levels when necessary to keep them up to date with the state of the art in nuclear safety.

Appendix 1

Summary of national status towards harmonisation

Country	Summary of national status
Belgium	The Federal Agency for Nuclear Control (the FANC), constitutes the Belgian Safety Authority. The FANC ensures the overall supervision of all civil nuclear activities in Belgium. The Regulatory body is constituted by the FANC and by Bel V, the technical subsidiary body of the FANC. In order to include the Reference Levels into the Belgian regulation which was mainly addressing radioprotection issues, a regulatory project with high priority started from 2007 at the FANC. This project has been conducted in close collaboration with Bel V. It is worth to mention that this regulatory proposal will also transpose articles 6 and 7 (partially) of the European Directive on Nuclear Safety (2009/71/EURATOM) into the Belgian Regulation. It is expected that the final regulatory text could be submitted to the Government for approval early 2011 and will be published in the official journal mid 2011. On the practical implementation side, a Consultative Committee composed of high level staff from the Regulatory body and from the operator has been set up for the formal follow-up of the action plan proposed by the operator to address the 35 RLs having been scored a "C" in the benchmarking exercise in 2006 . This committee performs a systematic review of the ongoing actions for each NPP. Formal closure of actions is proposed by the operator to the Regulatory body. The Regulatory Body, taking also into account the follow up performed by Bel V on the NPPs sites, approves the closure of the actions and acts this closure in the meeting report. At present (end 2010) 15 actions have been declared closed. A few actions that require the highest manpower effort (on PSA and Fire protection) are expected to be completed end 2015.
Bulgaria	In Bulgaria the process of harmonization with the reference levels for reactor safety involves development and enforcement of new Nuclear Act, new regulations and new regulatory guides. The Nuclear Act, the regulations as well as most of the regulatory guides included in the National Action Plan for Harmonization have been adopted and published by the end of 2010. The remaining two regulatory guides will be formally published in the beginning of 2011.

Country	Summary of national status
Czech Republic	In the 2006 harmonisation efforts started to harmonise Czech national nuclear legislation with the WENRA RLs. Results of the 2006 RHWG report identified that in comparison with Czech legislation there was certain level of inconsistency in 172 RLs (from 288). Based on the results of the RHWG report and in accordance with the agreements in WENRA the SUJB prepared and approved an Action plan in order to harmonize the legislation till 2010. The Action plan included amendment of Decree No. 195/1999 Coll., "on Basic Design Criteria for Nuclear Facilities with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness". Selected parts of WENRA reference levels were planned to be harmonized till the year 2010 by issuing new or updated Regulatory Safety Guides. This is possible because general harmonisation plan includes important change in hierarchy of Czech legislative documents by lifting up the level of Regulatory Safety Guides by publishing them in official Journal. The amendment of Decree No. 195/1999 Coll., "on Basic Design Criteria for Nuclear Facilities with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness" was prepared and submitted to stakeholder consultation. This consultation is now close to its end. Five new SUJB Safety Guides have been published, five SUJB Safety Guides are under consultation/decision making process, and six SUJB Safety Guides are drafted or under revision for updating.
Finland	The Finnish nuclear energy legislation was revised in 2008. All nuclear safety regulatory guides, YVL Guides, are at the moment under revision. STUK has set an internal time schedule for this revision effort in such a way that all guides will be prepared at least to the level of a final draft before the end of 2010 and, that all new guides will be published before the end of 2011. This overall reform of the YVL Guides is progressing essentially according to the schedule. All WENRA reference levels will be included in the new YVL Guides practically as such. With these measures Finland fulfills the WENRA commitment with regard to harmonisation of safety regulations. On the implementation side, all WENRA reference levels are currently implemented at the Finnish nuclear power plants.

Country	Summary of national status
France	The action plan established by ASN at the end of 2005 to incorporate the remaining WENRA RLs into French regulatory framework was significantly modified following the publication of a new act (the act of 13 June 2006 on transparency and security in the nuclear field, referred to as the "TSN act") and one of its associated implementation decree (Decree No. 2007-1557 dated November 2, 2007) as well as the ASN change of status (ASN became an independent administrative authority). In addition to the already published TSN Act and November 2007 decree, the current roadmap to transpose WENRA RLs consists in ASN publishing or having published: ▪ one ministerial order stating the overarching provisions for nuclear installations; ▪ 10 ASN decisions and one ASN guide. At the end of 2010: ▪ all but one (dealing with pressurized equipments and updating an existing ministerial order) drafts have been written. However, two are still under ASN in-house review process; ▪ the draft ministerial order, six draft ASN decisions and one draft ASN guide have been submitted to stakeholder comments (and are available on ASN's web site). New versions are being prepared taking into account comments received; ▪ one ASN decision is going to be sent out for stakeholder comments in the first weeks of 2011 and a second one by the end of the first half of 2011. As a result, ASN is expecting to have all these regulations published in 2011 once the ministerial order has been signed by the Government. At the beginning of 2006, although few formal regulations were set to govern the design and operation of French nuclear power plants, most of the RLs were actually implemented. Among the roughly 300 RLs, about 15 needed improvements on their implementation. Mid 2007, EDF completed a self-assessment on the implementation of WENRA RLs. Mid-2010, at ASN's request, EDF completed a second self-assessment. Overall, compared to 2006, implementation of WENRA RLs for existing reactors has improved in France but is still not yet fully completed. The full implementation of all RLs is linked to the updating of the regulations and the transitory measures they will provide.

Country	Summary of national status
Germany	The applicable national higher-level nuclear rules and regulations in Germany date back to the 1970s and 1980s. Since 2006 several actions were performed to overcome the deficiencies in the 2006 national action plan with 90 “C” assessments and to implement the WENRA RL into national regulation. These actions include an updating of existing ordinances, the development of a new ordinance and mainly the development of new, modular regulations. The WENRA reference level have been systematically considered in that process. With the new Safety Criteria for Nuclear Power Plants, the draft of a new higher-level nuclear rules and regulations is now available. The Federal Ministry for the Environment published the draft of the new “Safety Criteria for Nuclear Power Plants - Revision D, of April 2009” on the Internet. The Federal Ministry for the Environment and the Länder of Baden-Wuerttemberg, Bavaria, Hesse, Lower Saxony and Schleswig-Holstein have agreed upon a comprehensive consultation procedure as a test phase for the new Safety Criteria. On this basis of practical experience gained from testing, the Federation and the Länder jointly review the rules and regulations by mid-2011. The Federation and the Länder are striving for a unanimous adoption of the nuclear rules and regulations. Publication by the Federal Ministry for the Environment in the Federal Gazette will not take place before the end of the procedure. Regarding the implementation of the reference levels 28 “C” assessments had to be considered. Nearly all, except of two reference level (SAMG and PSA for plant modification) have been implemented in the meanwhile in the German NPP.
Hungary	In accordance with the Atomic Act the Regulations shall be reviewed and updated as needed at least once in every five years on the basis of scientific achievements as well as domestic and international experience in Hungary. The latest review aimed – among others – at improving and harmonizing the Regulation based on the Reference levels. The technical review was finished at the end of 2009. Now, the draft of the new set of Regulations is under administrative collation. After the issuance of the new Governmental Decree on the Nuclear Safety Requirements of Nuclear Facilities and Related Regulatory Activities (expected in 2011) Hungary fulfils the WENRA commitment on harmonization of nuclear safety. On the implementation side, in accordance with the national action plan several tasks have been completed. Due to these activities 13 Reference Levels out of 37 have been implemented. For all other Reference Levels (24), the actions are on-going in line with the action plan. Typically they are related to the LM (Emergency Operating Procedures and Severe Accident Management Guidelines) and O (Probabilistic Safety Analysis) issues. The subtasks of action plan related to the Severe Accident Management Guidelines (SAMG) will be performed unit by unit. In the case of Unit 1 the activities will be completed by the end of 2011. All SAMG activities will be finished by the end of 2014. The actions related to the Probabilistic Safety Analysis will be also finalized by the end of 2014.

Country	Summary of national status
Italy	The action plan that was defined in 2006 took strictly into account the fact that Italian NPPs (Garigliano - BWR, Latina-Magnox, Trino-PWR and Caorso-BWR) were no longer in operation since many years ago. At present they still are, at different stages, in the process of being decommissioned. The main activities that are conducted on the sites are related to waste management (conditioning and site storage), fuel removal - for the NPPs still having fuel in the pools (Trino and Caorso) - as well as dismantling. Taking this specific situation into account, the national action plan for harmonization to the WENRA Reactor Safety Reference Levels was prepared considering only those reference levels that were relevant for the above activities, with the additional intent to wait in order to coordinate such actions with those required in the Action Plan for the harmonization to the waste storage and decommissioning reference levels, which unfortunately are still not in the final issue. In this light, attention was therefore basically addressed to management, organizational, quality and fire protection issues. Other issues are considered not to be a national priority for the time being. They will be reconsidered when concrete development steps toward the construction of new plants will be performed.
Lithuania	In Lithuania the process of harmonization with the reference levels for reactor safety involves development and enforcement of new regulations and updating of existing ones. As a result 73 (out of 120) RL's were incorporated into national requirements, 24 RL's are included in to the three new draft regulations, which are still in the stage of approval. Above regulations will be applied to the new build rather than for existing Ignalina NPP. By the end of September 2010 Ignalina NPP is in permanent shutdown stage. During the last years of operation 52 out of foreseen 75 RLs levels were implemented. Due to shutdown of Ignalina NPP the remaining RLs are agreed not to be implemented. It is planned to take into account WENRA RLs in to regulations devoted to new build as it is relevant.
The Netherlands	The present system of regulations is based on IAEA safety standards from the '80 and '90. Since the existing power plant has got the opportunity to be operated until the end of 2033, a revision has been drafted of nuclear safety regulation. It covers the following areas: design, operation and quality assurance. The revision is based on the latest versions of the IAEA safety standards including requirement documents and safety guides. The documents will be called again Nuclear Safety Rules ('Nucleaire Veilgheidsregels', NVR's) and contain adaptations to cover all WENRA RL's and national needs. The documents have been discussed over the last year with the stakeholders. The documents are ready for submission to licensee, which can be expected early 2011. The complete list of new NVR titles can be found in the Dutch report to the 5th Convention on Nuclear Safety. The licensee has implemented on a voluntarily basis the latest IAEA safety standards ahead of national regulations. For that reason the implementation of the WENRA RL's was not a great task and was completed in 2009.

Country	Summary of national status
Romania	The WENRA Reactor Safety Reference Levels have been incorporated into the Romanian regulatory framework through the following regulations: - Requirements on Fire Protection in Nuclear Power Plants (2006) – incorporating RLs in Issue S; - Requirements on Periodic Safety Review for nuclear power plants (2006) – incorporating RLs in Issue P; - Requirements on Probabilistic Safety Assessment for nuclear power plants (2006) – incorporating RLs in Issue O; - Nuclear Safety Requirements on the Design of Nuclear Power Plants (2010) – incorporating RLs in Issues E, F, G & N; The revision of the set of 13 regulations on quality management systems, covering activities related to all the phases of the lifetime of nuclear installations, started in 2007, takes account of the latest IAEA Requirements and Guides on Management Systems (GS-R-3, GS-G-3.1 and GS-G-3.5). The external consultation process for the new regulations has been finalised and they are due to be published in the first half of 2011. These new regulations cover all reference levels in Issue C; A regulation on commissioning and operation of NPPs is currently under drafting and will incorporate the remaining RLs. The intention is to have it published before the end of 2011. The compliance with the requirements in the reference levels is currently re-assessed by the licensee as part of the periodic safety review that is ongoing. This assessment has been required by CNCAN. The assessment of the licensee's implementation of the RLs has been assessed by CNCAN in support of the benchmarking performed within RHWG in 2005 and also as part of the assessment of the implementation of the regulations issued.

Country	Summary of national status
Slovakia	Based on the national benchmarking there were 85 RLs to be harmonized in total. National action plan for WENRA RLs implementation counted on a "one-step" approach, i.e. all RLs to be harmonized would be implemented into various levels of national legal documents (atomic act, decrees and safety guides) at once and the set of new revisions would be sent for official approval process as a batch. This "one-step" approach had been followed since 2007, when intensive works on the revision of the atomic act and decrees were launched. In May 2010 the final draft of these documents were finalized and sent for comments to other state ministries and authorities according to Slovak national legal procedure. Part of the RLs was incorporated directly in the Atomic act and the rest of these were incorporated into the seven existing decrees. The new revision of the Atomic Act has been sent for official legal approval process within the country in August 2010. The new revisions of the seven regulations are finalised and are expected to be sent for official legal process in the end of September 2010. The official approval process of both atomic act as well as of the set of regulations is expected to be finished by the end of 2011. By amendment of the Atomic Act and the above-mentioned list of regulations all of the RLs will be incorporated into the Slovak legislation.
Slovenia	The renovation of the national legal system after adopting the "2002 Act" went on by issuing the number of regulations. These regulations cover all WENRA Reference Levels, except Reference level D – Training. The new, updated version of Regulation JV4 are now going through the process of adoption. Practically all WENRA requirements are included in the domestic safety regulation which is on power and in use. The 5 requirements from the domain of Training (WENRA Ref. Level D) will be incorporated in the appropriate regulation during the year 2011. During the period 2006-20010 improvements have been achieved in WENRA requirements implementation in Krško NPP. Open issues allocated during the benchmark exercise have been implemented almost completely. Implementation is not confirmed yet for issues related to plant staff, (sufficiency and changes assessment, long term planning, management and supervision of contractors work), quality management system (QMS) (role in organizational changes and way of implementation of QMS), SAR update with relevant decommissioning data and PSA use to assess significance of operational occurrences. Entire implementation of these WENRA reference levels will be confirmed during the year 2011.

Country	Summary of national status
Spain	After the identification of the specific needs for national harmonization in 2006, the Spanish Nuclear Safety Council (CSN) has strengthened with the maximum priority the development of technical standards, regulations, safety guides and instructions, in accordance with the Action Plan for the harmonization in WENRA. As part of the regulatory efforts in the field of nuclear safety, within the framework of the mentioned action plan, the CSN has issued ten CSN Safety Instructions, three more will be published soon after the consultation/decision making process, two additional instructions are in drafting phase, and other one is being reviewed; CSN Safety Guides have been also reviewed or issued. These works, which are scheduled to finish at the beginning of the year 2011, fulfill the commitments of Spain with regard to harmonization of nuclear safety for existing reactors in Europe.
Sweden	Sweden is not fully able to satisfy the WENRA agreement to align the national safety requirements with all the reference levels by the end of 2010. This has mostly to do with circumstances outside the control of the Swedish Radiation Safety Authority, such as change of plans caused by the merger of SKI and SSI, and phasing in to legal changes. However, it can also be concluded that rather few gaps remain to be handled. After the final benchmarks in 2007, 46 gaps remained on the legal side. 34 of these will be closed through an ongoing revision of the safety regulations for nuclear installations (SSMFS 2008:1). This revision is planned to be finalised in 2011. The remaining 12 gaps on the legal side (regarding issues E, K and S) will be dealt with in a foreseen revision 2012 of SSM´s regulations on design and construction of power reactors (SSMFS 2008:17). This revision is depending on consultations with other Swedish authorities and some ongoing technical investigations of the bases for making some other changes to these regulations. In addition, SSM will have to decide whether to update SSMFS 2008:17 to apply also on new reactors, after a recent change of the Act on Nuclear Activities effective from 1 January 2011. On the implementation side eleven gaps remain, related mostly to analysis of certain events and conditions. These are all addressed in the ongoing modernization programs of the NPPs. The final measures were planned to be completed 2013. There is a possibility that single measures in the design extension envelope (issue F) will be completed for all reactors 2015.

Country	Summary of national status
Switzerland	In Switzerland, the legislation for the use of nuclear energy and on radiological protection is enacted exclusively at the federal (national) level. The main provisions for authorisations and regulation, supervision and inspections are established in the Nuclear Energy Act and the Radiological Protection Act. The legal rules and principles are put in concrete terms in the Nuclear Energy Ordinance, Radiological Protection Ordinance and in about 10 further ordinances. The main basis for implementation and enforcement are the Guidelines of the Swiss Federal Nuclear Safety Inspectorate (ENSI). In 2006, many WENRA reference levels were not covered by the Swiss regulation. At that time, the enactment of the new Nuclear Energy Act (2005) called for a "rewriting" of all ordinances and guidelines. This was a good opportunity to implement the WENRA reference levels. Since 2005, 4 new or fundamentally revised ordinances applicable to NPPs have been enacted. Concerning the level of Guidelines the output of new regulations was even more extensive: 2 new ENSI-guidelines were published in 2007, 6 in 2008, 7 in 2009 and 5 in 2010. The process is still underway. Currently, about 80% of the reference levels are covered by the Swiss regulation. Main gaps are in issues E, H and N. Complete harmonization is expected by the end of 2011 (3 new guidelines). Concerning implementation, almost all open points were resolved. Currently, 1 reference level (issue O) is still not implemented. The implementation process will be finished in 2011.
United Kingdom	In the UK the WENRA reactor safety reference levels (RLs) are considered to be fully incorporated into national requirements. Our technical assessment guides have been revised as necessary to incorporate information from the RLs as well as formally adopt the RLs as relevant good practice (meeting relevant good practice is required by law in the UK). The day-to-day use of these guides, together with sample inspections by the regulator and self assessment activities by the duty holders, give adequate confidence that the RLs are implemented, so far as is reasonably practicable, on operating nuclear power plants. Hence for operating nuclear power plants the UK has achieved WENRA's commitment "by the year 2010 to improve and harmonise our nuclear regulatory systems, using as a minimum the reference levels".

Appendix 2

Update of the national requirments to take into account the RLs

Country	BELGIUM
General presentation of the regulatory system	
The Federal Agency for Nuclear Control (the FANC), created by the law of 15 April 1994 constitutes the Safety Authority. The FANC ensures the overall supervision of all civil nuclear activities in Belgium. The FANC reviews license applications and submits decisions to the King for granting licenses for the high risk facilities, or grant licences himself for low-risk facilities. The regulatory body is constituted by the FANC and Bel V. Bel V was created in September 2007, as a subsidiary body of the FANC. According to the law of 22 December 2008, Bel V is given a mandate to perform regulatory missions that are legally delegated by the FANC. These missions include amongst others the systematic and periodic on-site inspections and the technical review of safety analysis performed by the Licensee in the frame of licence application or of modifications to the installations. A nuclear safety control structure with 3 levels is in place : first by the licensee's Health Physics Department (HPD), then by Bel V which performs by delegation of the FANC a number of inspections and regulatory tasks, and finally by the Safety Authority (FANC). The FANC in also charge of making proposals for updating the general regulations, transposing the relevant European directives, international treaties, etc. and of maintaining the internal coherence of the general regulations.	
Situation of the national regulations in 2006 with respect to the RLs	
Only a few issues were formally implemented into the Belgian Regulation. The existing nuclear regulation, is mainly targeted to radioprotections issues. The Safety of nuclear installations was treated in the Safety report of the nuclear installation. The Safety report is a legally binding document for the Licensee, in the sense that the operating License (granted by the King) contains an obligation to conform to the provisions of the Safety report. Considering that the Safety report was not a general regulation adopted according the Belgian legislative mechanisms nor was a public document, the Safety report has not been considered as a legal text, and only a minority of the WENRA issues was considered as legally implemented in the Belgian regulation.	
Actions started to incorporate RLs in national regulations	
A FANC regulatory project with high priority started in 2007 and continued in 2008. This project has been conducted in close collaboration with Bel V. As no other existing regulation dealt with Nuclear Safety, the WENRA issues were drafted in a regulation proposal with the same structure as the Reference Levels, which appeared on the base of analysis as being of a rather universal structure. . End 2008, a first part of the text was ready and submitted for comments to the licensee. The second part of the text has been submitted mid 2009 to the licensee. The comments of the licensee were reviewed by the FANC and Bel V from mid 2009 and the text was amended when considered appropriate. Beginning 2010, the regulatory proposal was submitted to the Scientific Council, which is an independent advisory committee to the FANC. As no similar regulation was available, the opportunity was taken to select some reference levels to be applicable to other facilities (mainly fuel cycle facilities and research reactors) and activities as	

well. The regulatory proposal was finally structured in this sense, i.e. in two parts: A first part applicable to all fuel cycle facilities (including NPPs and final waste repositories) and a second part applicable only to the nuclear power plants. It is worth to mention that this regulatory proposal will also transpose articles 6 and 7 of the new European Directive on Nuclear Safety (2009/71/EURATOM) into the Belgian Regulation.
Status of the national regulations in September 2010 and envisaged further actions
In July 2010, the regulatory proposal has been submitted for comments to the concerned official Belgian advisory bodies (like Health Council, Ministry of Labour,...), to the European Commission in the frame of Art. 33 EURATOM and to the concerned operators. Comments are awaited for end November, and it is expected that the final text could be submitted to the Government for approval by the end of 2010 or early 2011 and will be published in the official journal mid 2011. Envisaged further actions : Similar projects are on track in order to transpose the Reference levels developed by the WENRA Waste and Decommissioning Working Group. These projects will complete the proposed regulation in Nuclear Safety developed on the basis of the RHWG reference levels. Other regulatory projects related to final disposal of radioactive waste and, in the more far future, to other specific nuclear installations will complete this regulation.

Country	BULGARIA

General presentation of the regulatory system

In the Republic of Bulgaria, the Parliament has the authority to adopt legislative acts, while the Government adopts the secondary legislation for implementation of the laws. The rules and regulations are promulgated by a governmental decree. Each governmental authority issues instructions or guidance to provide directions concerning the implementation of the legislation.

A process of revision and update of the national nuclear legislation took place in the past years, which resulted in adoption of a new **Act on the Safe Use of Nuclear Energy** (ASUNE) in 2002 and renewal of the secondary legislation on its application in 2004. Act on Amendment and Supplement to the Act on Safe Use of Nuclear Energy was promulgated on 12 October 2010.

The ASUNE is the basic legislative act in the use of nuclear energy. It stipulates the state regulation of the safe use of nuclear energy and ionizing radiation, and the safety of radioactive waste and spent fuel management. The responsibilities of the licensees for ensuring nuclear safety and radiation protection are specified there as well. According to the Act, the Nuclear Regulatory Agency (NRA) is the regulatory body for nuclear safety in Bulgaria. The NRA Chairman is an independent specialized authority of the executive power and is vested with competencies for state regulation of the safe use of nuclear energy and ionizing radiation, and the safety of radioactive waste management and spent fuel management.

The secondary legislation comprises 19 regulations on the application of the ASUNE requirements regarding the safety of nuclear power plants (NPPs) and the sources of ionizing radiation.

By the virtue of the regulations, the NRA Chairman is authorized to issue Regulatory Guides with reference to the legally binding requirements. To achieve a comprehensive regulatory framework, a Plan for Development of Guides on Implementation of the ASUNE Regulations had been established in 2005.

Situation of the national regulations in 2006 with respect to the RLs

Taking advantage of the process of renovation of the nuclear legislation, a significant part of the RLs (about 240) had been incorporated in the new regulations, specifically in the following ones:

- Regulation for providing the safety of nuclear power plants;
- Regulation for the procedure for issuing licenses and permits for safe use of nuclear energy;
- Regulation of the conditions and procedure for notification of the NRA about events in nuclear facilities and sites with sources of ionizing radiation;
- Regulation for emergency planning and emergency preparedness in case of nuclear and radiation accident;
- Regulation of the conditions and procedure for acquiring professional qualification and for the procedure for issuing licenses for specialized training and certificates for qualification for use of nuclear energy.

Actions started to incorporate RLs in national regulations

The RLs, which imply more detailed requirements or guidance, were planned to be considered in Regulatory Guides (RGs). The national Action plan composed in 2006 as a result of the benchmarking activity with the reactor harmonization RLs reflects these measures. Even though the system of RGs had undergone some changes since 2006, the differences of type “C” identified during the benchmarking process have been addressed in the developed guides.

Status of the national regulations in September 2010 and envisaged further actions
The following RGs have been developed to cover the identified differences: - RG on deterministic safety analysis - RG on the use of PSA in support of the plant safety management - RG on protection against internal fires - RG on NPP operation - RG on management system for facilities and activities The first three of the listed guides have been already formally published, while the last two guides are expected to be published by the end of 2010.

Country	CZECH REPUBLIC
General presentation of the regulatory system	

Existing Czech legal framework for regulation of activities related to peaceful utilization of nuclear energy is implemented since 1997. The basis for this legislation was acts and regulations of former Czechoslovak Commission for Nuclear Energy developed since 1974.

The Act No. 18/1997 Coll., on Peaceful Utilization of Nuclear Energy and Ionizing Radiation (Atomic Act) and Related Decrees, the Act No. 552/1991 Coll., on State Inspection and Monitoring in the Wording of Act No. 166/1993 Coll., the Act No. 500/ 2004 Coll., on Administrative Proceedings (the Administrative Code) with later modifications stay as the basis of Czech nuclear legislation system. This set of laws defines the safety principles or criteria, details the procedures to be applied to obtain the necessary authorizations, and the mechanism for inspections and evaluations. Basic principles determine that the responsibilities derived from the usage of nuclear energy remain with the licensee holder.

In accordance with constitutional law the State Office for Nuclear Safety (SÚJB) is fully competent authority for regulation in all areas of peaceful utilisation of nuclear energy and ionising radiation. It is in charge of regulatory supervision of safety of nuclear, radiation and transport safety, radiation protection, nuclear safeguards and of emergency preparedness in case of radiation accidents. The SUJB is an independent body of state state administration , by law reporting directly to the Government. The SÚJB use its own part of state budget approved by the Parliament of the Czech Republic. The SÚJB is headed by a Chairperson appointed by the Government as a body. In practice SÚJB Chairperson reports to the Government through the Prime Minister.

By law the SÚJB is entitled to issue Regulations to complete and clarify requirements established by appropriate acts of Parliament, such as Atomic Act. Based on experience with use of existing Czech nuclear legislation a comprehensive novelization of Atomic Act has actually started.

The following decrees the most significant for regulation of new reactors licensing:

- **No. 214/1997 Coll.,** on Quality Assurance in Activities Related to the Utilization of Nuclear Energy and in Radiation Activities, and Laying Down Criteria for the Assignment and Categorization of Classified Equipment into Safety Classes, (Actually updated as No.:132/2008 Coll.)
- **No. 215/1997 Coll.,** on Sitting of Nuclear Facilities and Very Significant Ionizing Radiation Sources,
- **No. 106/1998 Coll.,** on Nuclear Safety and Radiation Protection Assurance during Commissioning and Operation of Nuclear Facilities (update is prepared in the frame of the Harmonization process initiated by WENRA)
- **No. 195/1999 Coll.,** on Basic Design Criteria for Nuclear Facilities with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness (update is prepared in the frame of the Harmonization process initiated by WENRA)
- **No. 144/1999 Coll.** on Physical Protection of Nuclear Materials and Nuclear Facilities and their Classification, amended in Decree of the SÚJB No. **500/2005** Coll.
- **No. 307/2002 Coll**, on Radiation Protection,
- **No. 318/2002 Coll**. on Details of Emergency Preparedness of Nuclear Facilities and Workplaces with Ionizing Radiation Sources and on Requirements on the Content of On-Site Emergency Plan and Emergency Rule, amended in Decree SÚJB No. **2/2004** Coll.
- **No. 132/2008 Coll.** on provision of technical safety for classified equipment.
- **No. 185/2003 Coll.** on Decommissioning of Nuclear Facility or Category III. or IV. Workplace.

A set of the Regulatory Safety Guides was issued during last years to complement basic provisions given in Atomic Act and subsequent regulations. The whole set is available in the printed version and electronically on SUJB web pages. The strategy for development of these guides has considerably changed in last years. Mainly with the view of fundamental change in structure of the Czech nuclear legislative pyramid - introducing a new status to the Regulatory Safety Guides. Based on new provisions prepared for comprehensive amendment of Atomic Act these would be newly published in official Gazette and by this considered as official part of the legal pyramid in nuclear area. This arrangement would allow, where appropriate, to declare requirements of the regulator in specific areas but preserving the right of the licensee to propose an alternative procedure or solution. Actually a complete revision of guides focused to nuclear safety is under way. Both above mentioned measures would allow to complete implementation of WENRA reference levels to national legislative pyramid in full.

Situation of the national regulations in 2006 with respect to the RLs

In the 2006, the coverage of the RLs in the Czech national legislation was evaluated with respect to (at that time) existing legislation (Laws and Decrees).

Results of the 2006 RHWG report identified that there is considerable of RLs (172 from 288) have to be addressed to reach harmonization (C or B categories). The picture was evidently better for implementation of RLs at the plants, only 16 in "C" category (from 288).

Actions started to incorporate RLs in national regulations

Based on the results of the RHWG report and in accordance with the agreements in WENRA the SUJB prepared and approved an Action plan in order to harmonize the legislation till 2010.

The Action plan included the amendment of the Decree No. 195/1999 Coll., "on Basic Design Criteria for Nuclear Facilities with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness". Selected parts of WENRA reference levels are planned to be harmonized till the year 2010 by issuing of new or updated Regulatory Safety Guides

Status of the national regulations in September 2010 and envisaged further actions

The formal part of the harmonisation process is in some delay. The reason is formal - Atomic Act needs to undergo procedural/formal amendments. Remediation of this is possible only through Act of Parliament. The process is possible to start only after 2010 summer parliamentary elections. On the other hand all of the technical content of changes/amendments to individual regulations and guides is being prepared in line with Action plan.

Amendment of the Decree No. 195/1999 Coll., "on Basic Design Criteria for Nuclear Facilities with Respect to Nuclear Safety Radiation Protection and Emergency Preparedness" was prepared and was submitted to stakeholder consultation.

Five new SUJB Safety Guides have been published, five SUJB Safety Guides are under consultation/decision making process, and six SUJB Safety Guides are drafted or under revision for updating.

It is expected that all those Safety guides will be published soon around the end of the 2010 year. The overview of the set of the Guidelines is in following table. The set of Guidelines focused to construction of new plants is planned to complete and issue out of the harmonisation process.

1. Plant and system design	2. Safety management of a nuclear facility	3. Production and Construction
BN-JB-1.1 Requirements to Nuclear Safety, Radiation protection and	**BN-JB-2.1** Requirements to the Organisation, operating Nuclear facility	

Emergency Preparedness of Nuclear Facility		
BN-JB-1.2 Selection of the Design basis and Beyond Design Basis Events for Nuclear Power Plants	**BN-JB-2.2** Ageing Management on Nuclear Power plants	
BN-JB-1.3 Guideline for Safety Classification of Structures, Systems and Components of Nuclear Facilities	**BN-JB-2.3** Probabilistic Safety Assessment	
BN-JB-1.4 Protection against internal fires on Nuclear Facility	**BN-JB-2.4** Utilisation of operational experience on nuclear facility	
BN-JB-1.5 Approach to the assessment of new types of nuclear fuel in Czech Republic – Licensing requirements to the fuel system and core design	**BN-JB- 2.6** Guideline for the involvement of the Management Systems and Quality Assurance systems	
BN-JB-1.6 Rules and Basis for the Assessment of Computation Codes for the Nuclear Safety Analysis	**BN-JB-2.7** Guideline for the Nuclear Facilities personnel education, training and qualification verification	
	BN-JB-2.8 Maintenance, Operational Surveillance and testing on Nuclear Facility	
	BN-JB-2.9 Requirements to implementation of EOPs and SAMG on Nuclear Facilities	
	BN-JB-2.10 Modifications of Structures, systems, and processes on Nuclear Facility	

Country	FINLAND
General presentation of the regulatory system	

Legislative and regulatory framework

The current nuclear legislation in Finland is based on the Nuclear Energy Act from 1987. The Act has been changed 17 times during the years it has been in force: most changes are minor and originate from changes to other Finnish legislation. Contrary to these minor changes, nuclear legislation was updated and reformed in 2008 to correspond to current level of safety requirements and the new Finnish Constitution which came into force in 2000. The supporting Nuclear Energy Decree is from 1988 and was also reformed in 2008.

The current radiation legislation is based on the Radiation Act and Decree, both of which are from 1991 and take into account the ICRP Publication 60 (1990 Recommendations of the International Commission on Radiological Protection). Section 2, General principles, and Chapter 9, Radiation work, of the Act are applied to the use of nuclear energy.

Based on the Nuclear Energy Act, the Government issued in 2008 the following regulations:

- Government Decree on the Safety of Nuclear Power Plants (733/2008)
- Government Decree on the Security in the Use of Nuclear Energy (734/2008)
- Government Decree on Emergency Response Arrangements at Nuclear Power Plants (735/2008)
- Government Decree on the Safety of Disposal of Nuclear Waste (736/2008).

These new Government Decrees establish the mandatory nuclear safety (and security) requirements in Finland. The main reason for publishing them was the need to update safety requirements and to create a basis for the overall revision of Finnish regulatory guides (YVL Guides). It was also essential to examine the Decrees to verify the constitutionally appropriate legislative level of requirements. As a result of this examination, all requirements having principal nature were transferred from the Decrees to the Nuclear Energy Act and some requirements presented earlier in YVL Guides were transferred to the Decrees.

The Nuclear Safety Directive (Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations) affects slightly Finnish nuclear legislation. Also, international peer reviews concerning physical protection and waste management, both carried out in 2009, cause some amendments to legislation and/or other regulations. All these changes are currently under preparation.

Provision of regulatory guidance

According to the Section 7 r of the Nuclear Energy Act, STUK has a mandate to specify detailed safety requirements concerning the implementation of safety level in accordance with the Act. The safety requirements of STUK are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety level in accordance with this Act, STUK may approve this procedure or solution.

The procedure to apply new guides to existing nuclear facilities is such that the publication of an YVL Guide does not, as such, effect any previous decisions made by STUK. After having heard those concerned, STUK makes a separate decision on how a new or revised YVL Guide applies to operating nuclear power plants, or to those under construction, and to licensee’s operational activities. To new nuclear facilities, however, the guides apply as such.

Nowadays the most important references considered in rulemaking are the IAEA safety standards and WENRA reference levels. Other sources of safety information are worldwide co-operation with other countries using nuclear energy (e.g. MDEP, VVER Forum, OECD/NEA). The Finnish policy is to participate in the international discussion on developing safety standards and adopt or adapt the new safety requirements into national regulations. At the moment STUK has a set of about 70 regulatory guides in force. The regulatory guides have been continuously re-evaluated for updating.

More information about Finnish regulations can be obtained at: http://www.stuk.fi/en_GB/

Overall reform of YVL Guides

After revising the Nuclear Energy legislation in 2008, also the existing YVL guide system has been taken under work. The main objectives of this effort are the following:

- to restructure the guide system better to reflect the various areas of safety; at the same time to limit the total amount of guides and need for cross-referencing between the guides
- to compile requirements concerning related safety issues to the same guide making it easier to use by the licensees and other stakeholders; also they will be coupled to the stage of licensing process
- to rewrite the separate requirements in such a way that each requirement will have its own number, be short and clearly stating who-what-when shall be doing something; requirements are expressed in shall-format , descriptive text is provided only when necessary
- when considering the requirements, special attention is paid for the opportunities to limit unnecessary prescriptiveness
- to update the contents of the regulatory guides, especially with the lessons learnt from the Olkiluoto3 –project.

STUK has set an internal time schedule for this revision effort in such a way that all guides of the new system will be prepared at least to the level of a final draft before the end of 2010 and, that all new guides will be published before the end of 2011.

A Safety management of a nuclear facility	B Plant and system design	C Radiation safety of a nuclear facility and environment	D Nuclear materials and waste	E Structures and equipment of a nuclear facility
A.1 Regulatory control of the safe use of nuclear energy	B.1 Design of the safety systems of a nuclear facility	C.1 Structural radiation safety of a nuclear facility	D.1 Regulatory control of nuclear non-proliferation	E.1 Manufacture and use of nuclear fuel
A.2 Siting of a nuclear facility	B.2 Classification of systems, structures and equipment of a nuclear facility	C.2 Radiation protection and dose control of the personnel of a nuclear facility	D.2 Transport of nuclear materials and waste	E.2 Construction plan of the mechanical components and structures of a nuclear facility
A.3 Management systems of a nuclear facility	B.3 Safety assessment of a NPP	C.3 Control and measuring of radioactive releases to the environmental of a nuclear facility	D.3 Handling of spent nuclear fuel	E.3 Regulatory control of the mechanical components and structures of a nuclear facility
A.4 Organisation and personnel of a nuclear facility	B.4 Nuclear fuel and reactor	C.4 Radiological control of the environment of a nuclear facility	D.4 Handling of low- and intermediate-level waste and decommissioning of a nuclear facility	E.4 Verification of strength of pressure equipment of a nuclear facility
A.5 Construction of a NPP	B.5 Reactor coolant circuit of a NPP	C.5 Emergency preparedness arrangements of a NPP	D.5 Final disposal of nuclear waste	E.5 In-service inspections of the mechanical components and structures of a nuclear facility
A.6 Operation and accident management of a NPP	B.6 Containment of a NPP			E.6 Buildings and structures of a nuclear facility
A.7 Risk management of a NPP	B.7 Preparing for the internal and external threats to a nuclear facility			E.7 Electrical and I&C equipment of a nuclear facility
A.8 Ageing management of a nuclear facility	B.8 Fire protection of a nuclear facility			E.8 Oversight of inspection organisations

The re-structured system of regulatory YVL Guides

Situation of the national regulations in 2006 with respect to the RLs
There were 41 WENRA reference levels which were not included in the STUK's YVL Guides at the time when the reference levels were published and the self-assessments made.
Actions started to incorporate RLs in national regulations
Finland made an action plan how these missing reference levels will be included in the national regulatory requirements by the end of year 2010. Some of the missing reference levels have already been taken into account in updating the existing YVL Guides but some are still waiting for the overall reform of the YVL Guides which is currently ongoing at STUK (see above). Considering the WENRA reference levels published in 2007 and 2008, the Finnish policy is to include all of them in the revised regulatory guide system. This is confirmed already during the work through a systematic approach to earmark all the reference levels to certain guides.
Status of the national regulations in September 2010 and envisaged further actions
STUK has set an internal time schedule for this revision effort in such a way that all guides of the new system will be prepared at least to the level of a final draft before the end of 2010 and that all new guides will be published before the end of 2011. All reference levels will be included in the new YVL Guides practically as such.

Country	FRANCE
General presentation of the regulatory system	

➢ *Law and regulations issued by the government*

The legislative base governing the safety of nuclear installations in France is the act of 13 June 2006 on transparency and security in the nuclear field, referred to as the "**TSN act**", which fundamentally recasts the legal framework applicable to nuclear activities and their regulation. The TSN act introduces an integrated system based on a broader conception of nuclear safety, covering accident prevention and mitigation as well as protection of the health of persons and the environment, including during normal operation. The TSN act also establishes a nuclear safety authority (ASN, www.asn.fr), an independent administrative authority with responsibility for regulating nuclear safety and radiation protection and informing the public in these areas.

The government retains the power to set forth by decree or order any general regulations applicable to nuclear activities, after consulting formally ASN on these drat texts. It also takes a limited number of major individual decisions concerning nuclear facilities, notably for licensing their creation and dismantling.

Decree No. 2007-1557 sets forth the framework according to which new procedures will apply; it encompasses the full lifetime cycle of nuclear facilities (from the creation and commissioning licences up to final shutdown and dismantling. This decree describes in detail the applicable procedures for adopting general regulations and making individual decisions relating to nuclear facilities.

General technical regulations set forth by **ministerial orders** deal currently with five major topics (see paragraph above on the status in 2006). All were issued before the change of regulatory framework initiated by the TSN act and they will progressively by superseded by a new ministerial order and a set of ASN technical regulatory decisions (see paragraph below on regulation updating).

➢ *ASN decision and guidance*

ASN may complement (in particular implementation modalities) laws, decrees or orders by **technical regulatory decisions**, which are legally binding once validated by the relevant Minister, and takes individual decisions concerning nuclear activities (e.g., licences for commissioning nuclear facilities…) and sets forth individual requirements.

ASN used to issues basic safety rules (**RFS**) on various technical subjects concerning nuclear facilities (for example the use of PSA) ; these rules are recommendations, not legally binding, defining the safety objectives and describing practices which ASN considers satisfactory for achieving the objectives. A licensee may decide not to comply with the provisions of a RFS, providing he can demonstrate that the safety objectives defined by the rule can be achieved by the alternative means which he proposes to implement. Nowadays, ASN issues **guides**, not legally binding, that supersedes some of the RFS, clarify ASN's expectations or interpretation or regulations (e.g. criteria for event reporting) or provides recommendations.

➢ *Other documents*

Finally, there are ministerial letters, which were issued to the operator for each type of reactors before construction and aimed at defining the regulatory position on the main safety options.

Situation of the national regulations in 2006 with respect to the RLs

At the beginning of 2006, nuclear safety regulations in France, especially legally binding regulations, were seldom :

- one decree (N° 63-1227) dealt primarily with administrative processes to create, operate and dismantle nuclear facilities and another one (No 95-540) dealt with water intake and effluent discharge control ;
- general technical regulations set forth by ministerial orders dealt with five major topics: quality management (order of 10 August 1984), pressurised equipment (order of 26 February 1974 for the construction of PWR main primary system ; in service inspection of PWR main primary system and the main secondary systems were covered by the order of 10 November 1999), external nuisances and risks resulting from INB operation (order of 31 December 1999), water intake and effluent discharges to the environment (order of 26 November 1999).

From a more technical point of view, about 40 basic safety rules (RFS) – not binding but stating accepted practices – were published by ASN on various topics, some of them specific to NPP, other covering the whole range of nuclear facilities.

As only one utility (EDF) was operating all French NPP, much of the technical rules governing the design and operation of these NPP were set in ASN letters to EDF, usually letters accepting or amending EDF's proposals. EDF has also implemented safety provisions on its own.

As a consequence, only one third of WENRA RLs were actually covered by French safety regulations in force at the beginning of 2006.

Actions started to incorporate RLs in national regulations

France national action plan to update national regulation by transposing WENRA RLs was prepared since the end of 2005 and finally endorsed by ASN in June 2006. The plan was to write 5 new ministerial orders (safety policy and management, safety approach for PWRs, design of PWR, operation of PWRs, emergency preparedness and response) and a few associated guides. These text would also update two existing ministerial order (order of 10 August 1984, order of 31 December 1999). The initial schedule was to engage in stakeholder consultation mid-2007 and have regulations published in the second half of 2009.

An unexpected major change in France nuclear safety regulatory regime happened mid-2006 (publication of a new act : TSN act) then at the end of 2006 with the first meeting of ASN 5 commissioners, which meant the entry into force of the TSN act.

Following the publication of the TSN act (June 2006) and its associated decree on Basic Nuclear Installation regulation principles (November 2007), the initial ASN regulatory project and associated schedule related to WENRA activities underwent a major update in 2008. Two points are too be highlighted :

- one issue, not fully resolved today is the clear cut between provisions to be set forth in ministerial orders vs those to be in ASN's (regulatory) decisions. Provisions currently in draft ASN's decisions may be later transferred to the draft order (or vice-versa). ASN still plans to have all WENRA reference level addressed in the order(s) and decisions, or where relevant in published ASN guidance.
- another issue was to decide whether some reference levels are applicable only for reactors or for all types of nuclear facilities. The choice made was to privilege as much as possible provisions applicable to all nuclear facilities.

As a consequence, WENRA RLs will be transposed by :

1) some provisions of the TSN Act and November 2007 decree (already published);
2) one ministerial order stating the overarching provisions;
3) 10 ASN technical regulatory decisions dealing with:

a) ASN decision on safety policy and management of nuclear facilities;
b) ASN decision on Modifications of nuclear facilities;
c) ASN decision on Periodic Safety Reassessment of nuclear facilities;
d) ASN decision on Safety Analysis Report of nuclear facilities;
e) ASN decision on general operating rules (RGE) of nuclear facilities;
f) ASN decision on operation of nuclear facilities;
g) ASN decision on design of PWR;
h) ASN decision on emergency management ;
i) ASN decision on fire;
j) ASN decision on nuclear pressurized equipment (which will mostly supersede current ministerial order);

A very limited number of ASN Guide, mainly one Safety Policy.

<u>Status of the national regulations in September 2010 and envisaged further actions</u>

All the regulation/technical regulatory decisions drafts have been written, except the one on nuclear pressurised equipments (but current regulation enables consistency with WENRA RLs).

The status of the regulations development, as of September, 2010 is presented hereafter :

	ASN preliminary in-house formal review	Stakeholder consultation	Updating of the draft	ASN in-house formal review	Issuance of the regulations
Ministerial order	✓	✓	In process		
ASN decision on safety policy and management of nuclear facilities	✓	✓			
ASN decision on Modifications of nuclear facilities	✓	✓	In process		
ASN decision on Periodic Safety Reassessment of nuclear facilities	✓	✓	In process		
ASN decision on Safety Analysis Report of nuclear facilities	In process				
ASN decision on general operating rules (RGE) of nuclear facilities	✓	✓			
ASN decision on operation of nuclear facilities;	✓	✓			
ASN decision on design of PWR	In process				
ASN decision on emergency management	✓	✓	In process		
ASN decision on fire					
ASN decision on nuclear pressurized equipment					

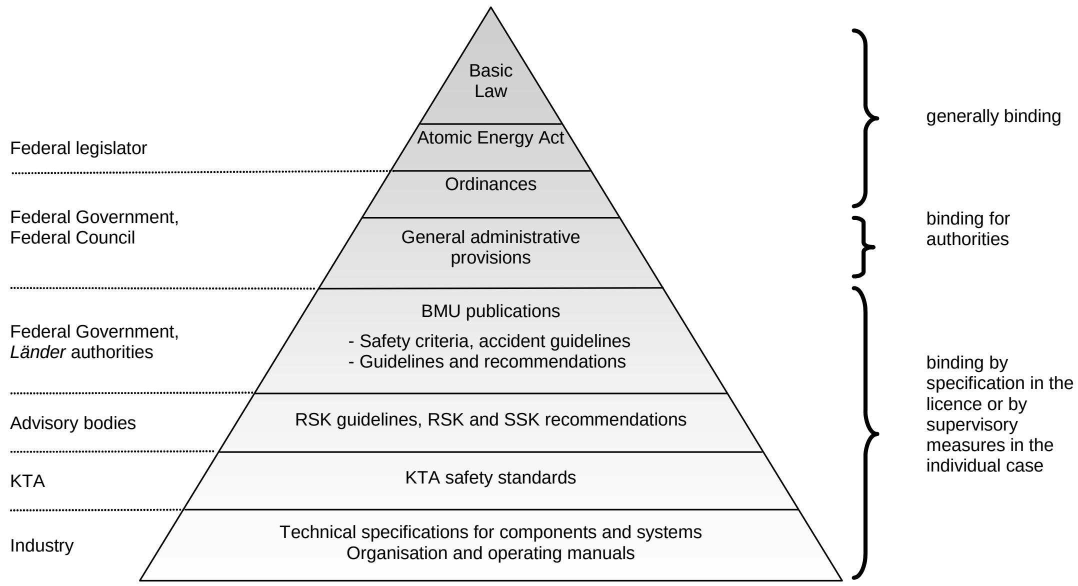
Once stakeholder consultation is over, ASN will update the draft to take into account comments received (if major technical issue were highlighted, ASN would consult its standing experts groups to get advice on the most appropriate way to handle the issue) and have the final text signed by its commissioner.

Once the ASN decision is signed by the commissioners, it then has to be validated (or rejected) by the Government which has a 2 months period to do so.

Mot of ASN decision provisions are closely related to general provision set forth in the ministerial order as they actually give details on the regulatory requirements. ASN current priority is to work closely with the Government to dispose of the comments received and have the order signed by the Government as soon as possible.

Taking into account these final steps, the ministerial order may be signed by the end of 2010 but more likely in the first half of 2011. ASN's plan is to issue its decisions once the order is signed, during the first half of 2011.

Country	GERMANY
General presentation of the regulatory system	
The Republic of Germany is a federal state. The responsibilities for legislation and law enforcement are assigned to the organs of the Federation and the *Länder* according to their scope of functions. Specifications are given by provisions of the Basic Law of the Federal Republic of Germany. The Federal Government has the legislative competence for the use of nuclear energy for peaceful purposes according to Article 73 para 1 number 14 in conjunction with Article 71 of the Basic Law. According to Section 24 para 1 of the Atomic Energy Act in conjunction with Article 87c, 85 of the Basic Law, the Atomic Energy Act and the statutory ordinances based thereon are executed - with some exceptions - by the *Länder* on behalf of the Federation. In this respect, the *Länder* authorities are under the oversight of the Federation with regard to the legality and expediency of their actions. The competent supervisory and licensing authorities report to the Federation on law enforcement on demand. The Federation has the right to require the submission of reports and documents and may, in the individual case, issue binding directives to the *Land* authority. The Federation may assume the competence for the subject matter, i.e. the decision in the cause, by exercising its right to issue directives. The competence to execute the duties, i.e. the execution of the decision towards the applicant or licensee, remains with the competent *Land* authority. Within the framework of nuclear procedures, other legal regulations, such as the immission control act, water law and construction law, also have to be considered. Legal regulations on assessing the environmental impact are usually part of the nuclear licensing procedure. In Germany, decisions of the public administration, so-called administrative acts, can be appealed before the administrative courts by the party concerned, e.g. by applicants and licensees and also by third parties of the public concerned (guarantee of recourse to the courts according to Article 19 para 4 of the Basic Law). An action is brought against that authority which issued the notice/administrative act, i.e. the respective competent *Land* authority. This also applies to the case that the *Land* took a decision due to a directive issued by the Federation. The parties concerned may also take legal actions in case of failure of the authorities to act. So, e.g., the plant operators may claim for granting of licences applied for or the residents for issuance of a regulatory order to cease operation of a nuclear installation.	
Situation of the national regulations in 2006 with respect to the RLs	
The figure presents the hierarchy of the national regulations, the authority or institution issuing them and their degree of bindingness.	



Acts, ordinances and administrative provisions: The **Basic Law** includes provisions on the legislative and administrative competencies of the Federation and the *Länder* regarding the use of nuclear energy. Moreover, fundamental principles are established that are also applicable to the nuclear law.

With the basic rights, in particular the right to life and physical integrity, it determines the standard to be applied to the protective and preventive measures at nuclear power plants which is further specified in the above hierarchy levels of the pyramid. The principle of proportionality and guaranty of property, laid down in the Basic Law, must also be considered.

The **Atomic Energy Act** was promulgated on December 23, 1959, right after the Federal Republic of Germany had officially renounced any use of atomic weapons. Since then, it has been amended several times. The purpose of the Atomic Energy Act after the amendment of 2002 is to end the use of nuclear energy for the commercial production of electricity in a structured manner and to ensure on-going operation until the date of discontinuation, as well as to protect life, health and property against the hazards of nuclear energy and the detrimental effects of ionising radiation and, furthermore, to provide for the compensation for any damage and injuries incurred. It also has the purpose of preventing the internal or external security of the Federal Republic of Germany from being endangered by the utilisation of nuclear energy.

The Atomic Energy Act includes the general national regulations for protective and preventive measures, radiation protection and the disposal of radioactive waste and irradiated fuel elements in Germany and is the basis for the associated ordinances.

Further to purpose and general provisions, the Atomic Energy Act also comprises surveillance regulations, general regulations on competencies of the administrative authorities, liability provisions and provisions on the payment of fines.

With respect to the protection against the hazards from radioactive materials and to the supervision of their utilisation, the Atomic Energy Act requires that the construction and operation of nuclear installations is subject to regulatory licensing. Prerequisites and procedures for licensing and performance of supervision are specified, including the regulations for consulting experts (Section 20 of the Atomic Energy Act) and charging of costs (Section 21 of the Atomic Energy Act). According to Section 7 of the Atomic Energy Act, a licence is required for the construction, operation or any other holding of a stationary installation for the production, treatment, processing or fission of nuclear fuel, or for essentially modifying such installation or its operation.

However, most of the regulations laid down there are not exhaustive and are further specified both regarding the procedures and the substantive legal requirements by ordinances and regulatory guidance instruments.

In addition to the Atomic Energy Act, the **Radiation Precautionary Act** of 1986, which came about in the wake of the reactor accident at Chernobyl, specifies the tasks of environmental monitoring also in the case of events with significant radiological effects.

For more details regarding the legal regulations, the Atomic Energy Act includes authorisations for issuing **ordinances** (cf. listing in Section 54 para 1 of the Atomic Energy Act). These ordinances require approval by the *Bundesrat* (Federal Council). The *Bundesrat* is a constitutional body of the Federation in which the governments of the *Länder* are represented.

The table presents the current ordinances on protective and preventive measures relevant to the scope of this report.

	Brief description on the legislative content
StrlSchV	**Radiation Protection Ordinance** Principles and limits of radiation protection, requirements on organisation of radiation protection, personal monitoring, environmental monitoring, accident management, design against incidents and accident planning values
AtVfV	**Nuclear Licensing Procedure Ordinance** Application documents (one safety analysis report), involvement of the public, safety specifications (operational limits and conditions for safe operation), procedures and criteria for major modifications (public participation)
AtSMV	**Nuclear Safety Officer and Reporting Ordinance** Position, duties, responsibilities of the nuclear safety officer, reporting of special events in nuclear installations
AtZüV	**Nuclear Reliability Assessment Ordinance** Checking of personal reliability for protecting against the diversion or major release of radioactive material
AtDeckV	**Nuclear Financial Security Ordinance** Financial security pursuant to the Atomic Energy Act
AtKostV	**Cost Ordinance under the Atomic Energy Act** Fees and costs in nuclear procedures

Ordinances may include additional authorisations for issuing **general administrative provisions**. General administrative provisions regulate the actions of the authorities, thus only having a direct binding effect for the administration. However, they have an indirect effect if serving as a basis for concrete administrative decisions.

Regulatory guidelines published by BMU: After having consulted the *Länder*, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) prepares regulatory guidelines. These are, among others, safety criteria, accident and other guidelines and recommendations. In general, these are regulations passed in consensus with the competent licensing and supervisory authorities of the *Länder* on the uniform application of the Atomic Energy Act. The recommendations of the BMU, however, describe its view on general questions related to nuclear safety and the administrative practice, and serve as orientation for the *Länder* authorities regarding the enforcement of the Atomic Energy Act. The regulatory guidelines are not binding for the *Länder* authorities in contrast to the general administrative provisions. Their relevance is also given by the right of the BMU to issue binding individual directives for particular cases to the *Länder* authorities.

Currently, about 60 BMU regulatory guidelines exist in the field of nuclear technology. Regarding the scope of this report these are regulations pertaining to

- general safety requirements for nuclear power plants ("Safety Criteria"),
- details on the design basis accidents to be considered in the design of pressurised water reactors (since 1982 for the last three nuclear power plants built of construction line 4),
- accident management measures to be planned by the plant operators with regard to postulated severe accidents,
- reporting criteria for reportable events at nuclear power plants and research reactors,
- periodic safety reviews for nuclear power plants,
- technical documents to be prepared regarding construction, operation and decommissioning of nuclear power plants,
- documents to be supplied with the application for a licence,
- procedures for the preparation and performance of maintenance and modification work in nuclear power plants, and
- qualification of the personnel in nuclear installations.

Recommendations of the RSK or the SSK; RSK guidelines: The BMU requests the Reactor Safety Commission (RSK) and the Commission on Radiological Protection (SSK) for advice on important issues related to licensing and supervisory procedures, development of rules and regulations or safety research. Depending on the issues to be discussed, Länder authorities, plant operators or the industry also participate in the discussions. The results of these discussions are statements or recommendations for the BMU. After own verification, the BMU implements the results in the respectively appropriate manner.

The so-called RSK guidelines play a special role. In the last version of these guidelines of 1996, the RSK compiled the fundamental safety requirements for nuclear power plants with pressurised water reactors. The nuclear licensing authorities of the *Länder* have taken the RSK guidelines as an assessment basis within the framework of the regulatory guidance instruments for plants whose licences on the site and safety concept were to be granted after entry into force of the RSK guideline and made them binding for the plant operator by the licence permit. For plants that were granted a licence before, the RSK guidelines were referred to for assessing the adequacy of the further development of plant safety.

KTA safety standards: The Nuclear Safety Standards Commission (KTA) was established at the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety. It is made up of the five interest groups: representatives of the manufacturers, the plant operators, the federal and *Länder* authorities, the expert organisations and representatives of general concerns, e.g. of the unions, the industrial safety and the liability insurers.

The regulatory powers of the legislator and administrative action by the competent authorities are not restricted by the KTA process. It is possible to formulate necessary requirements, guidelines and recommendations and to implement them on the basis of the Atomic Energy Act regardless of the consensual formulation of KTA safety standards.

On the basis of the regular reviews and, where required, amendment of the issued safety standards at intervals of no more than five years, the standards are adjusted to the state of the art in science and technology. In themselves, KTA safety standards are not legally binding. However, due to the nature of their origin and their high degree of detail, they have a far-reaching practical effect.

Until today, the KTA has issued a total of 91 safety standards and 3 draft standards. 12 draft standards are in preparation and 50 safety standards are in the process of being revised. The following draft safety standards relevant to the scope of this report are in preparation :

- [KTA 1203] "Requirements for the Emergency Manual“,
- [KTA 1402] “Management Systems for the Operation of Nuclear Facilities",
- [KTA 1403] “Ageing Management in Nuclear Power Plants",
- [KTA 3206] “Demonstration of Break Preclusion for Pressure Retaining Components in Nuclear Power Plants“

Revision of the nuclear rules and regulations

The applicable national higher-level nuclear rules and regulations date back to the 1970s and 1980s. In science and practice there is consensus that the modernisation and further development of the higher-level nuclear rules and regulations is necessary. The drafting process for the development of the new rules and regulations started in September 2003.

This was the starting point for the RHWG benchmarking, which showed the high number of RL needed to be adressed for harmonization: 90 RL with a “C” assessment and 26 with a “B”.

Actions started to incorporate RLs in national regulations

Since 2006 several actions were performed to overcome the deficiencies in the 2006 national action plan and to implement the WENRA RL into national regulation. These actions include an updating of existing ordinances, the development of a new ordinance and mainly the development of new modular regulations. The activities can be summarized as follows:

- First draft of National Action Plan, Nov. 2006, published on BMU-homepage (harmonization planned via new “Safety Requirements for Nuclear Power Plants” by end of 2007);
- Revision D of “Safety Requirements for Nuclear Power Plants” available since April 2009. Pilot phase with test application until 10/2010 is ongoing;
- Update of “Nuclear Safety Officer and Reporting Ordinance” (AtSMV).

Status of the national regulations in September 2010 and envisaged further actions

The applicable national higher-level nuclear rules and regulations date back to the 1970s and 1980s. In science and practice there is consensus that the modernisation and further development of the higher-level nuclear rules and regulations is necessary. This view is also shared by the Federation and the *Länder*.

The drafting process for the development of the new rules and regulations started in September 2003.

With the new *Safety Criteria for Nuclear Power Plants*, the draft of a new higher-level nuclear rules and regulations is now available. The Federal Ministry for the Environment published the draft of the new “Safety Criteria for Nuclear Power Plants - Revision D, of April 2009” on the Internet.

The new Safety Criteria are to ensure the integration of existing rules, current practice, international requirements and new scientific findings, and to replace the *Safety Criteria for Nuclear Power Plants, as of 1977*, the *RSK guidelines for pressurised water reactors*, as of 1981 with updates of 1996, and the *accident guidelines* of 1983.

Against this background, the Federal Ministry for the Environment and the *Länder* of Baden-

Wuerttemberg, Bavaria, Hesse, Lower Saxony and Schleswig-Holstein have agreed upon a comprehensive consultation procedure for further action with the *Länder*, power utilities and science.

The agreed procedure aims to contribute to gain practical experience in the application of the new Safety Criteria and evaluate it in a process agreed between the Federation and the *Länder* (see below). The test phase started on 1 July 2009 and will end on 31 October 2010. On this basis of practical experience gained from testing, the Federation and the *Länder* jointly review the rules and regulations by mid-2011. The Federation and the *Länder* are striving for a unanimous adoption of the nuclear rules and regulations. Publication by the Federal Ministry for the Environment in the Federal Gazette will not take place before the end of the procedure.

The Federation and the *Länder* will apply the new Safety Criteria for Nuclear Power Plants on a trial basis and in parallel to the higher-level rules and regulations relevant so far in nuclear procedures. This application takes place in nuclear licensing procedures and modifications procedures requiring approval (including PSR, reportable events and hazard assessment) in order to gain experience with the application of all modules. In this respect, all modules of the draft of the new Safety Criteria relevant for the procedures to be selected are applied:

- MODULE 1 “Safety Criteria for Nuclear Power Plants: Fundamental Safety Criteria“;
- MODULE 2 “Safety Criteria for Nuclear Power Plants: Criteria for the Design and Operation of the Reactor Core“;
- MODULE 3 “Safety Criteria for Nuclear Power Plants: Events to be Considered for Pressurised and Boiling Water Reactors“;
- MODULE 4 “Safety Criteria for Nuclear Power Plants: Criteria for the Design of the Reactor Coolant Pressure Boundary, the Pressure Retaining Walls of the External Systems and the Containment System“;
- MODULE 5 “Safety Criteria for Nuclear Power Plants: Criteria for Instrumentation and Control and Accident Instrumentation“;
- MODULE 6 “Safety Criteria for Nuclear Power Plants: Criteria for Safety Demonstration and Documentation“;
- MODULE 7 “Safety Criteria for Nuclear Power Plants: Criteria for Accident Management“;
- MODULE 8 “Safety Criteria for Nuclear Power Plants: Criteria for Safety Management“;
- MODULE 9 “Safety Criteria for Nuclear Power Plants: Criteria for Radiation Protection“;
- MODULE 10 “Safety Criteria for Nuclear Power Plants: Criteria for the Design and Safe Operation of Plant Structures, Systems and Components“;
- MODULE 11 “Safety Criteria for Nuclear Power Plants: Criteria for the Handling and Storage of the Fuel Elements“;
- MODULE 12 “Safety Criteria for Nuclear Power Plants: Criteria for Electric Power Supply“

The new "Safety Criteria for Nuclear Power Plants" are intended to be an element of Germany to fill existing gaps in the nuclear rules and regulations.

Country	HUNGARY
General presentation of the regulatory system	

The Hungarian Parliament approved the current Act on Atomic Energy in December 1996 (the Act on Atomic Energy) which entered into force on July 1, 1997. The Act on Atomic Energy accounts for all legislative, authority-related and operational experience gained during the construction and operation of Paks NPP, it considers the technological development achieved since the issue of the previous Act on Atomic Energy, all international obligations, and also integrates the requirements of the Convention. The Atomic Act has reinforced the distributed regulatory system, which delegates the responsibilities for nuclear safety, radiation protection and environmental protection related to nuclear facilities to different authorities.

The Hungarian Atomic Energy Authority's scope of competence comprises nuclear safety licensing (at the levels of the facility, systems and components) and supervision of nuclear installations, registration and supervision of radioactive materials, licensing of transportation and packaging, licensing of nuclear exports and imports, evaluation and co-ordination of research and development, performance of authority-specific tasks related to nuclear emergency preparedness, approval of the emergency response plans of nuclear installations, and maintenance of international relations. It is also the duty of the Authority to perform the tasks generated by the treaty concluded with the International Atomic Energy Agency dealing with the non-proliferation of nuclear weapons, along with the registration and supervision of nuclear substances.

In 2003 the Parliament amended the Atomic Act CXVI of 1996, and according to this decision a dedicated minister (currently the Minister of National Development) appointed by the Prime Minister became the supervisor of the HAEA.

In 2005, a revised set of Nuclear Safety Requirements (Regulations) was issued as attachment to the Governmental Decree 89/2005(V.5) on the Nuclear Safety Requirements of Nuclear Facilities and Related Regulatory Activities. The Nuclear Safety Requirements consist of seven volumes. The first four volumes address the NPPs:

- Volume 1: Regulatory Procedures for NPPs,
- Volume 2: Management System of NPPs,
- Volume 3: Requirements of Design of NPPs,
- Volume 4: Requirements of Operation of NPPs.

Additionally to the legally binding requirements, the Director General of the HAEA issues guidelines containing recommendations on how the requirements should be implemented in the regulatory processes.

More information can be found on the web site of the HAEA: www.haea.gov.hu.

Situation of the national regulations in 2006 with respect to the RLs

The benchmark - performed in 2006 - showed that most of the reference levels were covered by the Regulations. 37 Reference Levels were evaluated as C (a difference exists, and should be addressed for harmonization). They were identified

- mainly within the issue E (Design Basis Envelope for Existing Reactors), LM (Emergency Operating Procedures and Severe Accident Management Guidelines) and O (Probabilistic Safety Analysis) and
- a few were related to issue A (Safety Policy), F (Design Basis Envelope for existing reactors), J (System for Investigation of Events and Operational Experience Feedback) and

S (Protection against internal fires).

Actions started to incorporate RLs in national regulations

The Nuclear Safety Requirements (Regulations) were used for the benchmarking of the Reference Levels in 2006. Most of the reference levels were covered by the Regulations, but based on the result of the benchmarking an action plan was established in order to make the Regulations complete.

The Regulations shall be reviewed and updated as needed at least once in every five years on the basis of scientific achievements as well as domestic and international experience.

The last review aimed – among others – to improve the Regulation based on the Reference levels.

Status of the national regulations in September 2010 and envisaged further actions

The review started in 2006 and finished at the end of 2009. The bases of the review were the WENRA reference levels (and the action plan), the recently issued IAEA standards, lessons learned from the use of Hungarian regulations and recommendations from the international reviews. Now, the draft of the new set of Regulations is under administrative collation:

- Volume 1. – Regulatory procedures of Nuclear Facilities
- Volume 2.– Management System of Nuclear Facilities
- Volume 3.– Design of NPPs
- Volume 4.– Operation of NPPs
- *(Volume 5.– Design and Operation of Research and Training Reactors)*
- *(Volume 6.– Design and Operation of Spend Fuel Storage Facilities)*
- Volume 7.– Siting
- Volume 8.– Decommissioning
- Volume 9.– Terminology

The new set of Regulation will cover all Reference Levels.

Country	ITALY
General presentation of the regulatory system	

The Regulatory System related to nuclear installations, presently in force Italy, is the result of an evolution of rules and standards that begun in the early ‘60s and that took the experience of licensing and operation of nuclear power plants of different types and generation into account. The Italian regulatory system is made up of three types of rules of different legal force depending on their origin; the first two types are the most relevant for this study: legislation by the Parliament and Decrees by Government or Ministries and Technical guides; the third type of rule is mainly made up by industrial standards.

Main legislation and ministerial decrees

In the Italian system the source of legally binding rules must be either an act of Parliament (statute) or a Legislative Decree; the Government can issue governmental or ministerial decrees binding in law. An important feature of legally binding rules concerning Safety and Radiation Protection is that contravention to obligations by operators and/or users constitutes a misdemeanor and entails a penal sanction; compliance can be enforced by means of criminal proceedings after due process of law.

The main corpus making up, inter alia, the Italian system are itemized below, as regards Statutes and Legislative acts:

- Act no. 1860 of 31 December 1962 published in the Italian Official Journal no. 27 of 30 January 1963, the basic Atomic Law on the peaceful uses of nuclear energy.
- The Presidential Decree no. 185 of 1964: "Safety of plants and protection of workers and general public against the risk of ionising radiation associated to the peaceful use of Nuclear Energy” replaced in 1996 by the Legislative Decree no. 230/1995.
- Legislative Decree no. 230 of 17 March 1995 published in the Supplement to Italian Republic’s Official Journal no. 136 of 13 June 1995, implementing six EURATOM Directives on radiation protection (EURATOM 80/836, 84/467, 84/466, 89/618, 90/641 and 92/3).
- Presidential Decree no. 1450 containing requirements and procedures for the acquisition of the operational personnel licences (1971).
- Presidential Decree no. 519/1975 “Civil responsibilities in the field of nuclear safety”.
- Legislative Decree no. 241 of 31 August 2000, implementing the 96/29/EURATOM directive regarding “Health protection of the population and workers against the risks deriving from ionising radiations”.

Several Acts of legislative force were issued for the institution of the Regulatory Body and for its subsequent re-organisations. The first one was Act no. 933 (1960), establishing the National Committee for Nuclear Energy (CNEN), and the last one was Law no. 133 (2008) instituting the Institute for Environmental Protection and Research (ISPRA).

The mandate of ISPRA is more generally addressed to Environmental Protection issues; one ISPRA Department has the mission to discharge the Regulatory Body responsibilities coming from the above-mentioned Laws. In this frame, the Institute performs licensing and inspection activities for any civil Nuclear Installation, performs inspections related to Physical Protection and Safeguards, provides technical support for setting up regulations, for planning and implementing Radiological Emergencies measures.

Technical guides

The issue of technical guides, previously carried out by the Directorate for Nuclear Safety and Health Protection, is now assigned in Law to ISPRA by article 153 of the Legislative Decree no. 230/1995. The guides contain recommendations and address to the implementation of rules of good practice. They "de facto" assume a mandatory nature during the regulatory assessment activity when the level of compliance of the application is verified. Twenty eight technical guides have been issued on Safety and Radiation Protection matters ranging from procedural to detailed technical guidance. They are publicly available and have been always issued after consultation of all the stakeholders.

Latest changes

After public announcements by the Italian government about the intention of adopting a new energy policy, including the construction of new nuclear power plants, a new Law has been issued in July 2009 n. 99 delegating the Government to issue legislative decrees regulating the siting of nuclear installations (in particular NPPs and long term waste storage facility) and to update the licensing procedures for construction and operation. The same Law establishes also new Agency for Nuclear Safety. It has to be implemented on the basis of specific decrees still to be issued. The Agency will be staffed by experts of the Nuclear Department of ISPRA and of the ENEA. Until the new Agency is fully implemented, its role continues to be performed by the Nuclear Department of ISPRA. The Agency will be charged of regulation and control in nuclear safety, radiation protection, safeguards and physical protection.

On February 2010 Legislative Decree n.31 has been issued, establishing in general terms the following procedures:

- for identification of suitable area and certification of sites
- for certification of operators
- for issuing a single authorization act for construction and operation.

The main authorizations will be granted through a concerted act by Ministry of Economic Development, Ministry of Environment, land and sea, and Ministry of Infrastructures, based upon the binding technical advise of the Nuclear Safety Agency.

In order the new regulatory system to become operative, implementation decrees are required mainly related to the Nuclear Safety Agency (Statute, Management nomination, organization, assignment of resources and headquarters location); moreover, a Nuclear strategy document justifying the nuclear option has to be issued according to the new Laws.

Of course, in case of restart of concrete nuclear program, due to the new licensing process defined by the Law, a process of updating technical guides will be needed and the reference levels defined by WENRA will be systematically considered.

Situation of the national regulations in 2006 with respect to the RLs

More than 100 RLs were found as not present in the Italian regulations but no modification was established to be needed due to status of the Italian NPP (being decommissioned) and due to the fact that no new plant construction was foreseen.

In any case the differences have been systematically traced.

Actions started to incorporate RLs in national regulations

The action plan that was defined in 2006 took strictly into account the fact that Italian NPPs (Garigliano - BWR, Latina-Magnox, Trino-PWR and Caorso-BWR) were no longer in operation since many years.

At present they still are, at different stages, in the process of being decommissioned. The main activities that are conducted on the sites are related to waste management (conditioning and site storage), fuel removal - for the NPPs still having fuel in the pools (Trino and Caorso) - as well as dismantling.
Taking this specific situation into account, the national action plan for harmonization to the WENRA Reactor Safety Reference Levels was prepared considering only those reference levels that were relevant for the above activities, with the additional intent to coordinate such actions with those required in the Action Plan for the harmonization to the waste storage and decommissioning reference levels. In this light, attention was therefore basically addressed to management, organizational, quality and fire protection issues. Other issues are considered not to be a national priority for the time being, they will be at the moment. They will be reconsidered when concrete development steps toward the construction of new plants will be performed.

The action plan was based on a two steps approach:

- Adopt reactor safety reference levels for which equivalent legal requirements in Italy are not available, after having evaluated their relevance for decommissioning and waste management, by issuing specific requirements to the existing facilities (letters to the licensee, or conditions to the new licenses);
- Issue the new regulations as proposed in the action plan on the basis of the final reference levels in the area of waste storage and decommissioning, by revising existing technical guides or issuing new ones.

In particular, new technical guides were foreseen to be issued on the following topics:

1) Management of nuclear facilities
2) Classification, conditioning and safe storage of radioactive waste
3) Safety Requirements on decommissioning of nuclear facilities
4) Fire protection of nuclear facilities

Status of the national regulations in September 2010 and envisaged further actions

This programme had to be confirmed in the frame of the definition of the Action Plans related to Waste storage and Decommissioning safety reference levels, which are still not finalized. Some drafting efforts have been however already done and such drafts are already used as review guidance.

Country	LITHUANIA

General presentation of the regulatory system

In compliance with the Law on Nuclear Energy and the Statute of VATESI approved by Government, as well as other legal documents, The State Nuclear Power Safety Inspectorate (VATESI) is the main regulatory and oversight institution of nuclear safety, which sets safety requirements, controls whether they are complied with, issues licenses and permits, performs safety assessments and other functions. VATESI mission is to perform the state regulation and oversight of safety at nuclear facilities in order to protect the public and environment against harmful effects of ionizing radiation.

VATESI competence covers state regulation and oversight of safety at nuclear installations, state regulation and oversight of nuclear waste management at nuclear installations; oversight of use of nuclear materials and technologies for peaceful purposes (the IAEA and EURATOM safeguards), state regulation and oversight of physical protection of nuclear installations and materials, emergency preparedness, state regulation and oversight of transportation of nuclear fuel cycle materials.

The main legal document governing nuclear energy is the Law on Nuclear Energy passed by Parliament in 1996. Other laws directly related to regulation of nuclear energy are:

- Law on Nuclear Waste Management;
- Law on Radiation Safety;
- Law on control import, export and transit of strategic commodities;
- Law on Civil protection;
- Law on Construction.

VATESI has a responsibility to issue two types of regulations: Requirements and Rules. Requirements establish the requirements that must be met to ensure safety, Rules - define the way how Requirements could be fulfilled. Both Requirements and Rules are mandatory for licensee. The application of standards is voluntary except the cases when regulations define specific standards to be applied. If the standard is voluntary accepted by licensee its application becomes mandatory.

The set of draft laws including new issue of Law on Nuclear Energy, linked to regulatory system, assurance of nuclear and radiation safety, licensing were developed presented to Parliament for approval. In the draft laws some changes are foreseen in licensing process due to plans to build a new NPP. The special new Law on Nuclear Safety will be devoted for assurance of nuclear safety and licensing of nuclear facilities. The approval of the laws is expected by the end of 2010. or in the beginning of 2011

Situation of the national regulations in 2006 with respect to the RLs

A set of legal documents was assessed. According to benchmarking results 173 of RLs were covered in substance by legal acts, 25 RLs were not fully covered, but the differences can be justified from a safety point of view, and 93 RLs should be addressed for harmonization.

Actions started to incorporate RLs in national regulations

Following a commitment to harmonize the regulatory requirements for nuclear safety with WENRA RLs VATESI has performed corresponding assessment and the Action plan for implementation of RLs into regulations was prepared in 2006. The action plan was updated in 2008 taking into account the final edition of WENRA RLs. According to the plan 14 nuclear safety regulations had to be issued in period of 2008-2010 years to cover 120 RLs, which were foreseen to be implemented in legal basis.

Status of the national regulations in September 2010 and envisaged further actions
By the end of September 2010 three new VATESI regulations were issued covering 73 from 120 RLs: • Requirements on the Operational Experience Feedback in the field of Nuclear Energy (2 RLs) • Requirements for deterministic safety analysis of Ignalina NPP (9 RLs) • Requirements for management systems in nuclear facilities (29 RLs) • The main requirements for assurance of safety of nuclear power plants with RBMK-1500 type reactors (33 RLs) By the end of September 2010 seven preliminary drafts of new regulations covering 34 RL's were passed for internal VATESI review: • Requirements for staff management in nuclear facilities (4 RLs) • Requirements for probabilistic safety analysis (17 RLs) • Rules for design of reactor containment systems for NPP (3 RLs) Above listed drafts will be further developed taking into account their significance after shut down of Ignalina NPP and development of new NPP project. Since the regulatory requirements system is periodically updated it is necessary to perform the monitoring and control the status of RL`s implementation. The order on implementation of WENRA RL's in the set of regulations is approved by Head of VATESI, which will help to keep track of implemented RL's and control further implementation of remaining 23 RLs and all RLs in the regulatory system, which is mostly important for regulation of potential new built and is being planed to be implemented in 2011-2012 (before corresponded regulation become actually needed)..

Country	THE NETHERLANDS
General presentation of the regulatory system	

Nuclear Energy Act

The basic legislation governing nuclear activities is contained in the Nuclear Energy Act

('Kernenergiewet' or Kew). It is a framework law. (1) The *registration* of fissionable materials and ores is regulated. (2) A licence is required in order to *transport*, *import, export*, be in *possession* of or *dispose* of fissionable materials and ores. (3) Licences are also required for *building, operating and decommissioning* nuclear installations (Section 15b), as well as for nuclear driven ships (Section 15c). The Act distinguishes between construction licences and operating licences.

Environmental Protection Act

According to this Act and the associated Environmental Impact Assessment Decree, the licensing procedure for the construction of a nuclear facility includes a requirement to draft an Environmental Impact Assessment (EIA) report. In certain circumstances, an EIA is also required if an existing plant is modified (e.g. change in fuel enrichment, decommissioning).

General Administrative Act (Awb)

The General Administrative Act sets out the procedure for obtaining a licence and describes the participation of the general public in this procedure (i.e. objections and appeals).

Decrees

A number of Decrees have also been issued containing additional regulations and these continue to be updated in the light of ongoing developments. Important examples of these in relation to the safety aspects of nuclear installations are:

- the Nuclear Installations, Fissionable Materials and Ores Decree (Bkse);
- the Radiation Protection Decree (Bs);
- the Transport of Fissionable Materials, Ores and Radioactive Substances Decree (Bvser);
- the Environmental Impact Assessment Decree.

Regulations and guides issued by regulatory body: the Nuclear Safety Rules (NVRs)

The Nuclear Energy Act (Article 21.1) provides the basis for a system of more detailed safety regulations concerning the design, operation and quality assurance of nuclear power plants. These are referred to as the Nuclear Safety Rules ('Nucleaire VeiligheidsRegels', NVRs). The regulations of the NVRs apply to an installation, as far as they are referenced in the licence. The NVRs are based on the Requirements and Safety Guides in the IAEA Safety Standards Series (SSS) from the '80 and '90.

The documents contain adaptations ('amendments' as they were termed) to the IAEA standards, but the character of the original IAEA standards is largely maintained. The amendments were formulated include the RLs and to adjust the documents to the circumstances and needs in the Netherlands.

For more detailed information on the regulatory framework, please see the Dutch report to the Convention on Nuclear Safety.

Situation of the national regulations in 2006 with respect to the RLs

The Netherlands system of regulations is based on IAEA standards from the '80 and '90. The

IAEA revised its standards very significantly from 2000. The Netherlands started late in revising its standards accordingly due to possible closure of the single NPP (Borssele). Several more modern safety subjects are not covered in the Dutch regulations: out of the 18 WENRA safety issues six are badly covered. Other safety issues are sometimes very well covered.
Actions started to incorporate RLs in national regulations
In the action plan from 2006 a NVR revision project was announced in line with the latest editions of the IAEA standards, including amendments where necessary. The action plan was limited to the areas of NPP design and operation plus Quality assurance. The planning was to finish the project by the end of 2008/beginning 2009.
Status of the national regulations in September 2010 and envisaged further actions
The following IAEA Requirements and Safety Guides have been considered in the NVR revision project and are now ready for publication by end of 2010: 1. Safety of Nuclear Power Plants, Design. Amended Requirements NS-R-1, 2000 2. Safety of Nuclear Power Plants, Operation. Amended Requirements NS-R-2, 2000 3. Site evaluation for nuclear installations. Amended Requirements NS-R-3, 2003 4. The management system for facilities and activities, Amended Requirements GS-R-3, 2006 5. Software for computer based systems important to safety in nuclear power plants. Amended Safety Guide NS-G-1.1, 2000 6. Safety assessment and verification for nuclear power plants. Amended Safety Guide NS-G-1.2, 2001. 7. Instrumentation and control systems important to safety in nuclear power plants. Amended Safety Guide NS-G-1.3, 2002. 8. Design of fuel handling and storage systems in nuclear power plants. Amended Safety Guide NS-G-1.4, 2003. 9. External events excluding earthquakes in the design of NPPs, Amended Safety Guide NS-G-1.5, 2003. 10. Seismic design and qualification for nuclear power plants. Amended Safety Guide NS-G-1.6, 2003. 11. Protection against internal fires and explosions in the design of NPPs, Amended Safety Guide NS-G-1.7, 2007 12. Design of emergency power systems for NPPs, Amended Safety Guide NS-G-1.8, 2004 13. Design of reactor coolant systems and associated in NPPs, Amended Safety Guide NS-G-1.9, 2004 14. Design of reactor containment systems of NPPs, Amended Safety guide NS-G-1.10, 2004 15. Protection against internal hazards other than fires and explosions in the design of NPPs, Amended Safety Guide NS-G-1.11, 2004 16. Design of the reactor core for NPPs, Amended Safety Guide NS-G-1.12, 2005 17. Radiation protection aspects of the design of NPPs, Amended Safety Guide NS-G-1.13, 2005 18. Fire safety in the operation of nuclear power plants. Amended Safety Guide NS-G-2.1, 2000. 19. Operational limits and conditions and operating procedures for nuclear power plants. Amended Safety Guide NS-G-2.2, 2000. 20. Modifications to nuclear power plants. Amended Safety Guide NS-G-2.3, 2001.

21. The operating organisation for nuclear power plants. Amended Safety Guide NS-G-2.4, 2002.
22. Amended Safety Guide NS-G-2.5, 2002(?)
23. Maintenance, surveillance and In-service inspection of NPPs, Amended Safety Guide NS-G-2.6, 2002
24. Radiation protection and radioactive waste management in the operation of NPPs, Amended Safety Guide NS-G-2.7, 2002
25. Recruitment, qualification and training of personnel for NPPs, Amended Safety Guide NS-G-2.8, 2002
26. Commissioning for NPPs, Amended Safety Guide NS-G-2.9, 2003
27. Periodic safety review of NPPs, Amended Safety Guide NS-G-2.10, 2003
28. A system for the feedback of experience from events in nuclear installations, Amended Safety Guide NS-G-2.11, 2006

Furthermore 22 other safety guides will be adopted (without amendments) by the end of 2010:

a. Application of the management system for facilities and activities. Safety Guide GS-G-3.1, 2006.
b. The management system for nuclear installations. Safety Guide GS-G-3.5, 2009.
c. External human induced events in site evaluation of nuclear power plants. Safety Guide NS-G-3.1, 2002.
d. Evaluation of seismic hazards for nuclear power plants. Safety Guide NS-G-3.3, 2003.
e. Meteorological events in site evaluation for nuclear power plants. Safety Guide NS-G-3.4, 2003.
f. Flood hazards for nuclear power plants on costal and river sites. Safety Guide NS-G-3.5, 2004.
g. Geotechnical aspects of site evaluation and foundations for nuclear power plants. Safety Guide NS-G-3.6, 2005.
h. Deterministic safety analysis for nuclear power plants. Specific Safety Guide SSG-2, 2010.
i. Development and application of level 1 probabilistic safety assessment for nuclear power plants. Specific Safety Guide SSG-3, 2010.
j. Development and application of level 2 probabilistic safety assessment for nuclear power plants. Specific Safety Guide SSG-4, 2010.
k. Safety assessment for facilities and activities. GSR Part 4, 2009.
l. Format and content of the safety analysis report for nuclear power plants. Safety Guide GS-G-4.1, 2004.
m. Ageing management for nuclear power plants. Safety Guide NS-G-2.12, 2009.
n. Evaluation of seismic safety for existing nuclear installations. Safety Guide NS-G-2.13, 2009.
o. Conduct of operation at nuclear power plants. Safety Guide NS-G-2.14, 2008.
p. Severe accident management programmes for nuclear power plants. Safety Guide NS-G-2.15, 2009.
q. Preparedness and response for nuclear or radiological emergency. Safety Requirements GS-R-2, 2002.
r. Arrangements for preparedness for a nuclear or radiological emergency. GS-G-2.1, 2007.
s. Decommissioning of nuclear power plants and research reactors. Safety Guide WS-G-2.1, 1999.

t. Decommissioning of facilities using radioactive material. Safety Requirements WS-R-5, 2006 (later).
u. Decommissioning of nuclear fuel cycle facilities. Safety Guide WS-G-2.4, 2001 (later).
v. Safety assessment for the decommissioning of facilities using radioactive material. Safety Guide WS-G-5.2, 2009 (later).

Country	ROMANIA

General presentation of the regulatory system

The legal basis for the regulation of nuclear safety in Romania is provided by the Law on the Safe Deployment, Regulation, Licensing and Control of Nuclear Activities (Law 111/1996).

CNCAN is empowered by the Law to develop regulations in order to detail the general legal requirements as well as any other regulations necessary to support the licensing and control activities. All the regulations issued by CNCAN are mandatory and enforceable. The regulations are developed in observance of relevant international standards and good practices.

Situation of the national regulations in 2006 with respect to the RLs

The Romanian regulations benchmarked as part of the WENRA harmonisation study are listed below:

- Nuclear Safety Requirements (NSR) - Nuclear Reactors and Nuclear Power Plants (1975), which contains provisions concerning licensing basis documentation, site evaluation criteria and design criteria for NPPs.
- Requirements for prevention and extinction of fires, applicable in the nuclear activities (1976);
- Nuclear Safety Requirements on Emergency Plans, Preparedness and Intervention for Nuclear Accidents and Radiological Emergencies (1993);
- Regulation on granting practice permits to operating, management and specific training personnel of Nuclear Power Plants, Research Reactors and other Nuclear Installations (2004);
- The set of regulations on Quality Management Systems for nuclear installations (NMC series, 2003) which contain provisions related to the quality assurance and safety of operation, maintenance, in-service inspection, testing, modifications, training of personnel, procurement activities, etc.
- Technical Prescriptions for Design, Execution, Assembling, Repair, Verification and Operation of Pipes under Pressure and of Elements of Pipes from Nuclear Plants and Facilities (NC2-83) issued by the State Inspectorate for Boilers, Pressure Vessels and Hoisting Installations (ISCIR).

Since the completion of the benchmarking, CNCAN has published the following regulations:

- Requirements on Containment Systems for CANDU Nuclear Power Plants (2005);
- Requirements on Shutdown Systems for CANDU Nuclear Power Plants (2005);
- Requirements on Emergency Core Cooling Systems for CANDU Nuclear Power Plants (2006);
- Requirements on Fire Protection in Nuclear Power Plants (2006).
- Requirements on Periodic Safety Review for nuclear power plants (2006).
- Requirements on Probabilistic Safety Assessment for nuclear power plants (2006).

Information on the results of the benchmark and on the action plan for endorsing all reference levels and for ensuring their implementation has been provided in the 4th national report of Romania under the Convention on Nuclear Safety.

Actions started to incorporate RLs in national regulations

The updating of the national regulations started immediately after the completion of the

benchmark. Several new regulations have been already issued, while the revision of the regulations available at the time of the benchmarking is in progress. Progress has also been made with regard to the implementation side, for the issues that need to be addressed for harmonisation. The revised action plan (according to the last version of the reference levels) has been included in the national report for the 4th Review Meeting under the Convention on Nuclear Safety. New regulations on siting and on design and construction of NPPs have been issued and the external consultation with the stakeholders has been completed. These regulations have been notified to the European Commission (EC) and it is expected that they will be formally published by the end of the year (2010). The regulation on design and construction of NPPs includes requirements that cover all reference levels in Issues E, F, G and N. The revision of the set of 13 regulations on quality management systems, covering activities related to all the phases of the lifetime of nuclear installations, started in 2007, takes account of the latest IAEA Requirements and Guides on Management Systems (GS-R-3, GS-G-3.1 and GS-G-3.5). The external consultation process for the new regulations has been finalised and they are due to be published by the end of 2010. The new regulations cover all reference levels in Issue C. A regulation on commissioning and operation of NPPs is currently under drafting and will incorporate the remaining RLs. The intention is to have the external consultation with stakeholders completed before the end of 2010. The compliance with the requirements in the reference levels is assessed by the licensees as part of the periodic safety review that is currently ongoing. This assessment has been required by CNCAN.
Status of the national regulations in September 2010 and envisaged further actions
The reference levels in Issues C, E, F, G and N have been covered by new regulations on management systems for nuclear facilities and activities and respectively by new regulations on design and construction of NPPs (due to come into force before the end of 2010), while the remaining RLs will be covered by a regulation on commissioning and operation of NPPs, which is currently under drafting (intended for official publication in late 2010 / early 2011).

Country:	SLOVAKIA
2006 Status in the country	
Based on the national benchmarking there were 85 RLs to be harmonized in total. Act No. 541/2004 Coll. on Peaceful Use of Nuclear Energy (Atomic Act) and a set of 13 regulations were in force in 2006.	
Regulatory system in the country:	
Pursuant to Atomic Act, the supervision of peaceful use of nuclear safety is performed by Nuclear Regulatory Authority (UJD) within its competencies. UJD is a central state administration body ensuring the performance of state regulatory activities in the field of nuclear safety of nuclear installations, including supervision of the management of radioactive waste, spent fuel and other fuel cycle phases, as well as of nuclear materials, including their control and records. Concerning the nuclear safety, the basic legal framework is laid down by the Act No. 541/2004 Coll. on Peaceful Use of Nuclear Energy (Atomic Act). Since 1st December 2004, this new Atomic Act has abrogated former Atomic Act No. 130/1998 Coll. as well as all of 13 regulations issued on the former Atomic Act No. 130/1998 Coll. basis. A new set of regulations that work out the new Atomic Act provisions in detail was accepted and approved by the Slovak Government Legislative Council in August 2005. There are regulation on special materials and equipments, on small quantities of nuclear materials, on details of the notification of events, on periodical safety assessment, on nuclear safety requirements, on the provision for physical protection, on professional qualification, on management of nuclear material, radioactive waste and spent fuel, on safeguards, on emergency planning, on shipment of radioactive materials, on requirements for quality system documentation, as well as details concerning quality requirements for nuclear installations, details concerning quality requirements for classified equipment and on documentation needed for certain decisions The Atomic Act regulates rights and obligations of natural and legal persons in peaceful use of the nuclear energy, nuclear material, radioactive waste, physical protection, shipment of nuclear material, radioactive waste and spent fuel, licensing procedure of the nuclear installations, nuclear safety, emergency planning, quality assurance system, staff training, civil liability for nuclear damage, shut-down of a nuclear installation for other than safety concerns, inspections, sanctions. However, radiation protection is not within the scope of this Atomic Act but remains within the competencies of Public Health Authority subordinated to the Ministry of Health as stated in Act No. 272/1994 Coll. Besides acts and regulations as legally binding, the UJD also formally issues Safety Guides, which contains methods suggested by the UJD to address special topics related to nuclear safety. Safety Guides composes of non-binding provisions but they may be important as criteria within the licensing procedure. The licensing procedure consists of three major stages: siting, construction commencement, and permanent operation. Before granting a license for permanent operation, the regulatory authority carries out control under the approved programs for hot and cold testing and grants approval for fuel loading, physical start up, energy start up and trial operation. The basic condition essential to licensing in terms of nuclear safety is to prepare and submit a Safety Analysis Report and other prescribed safety documentation and to meet the conditions of the regulatory authority's preceding licensing procedures and decisions. Under the nuclear installation licensing procedure, International Atomic Energy Agency standards and recommendations are used and applied. More information on the Slovak legislative and regulatory system can be found on the UJD web site: www.ujd.gov.sk.	
Actions were started to incorporate RLs in national regulations:	
National action plan for WENRA RLs implementation counted on a "one-step" approach, i.e. all RLs to be harmonized would be implemented into various levels of national legal documents	

(atomic act, decrees and safety guides) at once and the set of new revisions would be sent for official approval process as a batch.
<u>Status in 2010</u>:
This "one-step" approach had been followed since 2007, when intensive works on the revision of the atomic act and decrees were launched. In May 2010 the final draft of these documents were finalized and sent for comments to other state ministries and authorities according to Slovak national legal procedure. Based on the national benchmarking there were 85 RLs to be harmonized in total. Part of the RLs was incorporated directly in the **Atomic act** (Act No. 541/2004 Coll. on Peaceful use of nuclear energy and on amendment and alterations of several acts) and the absolute majority of these were incorporated into the following **decrees**: 1. UJD Decree No. 48/2006 Coll. on details of notification of operational events and events during transport, as well as details of investigation of their reasons. 2. UJD Decree No. 49/2006 Coll. on Periodic Assessment of Nuclear Safety. 3. UJD Decree No. 50/2006 Coll. on details concerning the nuclear safety requirements for nuclear installations in respect of their siting, design, construction, commissioning, operation, decommissioning and closure of repository, as well as criteria for categorization of classified equipment into safety classes. 4. UJD Decree No. 52/2006 Coll. on professional competency. 5. UJD Decree No. 55/2006 Coll. on Details in Emergency Planning for the Event of an Incident or an Accident. 6. UJD Decree No. 56/2006 Coll. on details concerning requirements for quality system documentation of authorization holder, as well as details concerning quality requirements for nuclear installations, details concerning quality requirements for classified equipment and details concerning the scope of their approval 7. UJD Decree No. 58/2006 Coll. on Laying Down Details on the Scope, Contents, and Manner of Maintaining Documentation of Nuclear Facilities Necessary for Individual Decisions. The new revision of the Atomic Act has been sent for official legal approval process within the country in August 2010. The new revisions of the above listed regulations are finalised and are expected to be sent for official legal process in the end of September 2010. By amendment of the Atomic Act and the above-mentioned list of regulations all of the RLs will be incorporated into the Slovak legislation.

Country	SLOVENIA
General presentation of the regulatory system	

The present Slovenian legislative and regulatory framework governing nuclear and radiation safety has a long-standing history which has its roots in former Yugoslav legislation. While at the beginning the legislation has focused mostly on the ionising radiation safety (act of 1965 and of 1976) in the 80's it incorporated also all basic provisions related to nuclear safety (act of 1984 and more than 10 regulations).

In the Republic of Slovenia the main act in the area of nuclear and radiation safety is the Act on Ionising Radiation Protection and Nuclear Safety (Off. Gaz. RS, 67/2002 – hereinafter referred to as »2002 Act«). As defined in the first Article of this act, its main purpose is »to regulate ionising radiation protection, with the aim of reducing the detrimental effects on health and reducing to the lowest possible level radioactive contamination of the environment due to ionising radiation resulting from the use of radiation sources, while at the same time enabling the development, production and use of radiation sources and performing radiation practices«

The 2002 Act entered into force on October 1, 2002. From that day two previous Acts ceased to apply, namely:

- Act on Radiation Protection and the Safe Use of Nuclear Energy (1984 Act), and
- Act on Implementing Protection Against Ionising Radiation and Measures on the Safety of Nuclear Facilities (1980 Act).

The 2002 Act was amended in 2003 and 2004. The 2002 Act allows for the regulations issued on the basis of the 1984 and 1980 Acts to apply until new regulations, which are to be adopted pursuant to provisions of the 2002 Act, are issued. Based on the 1984 Act only a part of one regulation is still in force.

Based on the 2002 Act 7 governmental decrees and 21 ministerial regulations (ten regulations issued by the minister of the environment, nine issued by the minister of health and two issued by the minister of the interior) were adopted and issued until 2010. All these regulations are legally binding.

A detailed list of the already adopted implementing regulations and those under preparation can be found at the SNSA web page http://www.ursjv.gov.si, but is not yet fully available in the English translation.

The comprehensive legislative and regulatory framework, which governs the areas related to nuclear and radiation safety, consists of the national legal frame and of those international instruments (multilateral and bilateral treaties, conventions, agreements/arrangements) to which Slovenia is a party.

Besides the main principles (among others also "primary responsibility for safety", "the causer-pays principle", "justification", "optimisation", "ALARA" and "the preparedness principle") the 2002 Act includes, with respect to nuclear and radiation safety area, also provisions on:

- reporting an intention to carry out radiation practices or to use radiation source;
- licensing of the radiation practice or use of radiation source;
- classification of facilities (nuclear, radiation and less important radiation facilities);
- licensing procedures with respect to siting, construction, trial operation, operation and decommissioning of nuclear, radiation and less important radiation facilities;
- radioactive contamination and intervention measures;
- radioactive waste and spent fuel management;
- import, export and transit of nuclear and radioactive materials and radioactive waste and spent fuel;
- physical protection of nuclear materials and facilities;
- non-proliferation and safeguards;

- administrative tasks and inspection;
- penal provisions.

It also includes provision on competent regulatory body. In nuclear and radiation safety the competencies are divided among two regulatory bodies, namely the Slovenian Nuclear Safety Administration (SNSA) which is accountable for nuclear safety and safety of industrial radiation sources and Slovenian Radiation Protection Administration (SRPA), accountable for radiation protection of patients, medical surveillance of exposed workers, surveillance of workplaces, dosimetry and dose registers and education in the area of radiation protection.

In the licensing process, the key document governing the technical and safety measures for the construction and operation of the nuclear facility is the Safety Analysis Report (SAR).

Further information on Regulatory body and legislative framework can be found at the same above mentioned web site.

Situation of the national regulations in 2006 with respect to the RLs

The act on nuclear safety (ZVISJV) has been already developed and adopted in Slovenian legal system. Old regulations, developed under the old Nuclear Act, were still on power except the regulation for operator training and licensing, which was already adopted (2005). This was the legal basis against which the benchmark exercise performed and Action Plan developed in 2006.

Some works have been dedicated to development of remaining new regulation, but in that time drafts were in very early stage.

Actions started to incorporate RLs in national regulations

The renovation of the national legal system after adopting the "2002 Act" went on by issuing the number of regulations.

Among them in the period 2006-2010 the following two new regulations were prepared and finally adopted at the end of 2009:

- Rules on radiation and nuclear safety factors (JV5) and
- Rules on operational safety of radiation and nuclear facilities (JV9)

These regulations cover all WENRA Reference Levels, except Reference level D –Training.

It should be mentioned that in these new regulations various grace periods for implementation of some of new requirements are foreseen.

The domain of WENRA Ref. Level D is part of the Regulation JV4 which was adopted during 2005, before the WENRA issued its reference levels. According to the WENRA benchmark exercises 5 of 10 WENRA requirements are not covered by the existing JV4 regulation. To overcome this SNSA have prepared the updated version of Regulation JV4 which includes the corresponding WENRA requirements. The new, updated version of Regulation JV4 are now going through the process of adoption. It is expected that it will be approved by the end of year 2010.

The table of concordance between the WENRA reference levels and Slovenian regulations are available on SNSA site:

http://www.ursjv.gov.si/fileadmin/ujv.gov.si/pageuploads/si/Porocila/Primerjava_WENRA_ZVISJV_JV5_JV9.pdf

English translation will be provided in the first months of 2011.

Status of the national regulations in September 2010 and envisaged further actions
Practically all WENRA requirements are included in the domestic safety regulation which is on power and in use. The 5 requirements related to Training (issue D RLs) will be incorporated in the appropriate regulation during the year 2011.

Country	SPAIN
General presentation of the regulatory system	
The nuclear regulatory framework rests on different laws and regulations such as the Nuclear Energy Act (Law 25/1964) as amended, the Law on the creation of the Nuclear Safety Council (Law 15/1980) as amended by law 33/2007 and the Electricity Industry Law (Law 54/1997). These set of laws define the nuclear regulatory framework establishing general safety principles or criteria, the processes applicable to obtain the necessary authorisations, and the mechanism for the regulatory inspections and control. Basic principles determine that the responsibilities derived from the usage of nuclear energy rests in the licensee of the installation. The Nuclear Safety Council (CSN) is the sole competent Authority for Nuclear Safety and Radiation Protection, independent from the Government and in charge of performing the regulatory inspections and control and supervision of nuclear and radioactive installations. The CSN reports to the Parliament. Electricity Industry law regulate the operation of the electricity industry and is applicable in certain areas to the nuclear industry. Law 33/2007 has amended the CSN creation law 15/1980. This amendment updates the legal framework for faults and penalties, and assigns to the CSN a stronger role in the enforcement procedure among other things. The Government issue decrees to complete and develop the requirements established by laws. The following decrees are the most significant: Royal Decree 1836/1999 as amended in 2008. Regulation on Nuclear and Radioactive Installations: this regulation establishes the licensing system for sitting, construction, commissioning, operation and decommissioning. Royal Decree 783/2001 Regulation on protection of public and workers against the risks of ionising radiations (revision 2001): it includes the basic criteria and measures for radiation protection, as established in the Directive 96/29 issued by the EURATOM board in this matter. Decree governing the coverage of nuclear risks (1967), as amended: it develops the Nuclear Energy Act in the field of the responsibility of the licensee, establishing the system for coverage for civil liability derived from such responsibility. Royal Decree 413/1997 governing the occupational protection of outside workers potentially exposed to ionising radiation due to their intervention in the controlled zone (1997): this regulation transposes the contents of EURATOM Directive 90/641, which regulates the obligations of the operator, the outside undertakings and the outside workers, in order to assure the protection of the outside workers intervening in the controlled zone of nuclear installations. Royal Decree 1546/2004 approving Basic Nuclear Emergency Plan, as amended by Royal Decree 1428/2009: it defines the co-ordinated action of the different Public Organisations in case of a nuclear accident. It defines the emergency plans for each province having a nuclear installation. A Ministerial Order of the Ministry of Industry, Tourism and Trade, issues the authorisation for operation (license) to each NPP. The CSN report is the base for the nuclear safety and radiation protection issues of the authorisation. The authorisation is valid (usually) for a period of 10 years and includes the appropriate limits and conditions under which the plant must operate. These limits and conditions are legally binding. The licensing documents (such as safety analysis report, technical specifications for operation, operations requirements, dose calculations manual, emergency plan, etc.) also referred to in the Royal Decree 1836/1999 and in each authorisation are legally binding documents for the licensee. The regulatory framework is such that the CSN is empowered to issue instructions, technical complementary instructions, circulars and safety guides. **CSN Instructions** The CSN Instructions (with the same legal status than governmental regulations) and Complementary Technical Instructions are both legally binding. The Instructions are technical	

standards on nuclear safety or radiation protection, directed to all installations. The CSN Instructions are published in the National Official Gazette.

The Complementary Technical Instructions usually develops a license condition established within the authorisation for operation to each licensee. They are directed specifically to each licensee.

CSN Safety Guides

The CSN Safety Guides containg methods suggested by the CSN to address special topics related to nuclear safety and radiation protection. These guides are not binding in a prescriptive way unless endorsed by the license. The user may apply methods and solutions different from those contained in the guides, as long as they are duly justified.

The Safety Guides covers the main areas of responsibility of CSN, such as nuclear power plants, research reactors, fuel cycle installations, environmental radiological surveillance, radioactive installations and equipment, transport of radioactive materials, radiation protection, security and waste management.

Situation of the national regulations in 2006 with respect to the RLs

As stated in the 2006 RHWG report “Harmonization of Reactor Safety in WENRA countries”, the harmonisation of the RLs is partially claimed through the legislation such as the Laws and the Royal Decree 1836/1999 on the regulation of nuclear and radioactive installations, the licence for operation of each plant and the limits and conditions included thereon.

As of 2006, the CSN instructions published so far were mainly in relation to radiation protection issues. For nuclear safety issues, the CSN had elaborate almost no legally binding requirements as defined in the 2006 RHWG report. There were only CSN orders and/or letters to the licensees requesting the compliance with different issues.

The results for Spain of the 2006 RHWG report highlighted that for a significant number of RLs (>150) there was a need to issue regulations or CSN Instructions to reach harmonisation (C categories) as far as the regulatory requirements were concerned.

Actions started to incorporate RLs in national regulations

Based on the results of the RHWG report and in accordance with the commitment established in WENRA the CSN set up, in 2006, an action plan for the development of CSN instructions and guides in order to be harmonised in 2010.

The action plan contemplates that fifteen CSN Instructions were going to be developed and that one existing CSN Instruction (IS-10) on notification of events need to be updated. In addition, 3 existing CSN Safety Guides were subject to revision and a new one was going to be elaborate.

Status of the national regulations in September 2010 and envisaged further actions

The CSN action plan progress adequately. As of September 2010 of the fifteen new instructions envisaged, the CSN has already developed, approved and published in the Official National Gazette ten of them; three other instructions are under consultation/decision making process and two instructions are in drafting. One is under revision for updating.

As regard to the safety guides the one to be review is complete and publish.

The titles of the ten CSN Instructions already published follows:

- Instruction IS-11 on Training and qualification for NPP control room staff. Published in Official State Gazette Number 100, dated 26 of April 2007

- Instruction IS-12 on defining the qualification and training requirements of non-licensed staff and non-licensed off-site personnel of nuclear power plants. Published in Official State Gazette Number 113, dated 11 of May 2007
- Instruction IS-15 on the requirements for the surveillance of the maintenance efficiency. Published in the Official State Gazette Number 281; dated 23 November 2007.
- Instruction IS-19 on the requirements for the NPP management system. Published in the Official State Gazette Number 270; dated 8 of November 2008.
- Instruction IS-21, on the requirements applicable to plant design modifications. Published in the Official State Gazette Number 43; dated 19 of February 2009.
- Instruction IS-22, on the requirements for management of aging and the plant operation beyond design. Published in the Official State Gazette Number 166; dated 10 of July 2009
- Instruction IS-23 on the in-service inspection of nuclear power plants. Published in the Official State Gazette Number 283; dated 24 of November 2009.
- Instruction IS-25 on the requirements and criteria for the probabilistic safety assessments and its applications on NPPs. Published in the Official State Gazette Number 153; dated 24 of June 2010.
- Instruction IS-26 on the basic safety requirements applicable to nuclear installations. Published in the Official State Gazette Number 165; dated 8 of July 2010.
- Instruction IS-27 on the general design criteria for NPPs. Published in the Official State Gazette Number 165; dated 8 of July 2010.

One safety guide is already updated:

- GS.1.10 periodic safety reviews of NPPs.

<u>State of development of Instructions and Guides under drafting and or consultation/decision making process</u>:

CSN instructions:

- 3 new CSN Instructions are under consultation/decision making process (on technical specifications for operation, emergency operating procedures, and on fire protection);
- 2 new CSN Instructions are in drafting (on operating experience and on accident analyses);
- In addition, 1 existing CSN Instruction (IS-10) on notification of events is under revision. It is in consultation phase.

CSN Safety guides:

- Revision of GS 1.6 on “notification of events to the CSN” is in drafting phase;
- Revision of GS 1.1 on “training of licensed personnel” is in consultation phase;
- In addition, a new safety guide on fire protection is been elaborated and is in consultation phase.

The publication of the three instructions and the safety guide that are in the consultation phase will be soon. For those instructions and safety guides that are in drafting phase its publication will take longer and might take place at the beginning of year 2011.

Country	**SWEDEN**
General presentation of the regulatory system	
Since 1 July 2008 Sweden has a new integrated regulatory body for nuclear safety and radiation protection; the Swedish Radiation Safety Authority (SSM). SSM is a merger of the two earlier regulatory bodies; the Swedish Nuclear Power Inspectorate (SKI) and the Radiation Protection Authority (SSI). SSM has taken over all the missions and tasks of the two earlier authorities. After the formation of SSM, a new series of regulations SSMFS was created. In this series all regulations formerly issued by SKI and SSI have been re-issued as SSM regulations. There are now 53 such regulations in force from 1 February 2009 (see: www.ssm.se). Also as a consequence of the formation of SSM, the Government has appointed a special investigator to review and propose changes in the nuclear legislation and possibly introduce the concept of Radiation Safety into the legislation. Decisions about this are expected earliest 2011. The following five Acts constitute the basic nuclear legislation of Sweden: - The Act (1984:3) on Nuclear Activities, - The Radiation Protection Act (1988:220), - The Environmental Code (1998:808), - The Act (2006:647) on Financing of the Management of Residual Products from Nuclear Activities, - The Nuclear Liability Act (1968:45). With exception for the Nuclear Liability Act, all Acts are supplemented by ordinances and other secondary legislation which contain more detailed provisions for particular aspects of the nuclear safety and radiation protection regime. General obligations in cases of accidents which can threaten life and the environment are included in the Act (2003:778) on Protection against Accidents and The Ordinance (2003:789) on Protection against Accidents. The following former SKI regulations, now SSM regulations, were referred to in the reactor harmonisation study. General Recommendations on how to interpret the regulations have been issued in direct connection to the regulations and are included in the respective SSMFS publication. The licensees have to follow these recommendations or take other measures which are justified to be equal from the safety point of view. **Regulations and General Recommendations concerning Safety in Nuclear Facilities (SSMFS 2008:1)**: Basic requirements on design, safety management, physical protection, emergency preparedness, assessment and reporting of safety, operations and maintenance, management of nuclear materials and waste, and decommissioning. **Regulations and General Recommendations concerning Mechanical Components in certain Nuclear Facilities (SSMFS 2008:13):** Requirements on measures, control- and inspection activities on mechanical components to be taken during plant modifications, maintenance and in-service inspections. **Regulations and General Recommendations concerning Design and Construction of Nuclear Power Reactors (SKIFS 2008:17):** Requirements on design principles, withstanding of	

failures, conditions and events, and requirements on the design and operation of the reactor core. **Regulations and General Recommendations concerning the competence of Operations Personnel at Reactor Facilities (SSMFS 2008:32):** Requirements on competence analysis, training and authorisation as well as requirements on simulators for operational training.
Situation of the national regulations in 2006 with respect to the RLs
During the reactor harmonisation study in the years 2003-2006 much work was done at SKI to revise and develop regulations. Several of the WENRA reference levels under discussion and benchmarking were entered into the general safety regulations and regulations on design and construction which came into force 1 January 2005. This meant that there were only a limited number of C-differences remaining from the benchmarks to be dealt with. The situation on the legal side 2006 showed a total number of 38 C-differences distributed over issues C (7), D (1), E (3), F (5), H (1), J (2), K (3), LM (6), O (3), P (1) and S (6). It should be noted that a number of reference levels were clarified and updated during 2007 as a result of the consultations with stakeholders and the new IAEA Safety Requirements GS-R-3. This mostly affected safety issues C, E and F. New benchmarks were made against these levels. The finally revised set of reference levels was published in January 2008.
Actions started to incorporate RLs in national regulations
Most of the C-differences remaining in the end of 2007 have been closed by a revision of SSMFS 2008:1 prepared during 2009. This revision will require a few other changes in the regulations after the Parliament decision of a change in the Act on Nuclear Activities performed in June 2010. However, these changes have nothing to do with the reference levels. The revision of SSMFS 2008:1 is expected to come into force 2011. A more exact date cannot be predicted for the moment. Work to revise SSMFS 2008:17 is planned to begin 2011 and is expected to be finalised in the end of 2012.
Status of the national regulations in September 2010 and envisaged further actions
For the moment there are 12 remaining C-differences to deal with. Six are connected to issue E, one to issue K and five to issue S. The remaining 12 C-differences on the legal side will be dealt with in a planned revision of SSM´s regulations on design and construction of power reactors (SSMFS 2008:17). This revision is depending on consultations with other Swedish authorities and some ongoing technical investigations of the bases for making some other changes to these regulations. In addition, the planned revision will also be affected by the mentioned Parliament decision to change the Act of Nuclear Activities. One part of this bill, which will come to effect January 1, 211, is to make it legal to replace the 10 existing reactors with new ones; given that there will be no governmental changes after the election to parliament in late September 2010. With the new bill SSM have to decide whether to update SSMFS 2008:17 to apply also on new reactors. In conclusion, Sweden is not fully able to satisfy the WENRA agreement to align the national safety regulations with all the reference levels by the end of 2010. To a certain extent this has to do with circumstances without SSM´s control such as the merger of SKI and SSI and the decided

changes in the nuclear law. However, it can also be concluded that only very few C-differences remain to be handled, such as some assumptions for the deterministic safety analysis and rules about fire protection systems and equipment. Such rules were earlier issued by another Swedish authority.

Country	SWITZERLAND

General presentation of the regulatory system

The legislation for the use of nuclear energy and on radiological protection is enacted exclusively at the federal (national) level (art. 90 and 118 Federal Constitution). The main provisions for authorisations and regulation, supervision and inspections are established in the Nuclear Energy Act (Kernenergiegesetz, 2005) and the Radiological Protection Act (Strahlenschutzgesetz, 1994). The legal rules and principles are put in concrete terms in the Nuclear Energy Ordinance (Kernenergieverordnung, 2005), Radiological Protection Ordinance (Strahlenschutzverordnung, 1994) and in about 10 further ordinances. The main basis for implementation and enforcement are the Guidelines of the Swiss Federal Nuclear Safety Inspectorate (ENSI).

More information about Swiss regulation can be obtained at: http://www.ensi.ch

Situation of the national regulations in 2006 with respect to the RLs

In 2006, many WENRA reference levels were not covered by the Swiss regulation.

At that time, the enactment of the new Nuclear Energy Act (2005) called for a "rewriting" of all ordinances and guidelines. This was a good opportunity to implement all the WENRA Reference Levels.

Actions started to incorporate RLs in national regulations

Since 2005, about 10 new or fundamentally revised ordinances have been enacted. Concerning the level of Guidelines the output of new regulations was even more extensive: 2 new ENSI-guidelines were published in 2007, 6 in 2008, 7 in 2009 and 5 in 2010. The process is still underway. This intensive phase of guideline drafting falls within the period where the authorities have to examine three applications for a general licence for new nuclear power plants.

Working out the new guidelines ENSI takes into consideration the WENRA Reference Levels (cf. Memo "Grundlagen der Aufsicht, AAU1192"). There is an explanatory report to each guideline; in this report it has to be demonstrated that the relevant WENRA Reference Levels are implemented in the guideline. If not, this has to be justified.

Status of the national regulations in September 2010 and envisaged further actions

Most of the WENRA reference levels are covered by the Swiss regulation.

New Ordinances (applicable to NPP):

No. (SR)	Title
732.112.2	Ordinance on Hazard Assumptions and Evaluation of Protection Measures against Accidents in Nuclear Installations.
732.114.5	Ordinance on Methodology and Boundary Conditions for Evaluation of Criteria for provisional Taking out of Service of Nuclear Power Plants
732.13	Ordinance on safety-classified Vessels and Piping in Nuclear Installations
732.134.1	Ordinance on Qualifications of Personnel in Nuclear Installations

New Guidelines (applicable to NPP):

No.	Title
G07	Organisation of nuclear installations
G15	Radiation protection objectives for nuclear installations in normal operation
G11	Safety-classified vessels and piping: Planning, manufacturing and installation

G13	Radiation protection measuring devices in nuclear installations: Concepts, requirements and testing
G14	Calculation of radiation exposure in the vicinity due to emission of radioactive substances from nuclear installations
A01	Requirements for deterministic accident analysis for nuclear installations: Scope, methodology and boundary conditions for technical accident analysis
A04	Application documents for modifications in nuclear power plants requiring a permit
A05	Probabilistic Safety Analysis (PSA): Quality and Scope
A06	Probabilistic Safety Analysis (PSA): Applications
A08	Analysis of source terms: Extent, methodology and boundary conditions
B02	Periodical reporting for nuclear installations
B03	Reports for nuclear installations
B04	Clearance of materials and areas from controlled zones
B06	Safety-related classified vessels and piping: maintenance
B07	Safety-related classified vessels and piping: Qualification of non-destructive testing
B11	Emergency excercises
B12	Emergency preparedness in nuclear installations

All ordinances based on the new Nuclear Energy Act entered are in force. Several guidelines are currently being drafted. The process of guideline drafting should be completed by the end of 2013.

Country	UNITED KINGDOM

General presentation of the regulatory system

The operators of nuclear plants in the UK must, like their counterparts in other industries, conform to the Health and Safety at Work etc Act 1974 (HSW Act). The HSW Act is goal setting in nature and places a fundamental duty on employers to ensure, so far as is reasonably practicable, the health, safety, and welfare at work of all their employees. It also imposes a duty to ensure that members of the public are not exposed to risks to their health or safety because of the activities undertaken. The Health and Safety Executive (HSE), which is the parent body for the Nuclear Installations Inspectorate (NII), enforces the HSW Act.

The Nuclear Installations Act 1965 (as amended) (NI Act) augments the HSW Act, preventing nuclear plants being installed or operated on a site until the HSE has granted a nuclear site licence to a corporate body. A licence is not transferable but a new licence may be granted to another corporate body, subject to the same evaluation process as for an initial licence.

Each licence contains a standard set of 36 licence conditions [(1)] for all plants to provide consistent safety requirements. They are phrased in general terms that make the licensee responsible for developing and applying detailed safety standards and procedures for the plant. Thus, each licensee can adopt arrangements that best suit their business, so long as safety is being properly managed. When considering a licence application, HSE scrutinises the suitability of the proposed organisation and location together with the hazards and risks associated with the proposed activities.

The licensee is responsible for the safety of their plant and must provide NII with a written demonstration of safety. This is known as the 'safety case': this covers all stages in the life of the plant from construction through to decommissioning and must be updated to reflect changing conditions. Under the NI Act, all significant safety-related activities need some form of permission from NII. This 'permissioning regime' prevents licensees from substantially modifying plant or altering operating arrangements without NII involvement. Assessment is the process by which the NII, on behalf of HSE, establishes whether the safety case is adequate and the Safety Assessment Principles (SAPs) are used for that purpose. These principles are published in a public document. NII also has other documents, such as Technical Assessment Guides (TAGs), Technical Inspections Guides (TIGs), and other specific guidance, that have been published or are being added progressively to its web site that inform licensees and the public about how NII assesses licensees' proposals and the requirements that need to be met for permission to be granted.

NII exercises control through primary powers provided by the licence conditions (as explained more fully in reference [(1)], NII may also exercise control through 'derived' powers when licensee's arrangements provide mechanisms for this). Finally, NII inspectors may also use their enforcement powers under the HSW Act to issue Prohibition and Improvement Notices and to prosecute for breaches of that Act. Breaches of licence conditions are offences under the HSW Act.

More information can be found on NII's web site: (http://www.hse.gov.uk/nuclear/index.htm).

Situation of the national regulations in 2006 with respect to the RLs

For the UK, coverage of the WENRA reactor safety reference levels (RLs) was being claimed through high level legislation such as HSW Act and NI Act, as well as SAPs (1992 version), TAGs and TIGs and the UK action plan at this time called on a variety of such sources to claim whole or partial harmonization. The action plan, however, identified that for a significant number of RLs (>120) there was a short fall that needed to be addressed to reach harmonisation (C categories) as far as the regulatory requirements were concerned.

Actions started to incorporate RLs in national regulations

Revision and reissue of the SAPs in 2006 [(2)], together with the planned upgrade of the TAGs provided a good opportunity to address the short falls identified in the Action Plan in a coordinated way.

During the revision of Technical Assessment Guide T/AST/005 (TAG 005) [(3)], the document was amended to formally adopt the RLs as Relevant Good Practice (RGP) as defined by HSE solicitors[(4)]. HSE's published enforcement policy [(5)] requires that RGP is met, hence failure to do so could lead to enforcement action. The fact that the RLs are in English is helpful to the UK as they don't have to be re-written; they can be (and are) used as they stand. In addition to including RLs as requirements in the TAG 005 we have a programme of TAG revisions and there is an internal requirement on authors to include those RLs relevant to the technical area. This latter is seen as an aid to clarity and our internal processes rather than necessary to meet the obligation to have the RLs as national requirements. TAG 005 has been through internal review and acceptance and published. In summary:

- The RLs have been formally adopted as national requirements within TAG 005.
- TAG 005 is part of our legal regulatory system; it details how HSW Act 1974 sections 2 and 3 are to be applied in the Nuclear Industry.
- TAG 005 is an official, open publication; it has been formally issued.
- Legal sanction can be applied to enforce compliance by licensees.

Hence the UK approach is judged to meet RHWG's stipulation regarding national requirements.

Status of the national regulations in September 2010 and envisaged further actions

The RLs are considered to be fully incorporated into the UK national requirements and no further actions are envisaged.

References:

(1) Nuclear Site Licence Conditions. (http://www.hse.gov.uk/nuclear/silicon.pdf)

(2) Safety Assessment Principles for Nuclear Facilities. (http://www.hse.gov.uk/nuclear/saps/saps2006.pdf)

(3) ND Guidance on the Demonstration of ALARP (As Low as Reasonably Practicable). (http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast005.htm)

(4) Assessing Compliance With the Law in Individual Cases and the Use of Good Practice. (http://www.hse.gov.uk/risk/theory/alarp2.htm)

(5) Enforcement Policy Statement. (http://www.hse.gov.uk/pubns/hse41.pdf)

Appendix 3

Implementation of the RLs on the nuclear power plants

Country	BELGIUM

Situation in 2006 with respect to the implementation of the RLs on the NPPs

After the publication of the WENRA RHWG report « Harmonization of Reactor Safety in WENRA Countries" (January 2006), each country was expected to develop an action plan to bring the status of the NPPs in conformity with the WENRA RLs.

For the Belgian NPPs, about 35 RLs had been scored a "C" in the benchmarking exercise and hence all these RLs were covered in the Belgian Action Plan. This Action plan was presented to WENRA in November 2006 and published on the FANC website (see http://www.fanc.fgov.be/GED/00000000/000/29.pdf).

The actions covered different safety issues, although the most important actions (concerning work effort) were related to PSA (Issue O) and Fire protection (Issue S).

Progress made up to September 2010 and envisaged further actions

A formal structure for the follow-up of the implementation of the actions plan has been put in place in 2007. This "WENRA consultative committee" is composed of managers and senior experts from the FANC and Bel V on the regulatory side and from Electrabel and its engineering office Tractebel Engineering on the operator side.

The Terms of references of this committee precise among others the role of this committee:

MANDATE and OBJECTIVES:

The consultative Committee brings together the regulator (AFCN/FANC), the authorized inspection organization (AVN) *[note: now Bel V]*, the licensee (Electrabel) and its engineering support (Tractebel Engineering) in an effort:

- To continue to achieve a common understanding of WENRA Reference Levels on Reactor Safety, integrating the possible amendments made by WENRA
- To update the self-assessment on the implementation status
- To adjust the existing implementation action plan in order to take into account the WENRA amendments
- To follow-up the implementation of the action plan (implementation side).
- On request of the FANC, to comment the FANC proposals for the legal side.

The activities of the Committee are without prejudice to the roles and responsibilities of the respective organisations participating to it.

Its deliverables will take the form of recommendations.

Once the above mentioned objectives will be achieved, it will be decided whether it is useful to extend the mandate of the Committee. If not, the Committee will be automatically dismissed

DELIVRABLES

The deliverables of the Committee are:

- The meeting reports including:
  - The identification of the tasks and the work process needed to meet the objectives
  - The issue of records of clarifications, positions and statements

- As appropriate, follow-up documents

This committee meets every three months. During the meetings, a review of the ongoing actions is realized. The timing of each action is confirmed or modified, and the action plan schedule is updated accordingly. Formal closure of actions is proposed by operator to the Regulatory body. The Regulatory Body, composed of the FANC and Bel V, taking also into account the follow up of Bel V (verifying the implementation at the 2 plant sites and in the 7 NPPs) , approves the closure of the actions and acts this closure in the meeting report.

<u>Present status of the Action Plan</u>

At present (September 2010) 15 actions have been declared closed. These concern the following Reference Levels:

- A.1.5
- D.1.2, 3.1, and 3.2
- H.7.1
- J.1.2
- L+M.6.1, 6.2, and 6.3
- N.2.8
- O.3.5
- P.2.2
- Q.1.2 and 5.3
- R.2.3

For all other Reference Levels, the actions are on-going. For some actions that require the highest manpower effort (on PSA and Fire protection) the planning runs until 2015.

Country	BULGARIA
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
After the revision of the reference levels in 2007 and the new benchmarks, about 30 actions to address differences of types 'C' and 'B' had been included in the National Action Plan on the implementation side. The B-differences relate to measures that were under implementation at the time of the benchmarking and have been subsequently completed. Among the identified differences of type 'C', the prevailing areas are connected with implementation of an integrated management system (issue C), implementation of accident management measures (issue F), implementation of symptom based EOPs and SAMGs (issue LM) and extension of the scope of PSA levels 1 and 2 (issue O).	
Progress made up to September 2010 and envisaged further actions	
A process of implementation of all actions of the National Action plan has been established. Due to the volume and the complexity of some measures however, their implementation will not be completed by the end of 2010. The measures scheduled beyond 2010 relate to the following: - implementation of an integrated management system; - verification, validation and implementation of SAMGs; - prevention of early containment bypass; - equipment qualification; - updating and extension of the scope of PSA level 2; - risk-informed optimisation of in-service inspection programs. Specific projects and programs for the implementation of these measures have been elaborated and are currently under way.	

Country	CZECH REPUBLIC

Situation in 2006 with respect to the implementation of the RLs on the NPPs

The results of Czech Nuclear Power Plants assessment in the 2006 RHWG WENRA report identified some reference levels that were not implemented. In 2007 RHWG agreed to perform a revision of the matrixes. After this revision (self assessment of the benchmark results) the codifications in some reference levels was changed and consequently the new actions were required or monitored on the implementation side in the following issues:

- D - Training
- F - Design Extension of Existing Reactors
- I - Ageing
- J - Operational Experience Feedback
- K - Maintenance
- L+M - Emergency Operating Procedures and Severe Accident Management Guidelines
- O - Probabilistic Safety Analysis

Progress made up to September 2010 and envisaged further actions

Situation of harmonisation on the Dukovany NPP was completely checked by licensee and SUJB during last PSR finished in 2007, corrective actions were checked in 2009 and 2010.

On Temelin NPP, which is in commercial operation from 2004, the PSR was finished in 2010 and the results were handed over to the SUJB for review. The set of corrective actions is the part of the PSR Reports. Simultaneously, the situation on Temelin NPP is well known to SUJB from licensing processes.

Because of the WENRA Reference levels were and will be used as reference requirements for ongoing and future PSRs, the licensee applied it in the review and the corrective actions are planned also according to this factor. The limiting factor for complete harmonisation is a long term planned process of modifications, structured according safety importance of issues and also according real plan of refuelling outages.

The modifications, focused to severe accident management are also limited by problems of low knowledge of the phenomena, influencing plant behaviour in accident conditions and causing difficulties with fixing of final design solution of the problem. Actually 5 of 16 identified RLs were completely unsolved, 6 were solved by realisation of corrective actions, for last 5 the licensee fixed technical solutions and financial resources but he waits for optimal situation in the plan of outages.

Country	**FINLAND**
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
STUK requested the Finnish NPP licensees to assess the implementation of the reactor safety reference levels in 2006. Based on the licensees' assessments and STUK's own assessment, it was concluded that on the implementation side, Finland had only one reference level which was considered not implemented at the Finnish nuclear power plants. This reference level was related to the evaluation of the licensee's organisational changes (RL B1.2).	
Progress made up to September 2010 and envisaged further actions	
The implementation of the reference level B1.2 was verified in connection with enforcement hearings of the revised STUK's Guide YVL 1.4 "Management Systems for Nuclear Facilities" in 2008. The process concerning applying new regulatory guides to existing nuclear facilities in Finland is described in Appendix 1. The current situation is that all the WENRA reference levels are now implemented at the Finnish NPPs.	

Country	FRANCE
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
At the beginning of 2006, although few formal regulations were set to govern the design and operation of French nuclear power plants, most of the RLs were actually implemented. Among the roughly 300 RLs, about 15 needed improvements on their implementation : - they were mainly concerning topic E (verification of the design), O (probabilistic safety analysis) and S (protection against internal fire); - a few were related to topic B (operating organisation), F (design basis envelope), H (operational limit and conditions), N (safety analysis report) and Q (plant modification).	
Progress made up to September 2010 and envisaged further actions	
Mid 2007, EDF completed a self-assessment on the implementation of WENRA RLs. Mid-2010, at ASN's request, EDF completed a second self-assessment. In 2010, the main improvements implemented compared to 2006 are stated below. As for the RLs on issue F (design extension of existing reactors, formerly included in E issue) RLs : - F4.4 (E5.6) passive autocatalytic hydrogen recombiners have been installed on all French NPP; - F4.6 (E5.7) concerning the prevention of high pressure core melt, a modification is being implemented to enhance the reliability of the command of the pressurizor relief valves on 900/1300 MWe series within VD3 upgrades and on N4 series within VD1 upgrades; - F4.7 (E5.9) As part of 900 MWe series VD3 and N4 series VD1, EDF is implementing a device to detect if molten core escapes the vessel to implement appropriate actions stated in its severe accident management guidance. Specifically for Fessenheim NPP, concerning the prevention of containment melt through, ASN directed EDF to increase the depth of basemat before its 40th anniversary PSR. As for the RLs on issue O (probabilistic safety analysis): - For N4 series, PSA level 1 was extended to cover beyond design basis accidents as part of VD1 PSR. However, the level 2 PSA is still missing; - For 1300 MWe series, a level 2 PSA has been completed (a level 2 PSA was already available for 900 MWe series); - For 1300 MWe series, in the process of their VD3-PSR, EDF has nearly completed one PSA dedicated to fire and one to internal flooding. For St Alban NPP, EDF has established a seismic PSA; - There is some use of PSA to identify the need for plant modification. An example is their use in the 900 MWe series VD3 PSR and in 1300 MWe series VD3 PSR process. As for the RLs on issue Q (plant modifications) : - Q5.4 : a systematic review of temporary plant modification is now performed at NPP and EDF initiated in 2010 an overall action plan to progressively reduce the number of such modifications. As for the RLs on issue S (protection against internal fire): - Fire hazard analysis, developed on deterministic basis, have been performed by EDF and	

where sent to ASN for review late in 2009 (application of ministerial order December 31, 1999);

- Probabilistic fire analysis is being performed for 1300 MWe plants as part of their VD3-PSR;
- For 900 MWe plants, modifications related to the level 1 PSA are being implemented as a result of their third PSR.

For the other topics related to issues B, F, H, N and Q, they were mostly related to the update of regulations (see relevant appendix).

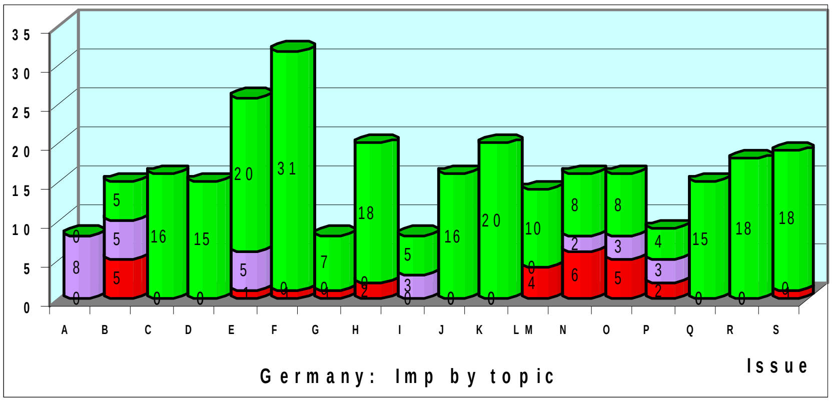
Overall, compared to 2006, implementation of WENRA RLs for existing reactors has improved in France but is still not yet fully completed.

Country	GERMANY

Situation in 2006 with respect to the implementation of the RLs on the NPPs

On the implementation side for Germany 28 RL were assessed with a “C”.

The national graph is shown below. Main areas were issues B “Organisational structure” (5), E “Design Basis Envelope for Existing Reactors” (1), F “Design Extension of Existing Reactors” (1), G “Safety Classification” (1), H “Operational Limits and Conditions” (2), LM “Emergency Operating Procedures and Severe Accident Management Guidelines” (4), N “Contents and updating of Safety Analysis Report” (6). O “Probabilistic Safety Analysis” (5), P “Periodic Safety Review” (2) and S “Protection against Internal Fires” (1).



Progress made up to September 2010 and envisaged further actions

Progress in Issue B: An assessment of the existing documentation and approaches with regard to requirements of the reference levels of Issue B has been carried out. The assessment showed that harmonisation is reached.

Progress in Issue E: The reference level has been changed so that beyond design basis accidents are not longer addressed. Thus the WENRA categorization “C” from 2005 is no more valid. Based on the German safety standards a set of plant specific initiating events, as well as technical acceptance criteria are defined. Initiating events are grouped into a limited number of categories. However, initiating events are not explicitly grouped according to their probability of occurrence. Radiological acceptance criteria are assigned in accordance with the radioprotection ordinance for normal operation, anticipated operational occurrences and design basis accidents. The assessment showed that harmonisation is reached with tolerable differences.

Progress in Issue F: The issue has been completely rewritten. The 2005 categorization “C” for one reference level is no longer valid. The assessment of the changed reference level showed that a harmonisation is reached with tolerable differences.

Progress in Issue G: The designer has implemented a classification system. This classification is based on the importance for safety of the SSCs (e.g. for Konvoi power plants the SSCs of the pressure retaining boundary are classified in K1 and conventional systems are classified in K5). For electrical systems and I&C a safety related classification is used according to RSK Guidelines and KTA Standards which comply to international safety standards. The assessment showed that harmonisation is reached.

Progress in Issue H: Adaptation of OLCs to plant modification and new insights gained are part of licensed instructions and also part of the experience feedback systems (internal and external). This is applied in the processes of the integrated management system. Beyond activities in periodic safety reviews there are occasional reviews of OLC. For normal and abnormal operation as well as DBAs OLCs are defined and documented in different parts of the operating manual (e.g. safety relevant limits, operational conditions of the plant). In case of deviations of OLCs procedures to bring the plant back into a safe state are described in the operating manual (e.g. non-availability allowance times, KMA/RMA-Meldungen, etc.). For DBAs there additionally exist safety goal oriented procedures to bring the plant back into a safe state. All Procedures, including procedures to deal with unclear plant states and conflicting measurements, are part of the training of shift personal e.g. in the simulator training. The assessment showed that harmonisation is reached.

Progress in Issue LM: EOPs for DBA are contained in the operation manual. EOPs for DBAs and selected severe accident conditions are contained in the emergency manual. SAMGs are currently being considered to be developed, so on this point harmonisation is not yet reached.

Progress in Issue N: On the background that safety issue N, footnote 47, opens the approach of an integrated set of documents, an analysis has been carried out to verify, that the existing documentation fulfils the requirements of Issue N.

Progress in Issue O: Level 1 PSAs have been performed for all plants. Level 2 PSAs for the operational state are developed. They are finalized for most plants and will be finalized for the other plants in the next years. Due to the very good results of PSA Level 1 for fire and shutdown states PSA Level 2 for these states are considered to deliver only negligible safety improvements. Therefore a Level 2 PSA for fire and shutdown states is currently not practiced. Furthermore PSA are not used to assess the adequacy of plant modifications.

Progress in Issue P: The compliance with licensing requirements is legally required through § 19.1 of the Atomic Energy Act. The safety significance of deviations from applicable current safety standards and best international practices are evaluated with respect to the impact on the fulfilment of the fundamental safety goals. This approach is named "Schutzzielorientierte Vorgehensweise (protection goal oriented review)". The assessment showed that harmonisation is reached.

Progress in Issue S: A fire analysis is part of each PSA Level 1 and has been developed for all NPPs. The assessment showed that harmonisation is reached.

Country	HUNGARY
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
As it was identified during the benchmark process in 2006, most of the Reference Levels were implemented at the NPP Paks. 37 Reference Levels were evaluated as C (a difference exists, and should be addressed for harmonization). They were identified: - mainly within the issue E (Design Basis Envelope for Existing Reactors), LM (Emergency Operating Procedures and Severe Accident Management Guidelines) and O (Probabilistic Safety Analysis) and - a few were related to issue A (Safety Policy), F (Design Basis Envelope for existing reactors), J (System for Investigation of Events and Operational Experience Feedback) and S (Protection against internal fires).	
Progress made up to September 2010 and envisaged further actions	
HAEA established a national action plan in collaboration with the NPP Paks to manage the not implemented Reference Levels in 2006. The NPP Paks made the last Periodic Safety Review in 2007. During this review the Licensee used the Reference Levels (version January 2006) as one of the international good practices. After the issuance of latest version of Reference levels (January 2008) the action plan was updated. In accordance with the schedule many tasks were performed. Due to these activities 13 Reference Levels from 37 were implemented. For all other Reference Levels (24), the actions are on-going in line with action plan. Typically they are related to the LM (Emergency Operating Procedures and Severe Accident Management Guidelines) and O (Probabilistic Safety Analysis) issues. The subtasks of action plan related to the Severe Accident Management Guidelines (SAMG) planned unit by unit. In the case of Unit 1 the activities will be completed by the end of 2011. All SAMG activities will be finished by the end of 2014. The actions related to the Probabilistic Safety Analysis will be also finalized by the end of 2014.	

<table>
<tr><th>Country</th><th>ITALY</th></tr>
<tr><td colspan="2"><u>Situation in 2006 with respect to the implementation of the RLs on the NPPs</u></td></tr>
<tr><td colspan="2">In the original action plan issue (2006), two issues were identified as not explicitly implemented in Italy, given the specific status of the nuclear power plants. They were referred to the issue C - Quality Management and were addressed to:
• the personnel training, requiring operating personnel to understand the safety consequences of the activities,
• the need to provide a documented self assessment of managers.</td></tr>
<tr><td colspan="2"><u>Progress made up to September 2010 and envisaged further actions</u></td></tr>
<tr><td colspan="2">Such requirements have been recently, explicitly required to be included in the Quality Assurance Programmes of the licensees, during the QAP general revision that follows the decommissioning license.
When, in 2007, the issue C reference levels were modified into Management System, as a result of the benchmark in Budapest, the lack of requirements proper of an updated Management System was considered justified in Italy, due to the plats' status. Nevertheless, the licensee has issued a Management System Manual, produced to update the requirements on the basis of the most recent standards; the implementing documents are going to be reviewed in the frame of QAP general revision mentioned above.</td></tr>
</table>

Country	LITHUANIA
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
According to benchmarking results 227 of RLs were covered in substance, 62 RLs were not fully covered, but the differences can be justified from a safety point of view, and 2 RLs should be addressed for harmonization.	
Progress made up to September 2010 and envisaged further actions	
Status of Ignalina NPP The first Ignalina NPP unit was shutdown by the end of 2004. By the end of September 2010 reactor core was defueled, nuclear fuel is still in storage pools located in the unit. In operation are necessary for such state supporting systems. By the end of 2009 second (the last in Lithuania) Ignalina NPP unit was shutdown and by the end of September 2010 was in permanent shutdown. Some systems, devoted particularly for operation, are already switched off and isolated. Reactor core still contains nuclear fuel, so necessary equipment is still in operation. ***Implementation of RLs*** The plan on implementation of RL's at Ignalina NPP was agreed with VATESI in 2007, where 75 RLs were foreseen to be implemented. By the end of 2009, 52 out of foreseen 75 RLs levels were implemented. As the final version of part C "Management system" was issued shortly before planned shutdown of the last (second) Unit by the end of 2009, it was recognized that implementation of 23 RL's from this part is not an urgent issue. Later, taking into account state of Ignalina NPP, VATESI agreed with proposal not to implement the rest RLs.	

Country	THE NETHERLANDS
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
After two 10-yearly PSRs including the IAEA requirements and guides, a lot of safety backfitting has been implemented in the Borssele NPP. The implementation is head of national regulations. In total 15 serious differences had to be addressed; in 2006 5 were not yet finalized.	
Progress made up to September 2010 and envisaged further actions	
Implementation of RLs was completed in 2009.	

Country	**ROMANIA**
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
The outcome of the benchmarking was that most of the RLs were actually implemented. The evidence that has been used for benchmarking is heavily relying on documentation that has been approved by CNCAN, having the updated safety analysis report as the major source of information for verification of the implementation. A number of plant's procedures, especially operating procedures and their technical basis' documents, inspection and maintenance procedures, as well as procedures relevant for the control of modifications, have also been checked for more detailed information relevant to specific reference levels. In addition, the industrial standards and codes used for the plant design and various operational programmes (e.g. periodic inspection programme, fire protection programme, etc.) have been consulted. As part of the verification process, CNCAN staff has also conducted inspections and interviews with different technical managers from the plant. For specific issues related to design, the design manuals for various systems and the accident analyses, as well as the probabilistic safety assessments have been consulted for ensuring the accuracy of the information presented during benchmarking. The RLs that were not implemented at the time of the benchmarking are related to the severe accident management programme (issues F, LM), development of PSA Level 2 (issue O) and performance of a PSR (issue P).	
Progress made up to September 2010 and envisaged further actions	
The SAMGs (severe accident management guidelines) are currently in process of being customised, work has started on the development of a PSA Level 2 and the first PSR for Cernavoda NPP Unit 1 is ongoing.	

Country:	SLOVAKIA
<u>2006 Status in the country (what were the main areas needing improvement)</u>:	
In 2006 there were 19 RLs evaluated as "C" in total to be implemented. Most of them were in Issue F – Design extension and LM - EOPS/SAMG.	
<u>Progress made till 2010:</u>	
Works to address all RLs on implementation side were commenced with works on the revision of national legal documents in parallel. All licensees were informed about the national action plan and were asked to prepare for harmonization on implementation side. All WENRA RLs were used as benchmark criteria during the Periodic safety review after 10 years operation at all NPPs in Slovakia between 2008 and 2010). Licensees prepared their own implementation projects that are subject to UJD regular reviews. A large project Implementation of hardware modifications for Severe Accidents mitigation commenced in 2008 on all operating NPPs in Slovakia. These modifications might be considered as the most demanding and resource intensive out of all RLs. The project is under continual oversight from UJD inspectors. It is scheduled till the end of 2016. There is no systematic approach to inspect implementation of RLs from UJD side at the moment. Nevertheless, it is expected that licensees will update its benchmarking till the end of 2010 and during the 2011 UJD will inspect implementation of already implemented RLs.	

Country	SLOVENIA
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
The implementation level of WENRA RL's found out during the benchmarking exercises was much higher than the level of harmonization of Slovenian regulation with WENRA RL's. Only 11 (C category) issues were not in place. Additional 8 issues were categorized as B, meaning that implementation is resolved in a different way or implementation was on the way but not finished.	
Progress made up to September 2010 and envisaged further actions	
During the period 2006-20010 improvements have been achieved in WENRA requirements implementation in Krško NPP. Open issues allocated during the benchmark exercise have been implemented to various degrees. Issues relates to Ageing Management (I) and Environmental Qualification which have been assessed as B, will be fully implemented by the end of the year 2010. Implementation is not performed yet completely for plant staff, sufficiency and changes assessment, staff long term planning, management and supervision of contractors work, quality management system (QMS) role in organizational changes and way of implementation of QMS, SAR update with relevant decommissioning data and PSA use to assess significance of operational occurrences. Implementation of these WENRA reference levels will be completed during the year 2011.	

Country	SPAIN
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
The national results for Spain in the 2006 report contained some reference levels that were not implemented (less than 20 "C"). Issues requiring actions for the implementation of the reference levels were: A, B, E, J, K, O, Q and S. In 2007, the RHWG performed a revision of the matrixes to take into account the updated reference levels on issues C, E and F mainly. During this revision, (self-assessment of the benchmark results) the codification of some reference levels was changed from "A" to "C" and vice versa. In consequence, new actions were required on the implementation side. In addition some reference levels that in the 2006 report were qualified as "C " were re-qualified as "A", as it was considered after discussions with the licensees that they were already implemented in the plants. In summary actions were required on the following issues: - C. Management System - F. Design Extension of Existing Reactors - Maintenance, In-service inspection and Functional Testing - L+M. Emergency Operating Procedures and Severe Accident Management Guidelines - O. Probabilistic safety analysis - S. Protection against internal fires.	
Progress made up to September 2010 and envisaged further actions	
For issue C "Management system" a working group was set up between CSN and the licensees in order to develop a guide with the criteria for the application of the management system. The working group held various meetings and the CSN has conducted visits to follow up the implementation status in the plants. As of January 2010, the reference levels of issue "C" are implemented in all plants. For issue L+M there is a standby situation in one plant (Trillo) because it is not decided yet how will be developed the SAMG guidelines for plants of similar design to the Trillo NPP. With the new CSN instructions published so far the reference levels of some of the aforementioned issues are now required and the licensees are in the process to implement the reference levels at the NPPs. For those instructions in draft/decision making process, the RFs are yet not implemented in all plants.	

Country	SWEDEN
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
In the fall of 2007, SKI invited the licensees to benchmark the implementation of the reference levels on the NPPs. The result was compared with SKI´s own assessment and showed quite few differences. Most of them had to do with different interpretation of some of the reference levels. After the revision of the reference levels in 2007 and the new benchmarks, 20 C-differences remained on the implementation side distributed over issue E (5), issue F (4), issue G (1), issue H (1), issue K (1), issue LM (1), issue N (2), issue O (3) and issue S (2). These C-differences had to do with analyses of some events and combination of events, some design issues such as anchoring of isolation valves, prevention of hydrogen explosion and supplementary control posts, environmental qualification, documentation of verification and validation of EOPs and SAMGs, descriptions in SAR of safety management and deterministic fire analysis.	
Progress made up to September 2010 and envisaged further actions	
Orders of SKI to supplement the SARs as well as the large modernisation programmes at the Swedish NPPs ongoing since 2007and planned to be finalised by 2013, have taken care of most of the remaining C-differences. Eight C-differences remain to deal with after 2010 distributed over issue E (2), issue F (2), issue G (1), issue O (1) and issue S (2). The most important design item remaining is to upgrade the supplementary control posts at some reactors. These actions are planned to be completed 2012. The last outstanding C-differences on the implementation side are scheduled to be completed 2013.	

Country	SWITZERLAND
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
In 2006, the main areas needing improvements were issues K (Maintenance, In-service Inspection and Functional Testing - 5 RLs to be addressed) and O (PSA - 4 RLs to be addressed).	
Progress made up to September 2010 and envisaged further actions	
The majority of open points were resolved in the past years. Examples: Issue K - Qualification of non-destructive testing: Reference Levels K/3.2, 3.10, 3.11: Since 2008 the Swiss Association for Technical Inspections (SVTI) has been responsible for the qualification of non-destructive testing in all NPPs. All NPPs fully comply with the Reference Levels mentioned before. Issue O - PSA: Reference Level O/1.1: The Level 1 PSA models include all relevant internal and external initiating events considering full power, low power as well as shutdown operation. Level 2 PSAs for full power are fully developed. The Level 2 PSAs for shutdown operational state are fully developed for the Gösgen and Beznau NPP. Those for the Mühleberg and Leibstadt NPP are scheduled for 2010/2011. Issue LM - Training/SAMG: Reference Level LM/6.1: Since 2007 all NPPs have their own full-scope-replica simulator. There is no longer any training using simulators which are not plant specific. SAMG have become a part of training for all staff involved. Emergency exercises, simulator training and classroom training are used. All NPPs fully comply with Reference Level LM/6.1. Issue N - SAR: Inspection / testing / operational feedback / ageing management Reference Level N/2.8: Inspection, testing, operational feedback and ageing management are addressed in the SAR of each NPP. All NPPs fully comply with Reference Level N/2.8. In order to ensure the implementation of all RLs, open points identified during the first self assessment performed by the ENSI (then HSK) were directly communicated to the licensees of the NPPs in a special meeting. The oversight of the implementation is performed by ENSI's Inspection Section. For each NPP there is a dedicated ENSI Inspector who coordinates the work according to the applicable oversight processes. The implementation will be finished in 2011.	

Country	UNITED KINGDOM
Situation in 2006 with respect to the implementation of the RLs on the NPPs	
For the UK, the position on the implementation side for the WENRA reactor safety reference levels (RLs) was somewhat better than the position on the regulatory side. [There were less than 20 Cs on the implementation side compared to greater than 120 Cs on the regulatory side].	
Progress made up to September 2010 and envisaged further actions	
Process by Licensees In the United Kingdom two licensees operate nuclear power plants. British Energy Generation Ltd operates both a fleet of Advanced Gas-cooled Reactors (AGRs) and a Pressurised Water Reactor (PWR). Magnox North Limited operates four gas-cooled Magnox reactors at two sites. Before the end of 2010 both these licensees will have completed a self assessment and produced a finalised report describing how they comply with the January 2008 WENRA reactor safety reference levels. Process by Regulator As noted in the United Kingdom regulatory side appendix, a well developed programme of Technical Assessment Guide (TAG) revisions is in place. During these revisions, RLs relevant to the technical area are included. These TAGs are used during the routine inspection and assessment activities of regulator, the Nuclear Installations Inspectorate (NII). Through this means, confidence that the RLs are implemented is built into the routine activities of the regulator. Envisaged Further Actions Taking into account both the licensee and regulator processes, there is adequate confidence that the RLs are fully implemented on operating nuclear power plants in the UK. The RHWG is currently producing a methodology to develop enhanced common understanding of selected RLs. Future UK participation in the application of this methodology will provide added confidence in the implementation side.	

Western European
WENRA
Nuclear Regulators Association

WENRA

Reactor Safety Reference Levels

January 2008

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

WENRA Reactor Safety Reference Levels

January 2008

Issue A: Safety Policy	
Document status: Final	Safety area: Safety Management

Reference levels

1. Issuing and communication of a safety policy

1.1 A written safety policy[1] shall be issued by the licensee.

1.2 The safety policy shall be clear about giving safety an overriding priority in all plant activities.

1.3 The safety policy shall include a commitment to continuously develop safety.

1.4 The safety policy shall be communicated to all site personnel with tasks important to safety, in such a way that the policy is understood and applied.

1.5 Key elements of the safety policy shall be communicated to contractors, in such a way that licensee's expectations and requirements are understood and applied in their activities.

2. Implementation of the safety policy and monitoring safety performance

2.1 The safety policy shall require directives for implementing the policy and monitoring safety performance.

2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the plant management.

3. Evaluation of the safety policy

3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, more frequent than the periodic safety reviews.

[1] A safety policy is understood as a documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as a visible part of an integrated organisational policy.

Issue B: Operating Organisation	
Document status: Final	Safety area: Safety Management

Reference levels

1. Organisational structure

1.1 The organisational structure for safe and reliable operation of the plant, and for ensuring an appropriate response in emergencies, shall be justified[2] and documented.

1.2 The adequacy of the organisational structure, for its purposes according to 1.1, shall be assessed when organisational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated[3] after implementation.

1.3 Responsibilities, authorities, and lines of communication shall be clearly defined and documented for all staff with duties important to safety.

2. Management of safety and quality

2.1 The licensee shall ensure that the plant is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

2.2 The licensee shall ensure that decisions on safety matters are preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

2.3 The licensee shall ensure that the staff is provided with the necessary facilities and working conditions to carry out work in a safe manner.

2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are analysed in a systematic way and continuously used to improve the plant and the licensee's activities.

2.6 The licensee shall ensure that plant activities and processes are controlled through a documented management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

[2] The arguments shall be provided that the organisational structure supports safety and an appropriate response in emergencies.
[3] A verification that the implementation of the organisational change has accomplished its safety objectives.

3. Sufficiency and competency of staff

3.1 The required number of staff for safe operation[4], and their competence, shall be analysed in a systematic and documented way.

3.2 The sufficiency of staff for safe operation, their competence, and suitability for safety work shall be verified on a regular basis and documented.

3.3 A long-term staffing plan[5] shall exist for activities that are important to safety.

3.4 Changes to the number of staff, which might be significant for safety, shall be justified in advance, carefully planned and evaluated after implementation.

3.5 The licensee shall always have in house, sufficient, and competent staff and resources to understand the licensing basis of the plant (e.g. Safety Analysis Report or Safety Case and other documents based thereon), as well as to understand the actual design and operation of the plant in all plant states.

3.6 The licensee shall maintain, in house, sufficient and competent staff and resources to specify, set standards manage and evaluate safety work carried out by contractors.

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[4] Operation is defined as all activities performed to achieve the purpose for which a nuclear power plant was constructed (according to the IAEA Glossary).

[5] Long term is understood as 3-5 years for detailed planning and at least 10 years for prediction of retirements etc.

Issue C: Management System	
Document status: Final	Safety area: Safety Management

1. Objectives

1.1 An integrated management system shall be established, implemented, assessed and continually improved by the licensee. The main aim of the management system shall be to achieve and enhance nuclear safety by ensuring that other demands[6] on the licensee are not considered separately from nuclear safety requirements, to help preclude their possible negative impact on nuclear safety.

2. General requirements

2.1 The application of management system requirements shall be graded so as to deploy appropriate resources, on the basis of the consideration of:

- The significance and complexity of each activity and its products;
- The hazards and the magnitude of the potential impact associated with each activity and its products;
- The possible consequences if an activity is carried out incorrectly or a product fails.

2.2 The documentation of the management system shall include the following:

- The policy statements of the licensee;
- A description of the management system;
- A description of the organisational structure of the licensee;
- A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
- A description of the interactions with relevant external organisations;
- A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved.

2.3 The documentation of the management system shall be understandable to those who use it. Documents shall be up to date, readable, readily identifiable and available at the point of use.

3. Management commitment

3.1 The licensee shall develop the goals, strategies, plans and objectives of the organization in an integrated manner so that their collective impact on safety is understood and managed.

3.2 The licensee shall ensure that it is clear when, how and by whom decisions are to be made within the management system.[7]

[6] Examples of such demands are health, environmental, security, quality and economic requirements.
[7] With respect to operational decisions that impact on nuclear safety.

3.3 The licensee shall ensure that management at all levels demonstrate its commitment to the establishment, implementation, assessment and continual improvement of the management system and shall allocate adequate resources to carry out these activities.

3.4 The licensee shall foster the involvement of all staff in the implementation and continual improvement of the management system.

4. Resources

4.1 The licensee shall determine the amount of resources[8] necessary and shall provide the resources to carry out the activities of the licensee and to establish, implement, assess and continually improve the management system.

5. Process implementation

5.1 The processes[9] that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the licensee organisation shall be identified, and their development shall be planned, implemented, assessed and continually improved. The sequence and interactions of the processes shall be determined.

5.2 The methods necessary to ensure the effectiveness of both the implementation and the control of the processes shall be determined and implemented.

5.3 Documents[10] shall be controlled. Changes to documents shall be reviewed and recorded and shall be subject to the same level of approval as the documents themselves. It shall be ensured that document users are aware of and use appropriate and correct documents.

5.4 Records shall be specified in the management system documentation and shall be controlled. All records shall, for the duration of the retention times specified for each record, be readable, complete, identifiable and easily retrievable.

5.5 The control of processes, or work performed within a process, contracted to external organizations shall be identified within the management system. The licensee shall retain overall responsibility when contracting any processes or work performed within a process.

5.6 Suppliers of products and services shall be selected on the basis of specified criteria and their performance shall be evaluated.

5.7 Purchasing requirements shall be developed and specified in procurement documents. Evidence that products meet these requirements shall be available to the licensee before the product is used.

5.8 It shall be confirmed[11] that activities and their products meet the specified requirements and shall ensure that products perform satisfactorily in service.

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[8] "Resources" includes individuals, infrastructure, the working environment, information and knowledge, and suppliers, as well as material and financial resources.

[9] This is not understood as a full process orientation of the management system. Also functional or organisational oriented routines and procedures could be used for certain activities together with cross cutting processes for other activities.

[10] Documents may include: policies; procedures; instructions; specifications and drawings (or representations in other media); training materials; and any other texts that describe processes, specify requirements or establish product specifications.

[11] Through inspection, testing, verification and validation activities before the acceptance, implementation, or operational use of products.

6. Measurement, assessment and improvement

6.1 In order to confirm the ability of the processes to achieve the intended results and to identify opportunities for improvement:

- The effectiveness of the management system shall be monitored and measured;
- The licensee shall ensure that managers carry out self-assessment of the performance of work for which they are responsible;
- Independent[12] assessments shall be conducted regularly on behalf of the licensee.

6.2 An organizational unit shall be established with the responsibility for conducting independent assessments. This unit shall have sufficient authority to discharge its responsibilities. Individuals conducting independent assessments shall not assess their own work.

6.3 The licensee shall evaluate the results of the assessments and take any necessary actions, and shall record and communicate inside the organisation the decisions and the reasons for the actions.

6.4 A management system review shall be conducted at planned intervals to ensure the effectiveness of the management system.

6.5 The causes of non-conformances shall be determined and remedial actions shall be taken to prevent their recurrence.

6.6. Improvement plans shall include plans for the provision of adequate resources. Actions for improvement shall be monitored through to their completion and the effectiveness of the improvement shall be checked.

[12] By an external organisation or by an internal independent assessment unit.

Issue D: Training and Authorization of NPP staff (jobs with safety importance)	
Document status: Final	Safety area: Safety Management

Reference levels

1. Policy

1.1 The licensee shall establish an overall training policy and a comprehensive training plan on the basis of long-term competency needs and training goals that acknowledges the critical role of safety. The plan shall be kept up to date.

1.2 A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

2. Competence and qualification

2.1 Only qualified persons that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The licensee shall ensure that all personnel performing safety-related duties including contractors have been adequately trained and qualified.

2.2 The Licensee shall define and document the necessary competence requirements for their staff.

2.3 Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

2.4 Staff qualifying for positions important to safety shall undergo a medical examination to ensure their fitness depending upon the duties and responsibilities assigned to them. The medical examination shall be repeated at specified intervals.

3. Training programmes and facilities

3.1 Performance based training programmes shall be established for all staff with tasks important to safety. The programmes shall cover basic training in order to qualify for a certain position and refresher training as needed.

3.2 All technical staff including on-site contractors shall have a basic understanding of nuclear safety, radiation safety, fire safety, the on-site emergency arrangements and industrial safety.

3.3 Representative simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures is effective. The simulator shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions[13].

3.4 For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator.[14]

3.5 Refresher training for control room operators shall include especially the following items as appropriate:

- Plant operation in normal operational states, selected transients and accidents;
- Shift crew teamwork;
- Operational experiences and modifications of plant and procedures.

3.6 Maintenance and technical support staff including contractors shall have practical training on the required safety critical activities.

4. Authorization

4.1 Staff controlling changes in the operational status of the plant shall be required to hold a authorization valid for a specified time period. The licensee shall establish procedures for their staff to achieve this authorization. In the assessment of an individual's competence and suitability as a basis for the authorization, documented criteria shall be used.

4.2 If an authorised individual:

- Moves to another position for which an authorization is required;
- Has been absent from the authorised position during an extended time period;

Re-authorisation shall be conducted after necessary individual preparations.

4.3 Work carried out by contractor personnel on structures, systems, or components that are important to safety shall be approved and monitored by a suitably competent member of licensee's staff.

[13] This type of simulator is known as a full-scope simulator.
[14] Time includes the necessary briefings.

Appendix E	**Issue: Design Basis Envelope for Existing Reactors**
Document status: Final	Safety area: Design

Reference levels:

1. Objective

1.1 The design basis[15] shall have as an objective the prevention or, if this fails, the mitigation of consequences resulting from anticipated operational occurrences and design basis accident conditions. Design provisions shall be made to ensure that potential radiation doses to the public and the site personnel do not exceed prescribed limits and are as low as reasonably achievable.

2. Safety strategy

2.1 Defence-in-depth[16] shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases. The design shall therefore provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment, and an adequate protection of the barriers.

2.2 The design shall prevent as far as practicable:

- challenges to the integrity of the barriers;
- failure of a barrier when challenged;
- failure of a barrier as consequence of failure of another barrier.

3. Safety functions

3.1 The plant shall be able to fulfil the following fundamental safety functions[17]:

- control of reactivity,
- removal of heat from the core and

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[15] The design basis shall be reviewed and updated during the lifetime of the plant (see ref level 11.1).
[16] Defined in the IAEA Safety Requirements NS-R-1, 2.9- 2.11. Further information is provided in INSAG-10.
[17] Under the conditions specified in the following paragraphs.

- confinement of radioactive material,

in the plant states: normal operation, anticipated operational occurrences and design basis accident conditions.

4. Establishment of the design basis

4.1 The design basis shall specify the capabilities of the plant to cope with a specified range of plant states[18] within the defined radiation protection requirements. Therefore, the design basis shall include the specification for normal operation and transients/accident conditions from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis.

4.2 A list of PIEs shall be established to cover all events that could affect the safety of the plant. From this list, a set of design basis events shall be selected with deterministic or probabilistic methods or a combination of both, and used to set the boundary conditions according to which the structures, systems and components important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

4.3 The design basis shall be systematically defined and documented to reflect the actual plant.

5. Set of design basis events

5.1 Internal events such as loss of coolant accidents, equipment failures, maloperation and hazards, and their consequential events, shall be taken into account in the design of the plant. The list of events shall be plant specific. (see Appendix for assessment of implementation)

5.2 The following types of natural and man made external events shall as a minimum be taken into account in the design of the plant according to site specific conditions:

- extreme[19] wind loading
- extreme outside temperatures
- extreme rainfall, snow conditions and site flooding
- extreme cooling water temperatures and icing
- earthquake
- airplane crash
- other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the nuclear power plant

[18] Normal operation, anticipated operational occurrences and design basis accident conditions.
[19] Definition of "extreme" is based on historical weather data for the site region

6. Combination of events

6.1 Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accident conditions, shall be considered in the design. Engineering judgement and probabilistic methods can be used for the selection of the event combinations.

7. Definition and application of technical acceptance criteria

7.1 Initiating events shall be grouped into a limited number of categories that correspond to plant states[20], according to their probability of occurrence. Radiological and technical acceptance criteria shall be assigned to each plant state such that frequent initiating events shall have only minor or no radiological consequences and that events that may result in severe consequences shall be of very low probability.

7.2 Criteria for protection of the fuel rod integrity, including fuel temperature, DNB, and cladding temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis event.

7.3 Criteria for the protection of the (primary) coolant pressure boundary shall be specified, including maximum pressure, maximum temperature, thermal- and pressure transients and stresses.

7.4 If applicable, criteria in 7.3 shall be specified as well for protection of the secondary coolant system.

7.5 Criteria shall be specified for protection of containment, including temperatures, pressures and leak rates.

8. Demonstration of reasonable conservatism and safety margins

8.1 The initial and boundary conditions shall be specified with conservatism.

8.2 The worst single failure[21] shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE.

8.3 Only safety systems shall be credited to carry out a safety function. Non-safety systems shall be assumed to operate only if they aggravate the effect of the initiating event[22].

8.4 A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis events[23].

8.5 The safety systems shall be assumed to operate at their performance level that is most penalising for the initiator.

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[20] See footnote 16

[21] A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.

[22] This means that non-safety systems are either supposed not to function after the initiator, either supposed to continue to function as before the initiator, depending on which of both cases is most penalising.

[23] This assumption is made to ensure the sufficiency of the shutdown margin. The stuck rod selected is the highest worth rod at Hot Zero Power and conservative values of reactor trip reactivity (conservative time delay and reactivity versus CR position dependence) are used. A stuck rod can be handled as single failure in the DBA-analysis if the stuck rod itself is the worst single failure.

8.6 Any failure, occurring as a consequence of a postulated initiating event, shall be regarded to be part of the original PIE.

8.7 The impact of uncertainties, which in specific cases are of importance for the results, shall be addressed in the analysis of design basis events.

9. Design of safety functions

General

9.1 The fail-safe principle shall be considered in the design of systems and components important to safety.

9.2 A failure in a system intended for normal operation shall not affect a safety function.

9.3 Activations and manoeuvring of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes after the initiating event. Any operator actions required by the design within 30 minutes after the initiating event shall be justified[24].

9.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components[25], redundancy, diversity[26], physical and functional separation and isolation.

Reactor shutdown functions

9.5 The means for shutting down the reactor shall consist of at least two diverse systems.

9.6 At least one of the two systems shall, on its own, be capable of quickly[27] rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.

Heat removal functions

9.7 Means for removing residual heat from the core after shutdown, and during and after anticipated operational occurrences and accident conditions, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Confinement functions

9.8 A containment system shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. This system shall include:

- leaktight structures covering all essential parts of the primary system;
- associated systems for control of pressures and temperatures;
- features for isolation;

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[24] The control room staff has to be given sufficient time to understand the situation and take the correct actions. Operator actions required by the design within 30 min after the initiating event have to be justified and supported by clear documented procedures that are regularly exercised in a full scope simulator.

[25] Proven by experience under similar conditions or adequately tested and qualified.

[26] The potential for common cause failure shall be considered to determine where diversity should be applied to achieve the necessary reliability.

[27] Within 4-6 seconds, i.e. scram system.

- features for the management and removal of fission products, hydrogen, oxygen and other substances that could be released into the containment atmosphere.

9.9 Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

9.10 Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

10. Instrumentation and control systems

10.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems and the containment, and for obtaining any information on the plant necessary for its reliable and safe operation. Provision shall be made for automatic recording[28] of measurements of any derived parameters that are important to safety.

10.2 Instrumentation shall be adequate for measuring plant parameters and shall be environmentally qualified for the plant states concerned.

Control room

10.3 A control room shall be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences and design basis accidents.

10.4 Devices shall be provided to give in an efficient way visual and, if appropriate also audible indications of operational states and processes that have deviated from normal and could affect safety. Ergonomic factors shall be taken into account in the design of the control room. Appropriate information shall be available to the operator to monitor the effects of the automatic actions.

10.5 Special attention shall be given to identifying those events, both internal and external to the control room, which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

[28] By computer sampling and/or print outs.

10.6 For times when the main control room is not available, there shall be sufficient instrumentation and control equipment available, at a single location that is physically and electrically separated from the control room, so that the reactor can be placed and maintained in a shut down state, residual heat can be removed, and the essential plant parameters can be monitored.

Protection system

10.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:

- no single failure results in loss of protection function; and
- the removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

10.8 The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in operation. Exceptions shall be justified.

10.9 The design of the reactor protection system shall minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal operation and anticipated operational occurrences. Furthermore, the reactor protection system shall not prevent operators from taking correct actions if necessary in design basis accidents.

10.10 Computer based systems used in a protection system, shall fulfil the following requirements:

- the highest quality of and best practices for hardware and software shall be used;
- the whole development process, including control, testing and commissioning of design changes, shall be systematically documented and reviewed;
- in order to confirm confidence in the reliability of the computer based systems, an assessment of the computer based system by expert personnel independent of the designers and suppliers shall be undertaken; and
- where the necessary integrity of the system cannot be demonstrated with a high level of confidence, a diverse means of ensuring fulfilment of the protection functions shall be provided.

Emergency power

10.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

11. Review of the design basis

11.1 The actual design basis shall regularly[29], and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach to identify needs and opportunities for improvement. Reasonably practicable measures shall be taken with respect to backfitting or other measures justified from a safety point of view.

Appendix

Interpretation of the reference level 5.1, for the purpose of benchmarking of implementation, in terms of types of internal events to be included in the safety analysis as a minimum:

The list mainly applies on PWR and BWR. For the other designs: AGR, CANDU and RBMK used in single WENRA countries, the list has to be adapted to the reactor type and implementation checked as self-assessment by the concerned country. The final list will in all cases be plant type specific.

Initiating events

- small, medium and large LOCA (break of the largest diameter piping of the Reactor Coolant Pressure Boundary)
- breaks in the main steam and main feed water systems
- forced decrease of reactor coolant flow
- forced increase or decrease of main feed water flow
- forced increase or decrease of main steam flow
- inadvertent opening of valves at the pressurizer (PWR)
- inadvertent operation of the emergency core cooling system ECCS
- inadvertent opening of valves at the steam generators (PWR)
- inadvertent opening of main steam relief/safety valves (BWR)
- inadvertent closure of main steam isolation valves
- steam generator tube rupture (PWR)
- uncontrolled movement of control rods
- uncontrolled withdrawal/ejection of control rod
- core instability (BWR)
- chemical and volume control system (CVCS) malfunction (PWR)
- pipe breaks or heat exchanger tube leaks in systems connected to the RCS and located partially outside containment (Interfacing System LOCA)
- fuel handling accidents

[29] Regularly is understood as an ongoing activity to analyse the plant and identify opportunities for improvement. The periodic safety reviews are complementary tools to verify and follow up on this activity in a longer perspective. Significant new safety information is understood as new insights gained from e.g. safety analyses and the development of safety standards and practices.

- loss of off-site power
- load drop by failure of lifting devices

Initiating events as well as consequential events (could be both types)

- fire
- explosion
- flooding

Consequential events

- missile generation, including turbine missiles
- release of fluid (oil etc) from failed systems
- vibration
- pipe whip
- jet impact

Appendix F	**Issue: Design Extension of Existing Reactors**
Document status: Final	Safety area: Design

Reference levels:

1. Objective

1.1 The design extension[30] analysis shall examine the performance of the plant in specified accidents beyond the design basis, including selected severe accidents, in order to minimise as far as reasonably practicable radioactive releases harmful to the public and the environment in cases of events with very low probability of occurrence.

2. Selection and analysis of Beyond Design Basis Events

2.1 Beyond design basis events shall be selected[31] and considered in the safety analysis to determine those sequences for which reasonable practicable preventive or mitigative measures can be identified and implemented (see Appendix for assessment of implementation).

2.2 Realistic assumptions and modified[32] acceptance criteria may be used for the analysis of the beyond design basis events.

3. **Instrumentation for the management of beyond design basis accident conditions**

3.1 Adequate instrumentation shall exist which can be used in severe accident environmental conditions in order to manage such accidents according to guidelines/procedures for severe accidents.

3.2 Necessary information from instruments shall be relayed to the control room as well as to a separately located supplementary control room/post and be presented in such a way to enable a timely assessment of the plant status and critical safety functions in severe accident conditions.

[30] Design extension is understood as measures taken to cope with additional events or combination of events, not foreseen in the design of the plant. Such measures need not involve application of conservative engineering practices but could be based on realistic, probabilistic or best estimate assumptions, methods and analytical criteria.
[31] Based on a combination of deterministic and probabilistic assessments as well as engineering judgement.
[32] Modified in relation to the conservative criteria used in the analysis of the design basis events.

4. Protection of the containment against selected beyond design basis accidents[33]

4.1 Isolation of the containment shall be possible in a beyond design basis accident.[34] However, if an event leads to bypass of the containment, consequences shall be mitigated.

4.2 The leaktightness of the containment shall not degrade significantly for a reasonable time after a severe accident.

4.3 Pressure and temperature in the containment shall be managed in a severe accident.

4.4 Combustible gases shall be managed in a severe accident.

4.5 The containment shall be protected from overpressure in a severe accident[35].

4.6 High pressure core melt scenarios shall be prevented.

4.7 Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

Appendix

Interpretation of the reference level 2.1, for the purpose of benchmarking of implementation, in terms of types events to be analysed for design extension as a minimum, if not already considered in the design basis:

- anticipated transient without scram (ATWS)
- station black out
- total loss of feed water
- LOCA together with the complete loss of one emergency core cooling system[36]
- uncontrolled level drop during mid-loop operation (PWR) or during refuelling
- total loss of the component cooling water system
- loss of core cooling in the residual heat removal mode
- loss of fuel pool cooling
- loss of ultimate heat sink function
- uncontrolled boron dilution (PWR)
- multiple steam generator tube ruptures (PWR, PHWR)
- loss of required safety systems in the long term after a Postulated Initiating Event

[33] These reference levels aim at providing protection at the level 4 of the defence-in-depth. Such protection could be provided by existing equipment that has been assessed, and if needed modified, to perform the relevant function in a severe accident condition or additional equipment on a best estimate basis.

[34] Special attention needs to be given for certain reactor types to the analysis of severe accident conditions with an open containment during certain shutdown states. Should such an accident occur, it should be possible to achieve timely containment isolation or implement equally effective compensatory measures. Therefore consideration has to be given to the time needed for the restoration of containment isolation and effective leaktightness, taking into account factors such as the progression of the accident sequences.

[35] This reference level could be seen as a special case of reference level 4.3. However, it is kept for clarity as a separate reference level since it might call for specific measures to protect against fast as well as slow containment overpressurization.

[36] Either the high pressure or the low pressure emergency core cooling system

Issue G: Safety Classification of Structures, Systems and Components	
Document status: Final	Safety area: Design

Reference levels

1. Objective

1.1 All SSCs[37] important to safety shall be identified and classified on the basis of their importance for safety.

2. Classification process

2.1 The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment.

2.2 The classification shall identify for each safety class:

- The appropriate codes and standards in design, manufacturing, construction and inspection;
- Need for emergency power supply, qualification to environmental conditions;
- The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;
- The applicable quality requirements

3. Ensuring reliability

3.1 SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.

3.2 The failure of a SSC in one safety class shall not cause the failure of other SSCs in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.

4. Selection of materials and qualification of equipment

4.1 The design of SSCs important to safety and the materials used shall consider the effects of operational conditions over the plant lifetime and the effects of design basis accidents on their characteristics and performance.

4.2 A qualification procedure shall be adopted to confirm that SSCs important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions[38] over the lifetime of the plant and when required in anticipated operational occurrences and accident conditions.

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[37] SSCs include software for I&C.

[38] Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof.

Issue H: Operational limits and conditions	
Document status: Final	Safety area: Operation

Reference levels

1. Purpose

1.1 OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intentions as documented in the SAR.

1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

2. Establishment and review of OLCs

2.1 Each established OLC shall be justified based on plant design, safety analysis and commissioning tests.

2.2 OLCs shall be kept updated and reviewed in the light of experience, developments in science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.

2.3 The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review.

3. Use of OLCs

3.1 The OLCs shall be readily accessible to control room personnel.

3.2 Control room operators shall be highly knowledgeable of the OLCs and their technical basis. Relevant operational decision makers shall be aware of their significance for the safety of the plant.

4. Scope of OLCs

4.1 OLCs shall cover all operational plant states including power operation, shutdown and refuelling, any intermediate conditions between these states and temporary situations arising due to maintenance & testing.

5. Safety limits, safety systems settings and operational limits

5.1 Adequate margins shall be ensured between operational limits and the established safety systems settings, to avoid undesirably frequent actuation of safety systems.

5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.

6. Unavailability limits

6.1 Limits and conditions for normal operation shall include limits on operating parameters, stipulation for minimum amount of operable equipment, actions to be taken by the operating staff in the event of deviations from the OLCs and time allowed to complete these actions.

6.2 Where operability requirements cannot be met, the actions to bring the plant to a safer state shall be specified, and the time allowed to complete the action shall be stated.

6.3 Operability requirements shall state for the various modes of normal operation the number of systems or components important to safety that should be in operating condition or standby condition.

7. Unconditional requirements

7.1 If operating personnel cannot ascertain that the power plant is operating within operating limits, or the plant behaves in an unexpected way, measures shall be taken without delay to bring the plant to a safe and stable state.

7.2 Plant shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

8. Staffing levels

8.1 Minimum staffing levels for shift staff shall be stated in the OLCs.

9. Surveillance

9.1 The licensee shall ensure that an appropriate surveillance[39] program is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

10. Non-compliance

10.1 In cases of non-compliance, remedial actions shall be taken immediately to re-establish OLC requirements.

10.2 Reports of non-compliance shall be investigated and corrective action shall be implemented in order to help prevent such non-compliance[40] in future.

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[39] The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the surveillance programme include not only deficiencies in SSCs and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the plant is deviating from the design intent. *(NS-G-2.6 Para 2.11)*

[40] If the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, plant shall be deemed to have operated in non-compliance with OLCs.

Issue I: Ageing Management	
Document status: Final	Safety area: Operation

Reference levels

1. Objective

1.1. The operating organisation shall have an Ageing Management Programme[41] to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.

2. Technical requirements, methods and procedures

2.1 The licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.

2.3. The Periodic Safety Reviews shall be used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.

2.4. In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.

2.5. The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.

3. Major structures and components

3.1. Ageing management of the reactor pressure vessel[42] and its welds shall take all relevant factors including embrittlement, thermal ageing, and fatigue into account to compare their performance with prediction, throughout plant life.

3.2. Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

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[41] Ageing is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out).
An Ageing Management Programme (AMP) should be understood as an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components.

[42] Or its functional equivalent in other designs

Issue J: System for Investigation of Events and Operational Experience Feedback	
Document status: Final	Safety area: Operation

Reference levels

1. Programmes and Responsibilities

1.1 The licensee shall establish and conduct a programme to collect, screen, analyse, and document operating experience and events at the plant in a systematic way. Relevant operational experience and events reported by other plants shall also be considered.

1.2 Operating experience at the plant shall be evaluated to identify any latent safety relevant failures or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margin.

1.3 The licensee shall designate staff for carrying out these programmes, for the dissemination of findings important to safety and – where appropriate – for recommendations on actions to be taken. Significant findings and trends shall be reported to the licensee's top management.

1.4 Staff responsible for evaluation of operational experience and investigation into events shall receive adequate training, resources, and support from the line management.

1.5 The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

2. Collection and storage of information

2.1 The information relevant to experience from normal and abnormal operation and other important safety-related information shall be organized, documented, and stored in such a way that it can be easily retrieved and systematically searched, screened and assessed by the designated staff.

3. Reporting and dissemination of safety significant information

3.1 The licensee shall report events of significance to safety in accordance with established procedures and criteria.

3.2 Plant personnel shall be required to report abnormal events and be encouraged to report internally near misses relevant to the safety of the plant.

3.3 Information resulting from the operational experience shall be disseminated to relevant staff and shared with relevant national and international bodies.

3.4 A process shall be put in place to ensure that operating experience of events at the plant concerned as well as of relevant events at other plants is appropriately considered in the training programme for staff with tasks related to safety.

4. Assessment and investigation of events

4.1 An initial assessment of events important to safety shall be performed without delay to determine whether urgent actions are necessary.

4.2 The licensee shall have procedures specifying appropriate investigation methods, including methods of human performance analysis.

4.3 Event investigation shall be conducted on a time schedule consistent with the event significance. The investigation shall:

- Establish the complete event sequence;
- Determine the deviation;
- Include direct and root cause analysis;
- Assess the safety significance including potential consequences; and
- Identify corrective actions.

4.4 The operating organisation shall maintain liaison as appropriate with the organizations (manufacturer, research organization, designer) involved in design and construction, with the aims of feeding back information on operating experience and obtaining advice, if necessary, in case of equipment failures or abnormal events.

4.5 As a result of the analysis, timely corrective actions shall be taken such as technical modifications, administrative measures or personnel training to restore safety, to avoid event recurrence and where appropriate to improve safety.

5. Review and continuous improvement of the OEF process

5.1 Periodic reviews of the effectiveness of the OEF process based on performance criteria shall be undertaken and documented either within a self-assessment programme by the licensee or by a peer review team.

Issue K: Maintenance, in-service inspection and functional testing	
Document status: Final	Safety area: Operation

Reference levels

1. Scope and objectives

1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of SSCs important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. They shall take into account operational limits and conditions and be re-evaluated in the light of experience.

1.2 The programmes shall include periodic inspections and tests of SSCs important to safety in order to determine whether they are acceptable for continued safe operation of the plant or whether any remedial measures are necessary.

2. Programme establishment and review

2.1 The extent and frequency of preventive maintenance, testing, surveillance and inspection of SSCs shall be determined through a systematic approach on the basis of:

- Their importance to safety;
- Their inherent reliability;
- Their potential for degradation (based on operating experience, research and vendor recommendation);
- Operational and other relevant experience and results of condition monitoring.

2.2 In-service inspections of nuclear power plants shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

2.3 Data on maintenance, testing, surveillance, and inspection of SSCs shall be recorded, stored and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

2.4 The maintenance programme shall be periodically reviewed[43] in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on plant safety, and their conformance with applicable requirements.

2.5 The potential impact of maintenance upon plant safety shall be assessed.

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[43] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

3. *Implementation*

3.1 SSCs important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

3.2 Procedures shall be established, reviewed, and validated for maintenance, testing, surveillance and inspection tasks.

3.3 A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

3.4 Before equipment is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

3.5 The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined in the procedures.

3.6 Repairs to SSCs shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

3.7 Following any event due to which the safety functions and functional integrity of any component or system may have been challenged, the licensee shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

3.8 The reactor coolant pressure boundary shall be subject to a system leakage test before resuming operation after a reactor outage in the course of which its leak-tightness may been affected.

3.9 The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.

3.10 All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with requirements of the management system.

3.11 Any in-service inspection process shall be qualified[44], in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

3.12 When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall

[44] The ISI system qualification means to demonstrate that the combination of equipment, inspection procedure and personnel is appropriate for testing of a given inspection area according to a technical specification. It is recommended to uses as reference documents, eg the European Regulators Common Position on NDT Qualification, ENIQ methodology and/or IAEA – EBP-VVER-11 documents.

be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.

3.13 Surveillance measures to verify the containment integrity shall include: a) leak rate tests; b) tests of penetration seals and closure devices such as air locks and valves that are part of the boundaries, to demonstrate their leak-tightness and, where appropriate, their operability; c) inspections for structural integrity (such as those performed on liner and pre-stressing tendons).

Issue LM: Emergency Operating Procedures and Severe Accident Management Guidelines	
Document status: Final	Safety area: Operation

Reference levels

1. Objectives

1.1 A comprehensive set of emergency operating procedures (EOPs) for design basis accidents (DBAs) and beyond design basis accidents (BDBAs), and also guidelines for severe accident management (SAMG) shall be provided.

2. Scope

2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the plant state to a safe condition.

2.2 EOPs shall be provided to cover Beyond Design Basis Accidents up to, but not including, the onset of core damage. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent core damage.

2.3 SAMGs shall be provided to mitigate the consequences of severe accidents for the cases where the measures provided by EOPs have not been successful in the prevention of core damage.

2.4 EOPs for Design Basis Accidents shall be symptom-based or a combination of symptom based and event based[45] procedures. EOPs for Beyond Design Basis Accidents shall be only symptom based.

3. Format and Content of Procedures and Guidelines

3.1 EOPs shall be developed in a systematic way and shall be supported by realistic and plant specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, such as alarm response procedures and severe accident management guidelines.

3.2 EOPs shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate among EOPs and to proceed from EOPs to SAMGs.

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[45] Event-based EOPs enable the operator to identify the specific event and encompass:
- Information from significant plant parameters,
- Automatic actions that will probably be taken as a result of the event,
- Subsequent operator actions directed to returning the reactor to a normal condition or to provide for safe, extended and stable shutdown conditions.

Symptom-based EOPs enable the operator to respond to situations for which there are no procedures to identify accurately the event that has occurred. The decisions for measures to respond to such situations are specified in the procedures with respect to the symptoms and the state of systems of the plant (such as the values of safety parameters and critical safety functions).

3.3 SAMGs shall be developed in a systematic way using a plant specific approach. SAMGs shall address strategies to cope with scenarios identified by the severe accident analyses[46].

4. Verification and validation

4.1 EOPs and SAMGs shall be verified and validated in the form in which they will be used in the field, so far as practicable, to ensure that they are administratively and technically correct for the plant and are compatible with the environment in which they will be used.

4.2 The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations, using a simulator, where appropriate.

5. Review and updating of EOPs and SAMGs

5.1 EOPs and SAMGs shall be kept updated to ensure that they remain fit for their purpose.

6. Training

6.1 Shift personnel and on-site technical support shall be regularly trained and exercised, using simulators for the EOPs and, where practicable, for the SAMGs.

6.2 The transition from EOPs to SAMGs for management of severe accidents shall be exercised.

6.3 Interventions called for in SAMGs and needed to restore necessary safety functions shall be planned for and regularly exercised.

[46] Analysis aimed at identifying the plant vulnerabilities to severe accident phenomena, assessment of plant capabilities and development of accident management measures, including for containment protection as defined in Issue F (Design Extension of Existing Reactors) in RLs 4.1 to 4.7. It is understood that for these accident conditions also SAMGs shall be developed.

Issue N: Contents and updating of Safety Analysis Report (SAR)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Objective

1.1 The Licensee shall provide a SAR[47] and use it as a basis for continuous support of safe operation.

1.2 The Licensee shall use the SAR as a basis for assessing the safety implications of changes to the plant or to operating practices.

2. Content of the SAR

2.1 The SAR shall describe the site, the plant layout and normal operation; and demonstrate how safety is achieved.

2.2 The SAR shall contain detailed descriptions of the safety functions; all safety systems and safety-related structures, systems and components; their design basis and functioning in all operational states, including shut down and accident conditions.

2.3 The SAR shall identify applicable regulations codes and standards.

2.4 The SAR shall describe the relevant aspects of the plant organization and the management of safety.

2.5 The SAR shall contain the evaluation of the safety aspects related to the site.

2.6 The SAR shall outline the general design concept and the approach adopted to meet the fundamental safety objectives.

2.7 The SAR shall describe the safety analyses performed to assess the safety of the plant in response to postulated initiating events against safety criteria and radiological release limits.

2.8 The SAR shall describe the emergency operation procedures and accident management guidelines, the inspection and testing provisions, the qualification, and training of personnel, the operational experience feedback programme, and the management of ageing.

2.9 The SAR shall contain the technical bases for the operational limits and conditions.

2.10 The SAR shall describe the policy, strategy, methods, and provisions for radiation protection.

2.11 The SAR shall describe the on-site emergency preparedness arrangements and the liaison and co-ordination with off-site organizations involved in the response to an emergency.

[47] A consistent safety document or integrated set of documents constituting the licensing basis of the plant and updated under control of the regulatory body

2.12 The SAR shall describe the on-site radioactive waste management provisions.

2.13 The SAR shall describe how the relevant decommissioning and end-of-life aspects are taken into account during operation.[48]

3. Review and update of the SAR

3.1 The licensee shall update the SAR to reflect modifications, new regulatory requirements, and relevant standards, as soon as practicable after the new information is available and applicable.

[48] Guidance on the specific aspects that need to be addressed in the SAR is given in Chapter XV of the IAEA Safety Guide GS-G-4.1.

Issue O: Probabilistic Safety Analysis (PSA)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Scope and content of PSA

1.1 For each plant design, a specific PSA shall be developed for level 1 and level 2 including all modes of operation and all relevant initiating events including internal fire and flooding. Severe weather conditions and seismic events shall be addressed[49].

1.2 PSA shall include relevant dependencies[50].

1.3 The basic Level 1 PSA shall contain sensitivity and uncertainty analyses. The basic Level 2 PSA shall contain sensitivity analyses and, as appropriate, uncertainty analyses.

1.4 PSA shall be based on a realistic modelling of plant response, using data relevant for the design, and taking into account human action to the extent assumed in operating and accident procedures.

1.5 Human reliability analysis shall be performed, taking into account the factors which can influence the performance of the operators in all plant states.

2. Quality of PSA

2.1 PSA shall be performed, documented, and maintained according to requirements of the management system of the licensee.

2.2 PSA shall be performed according to an up to date proven methodology, taking into account international experience currently available.

3. Use of PSA

3.1 PSA shall be used to support safety management. The role of PSA in the decision making process shall be defined.

3.2 PSA shall be used[51] to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.

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[49] This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.

[50] Such as functional dependencies, area dependencies (based on the physical location of the components) and other common cause failures

[51] It is intended that such analyses will be done on a continuous basis, not just every ten years during the Periodic Safety Review.

3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects"[52].

3.4 PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.

3.5 Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.

3.6 The results of PSA shall be used to ensure that the items are included in the verification and test programmes if they contribute significantly to risk.

4. Demands and conditions on the use of PSA

4.1 The limitations of PSA shall be understood, recognized and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.

4.2 When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.

4.3 The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the SAR.

[52] Small deviations in the plant parameters that could give rise to severely abnormal plant behaviour.

Issue P: Periodic Safety Review (PSR)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Objective of the periodic safety review

1.1 The licensee shall have the prime responsibility for performing the Periodic Safety Review.

1.2 The review shall confirm the compliance of the plant with its licensing basis and any deviations shall be resolved.

1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available.

1.4 All reasonably practicable improvement measures shall be taken by the licensee as a result of the review.

1.5 An overall assessment of the safety of the plant shall be provided, and adequate confidence in plant safety for continued operation demonstrated, based on the results of the review in each area.

2. Scope of the periodic safety review

2.1 The review shall be made periodically, at least every ten years.

2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant and, as a minimum the following areas shall be covered by the review:

- Plant design as built and actual condition of systems, structures and components;
- Safety analyses and their use;
- Operating experience during the review period and the effectiveness of the system used for experience feed-back;
- Organisational arrangements;
- Staffing and qualification of staff;
- Emergency preparedness; and
- Radiological impact on the environment.

3. Methodology of the periodic safety review

3.1 The review shall use an up to date, systematic, and documented methodology, taking into account deterministic as well as probabilistic assessments.

3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices.

Issue Q: Plant modifications	
Document status: Final	Safety area: Operation

Reference levels

1. Purpose and scope

1.1 The licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely.[53]

1.2 The licensee shall control plant modifications using a graded approach with appropriate criteria for categorization according to their safety significance[54].

2. Procedure for dealing with plant modifications

2.1 The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met.

2.2 For modifications to SSC, this process shall include the following:

- Reason and justification for modification;
- Design;
- Safety assessment;
- Updating plant documentation and training;
- Fabrication, installation and testing; and
- Commissioning the modification.

3. Requirements on safety assessment and review of modifications

3.1 An initial safety assessment shall be carried out to determine any consequences for safety[55].

3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

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[53] RL 2.2 specifically addresses modifications to SSCs, all other reference levels relate to all type of modifications in the sense of IAEA NS-R-2, Para 7.1

[54] Para 4.5 of IAEA Guide NS-G-2.3 contains information about possible categories.

[55] This assessment is performed for the purpose of categorizing the intended modification according to its safety significance.

3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by personnel not immediately involved in their design or implementation.

4. *Implementation of modifications*

4.1 Implementation and testing of plant modifications shall be performed in accordance with the applicable work control and plant testing procedures.

4.2 The impact upon procedures, training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated.

4.3 Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated.

5. *Temporary modifications*[56]

5.1 All temporary modifications shall be clearly identified at the point of application and at any relevant control position[57]. Operating personnel shall be clearly informed of these modifications and of their consequences for the operation of the plant.

5.2 Temporary modifications shall be managed according to specific plant procedures.

5.3 The number of simultaneous temporary modifications shall be kept to a minimum. The duration of a temporary modification shall be limited.

5.4 The licensee shall periodically review outstanding temporary modifications to determine whether they are still needed.

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[56] Examples of temporary modifications are temporary bypass lines, electrical jumpers, lifted electrical leads, temporary trip point settings, temporary blank flanges and temporary defeats of interlocks. This category of modifications also includes temporary constructions and installations used for maintenance of the design basis configuration of the plant in emergencies or other unanticipated situations. Temporary modifications in some cases may be made as an intermediate stage in making permanent modifications. IAEA Guide NS-G-2.3, Para 6.1

[57] By relevant control position it is meant any control point important for the modified system and also any administrative aspect related to the system in which the temporary modification has been implemented.

Issue R: On-site Emergency Preparedness	
Document status: Final	Safety area: Emergency Preparedness

Reference levels

1. Objective

1.1 The licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) Regaining control of any emergency arising at their site, including events related to combinations of non-nuclear and nuclear hazards;

(b) Preventing or mitigating the consequences at the scene of any such emergency: and

(c) Co-operating with external emergency response organizations in preventing adverse health effects in workers and the public.

2. Emergency Preparedness and Response Plan

2.1 The licensee shall prepare an on-site emergency plan and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating plant activities and co-operating with external response agencies throughout all phases of an emergency.

2.2 The licensee shall provide for:

(a) Prompt recognition and classification of emergencies;

(b) Timely notification and alerting of response personnel;

(c) Ensuring the safety of all persons present on the site, including the protection of the emergency workers;

(d) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(e) Performing assessments of the situation on the technical, & radiological points of view (on and off site);

(f) Monitoring radioactive releases;

(g) Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons on site; and

(h) Plant management and damage control[58].

2.3 The site emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall

[58] Understood as urgent mitigatory repairs, controls, and other actions that are carried out, primarily at the site, while the emergency is still in progress.

also be co-ordinated with all other involved bodies and capable of extension should more improbable, severe events occur.

3. Organization

3.1 The licensee shall have people on-site at all times with the authority and responsibilities to classify and declare an emergency and, upon classification, to initiate promptly the appropriate on-site response[59].

3.2 Sufficient numbers of qualified personnel shall be available at all times for staffing appropriate positions promptly following the declaration and notification of an emergency.

3.3 Arrangements shall be made to provide technical assistance to operational staff. Teams for mitigating the consequences of an emergency (e.g. radiation protection, damage control, fire fighting, etc) shall be available.

3.4 Arrangements shall be made to alert off-site responsible authorities promptly.

3.5 The licensee shall identify those who are authorized to carry out the response functions assigned in the emergency plan.

4. Facilities and equipment

4.1 Appropriate emergency facilities shall be designated for responding to events on site and that will provide co-ordination of off-site monitoring and assessment throughout different phases of an emergency response.

4.2 An "On-site Emergency Control Centre", separated from the plant control room, shall be provided for on-site emergency management staff. Important information shall be available in the control centre about the plant and radiological conditions on and around the site. The centre shall have means of communicating with the control room, any supplementary control room, other important points on site, and with the on-site and off-site emergency response organizations[60].

4.3 Emergency facilities shall be suitably located and protected to enable the exposure of emergency workers to be controlled. Appropriate measures shall be taken to protect those occupying emergency facilities for a protracted time from hazards resulting from accidents[61].

4.4 Instruments, tools, equipment, documentation, and communication systems for use in emergencies shall be kept available and tested sufficiently frequently to demonstrate that they are in good working condition where they are unlikely to be affected by postulated accidents.

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[59] The on duty shift supervisor could be among those authorised to declare an emergency and to initiate the appropriate on-site response.

[60] The *On-site Emergency Control Centre* is the office accommodation and associated office services set aside on or near to the site for staff who are brought together to provide technical support the Operations staff during an emergency. It may have plant information systems available, but is not expected to have any plant controls.

[61] This refers, primarily, to ensuring that the *On-site Emergency Control Centre* and other locations where staff are expected to spend a significant time are located somewhere that the staff can reach and work throughout an extended emergency with minimum risk to health. This will require location away from areas that are likely to be damaged of affected by radiation fields and, where appropriate, this will include provision of recirculatory air conditioning and continuous radiation monitoring systems.

5. Training, drills and exercises

5.1 Arrangements shall be made to identify the knowledge, skills, and abilities needed for personnel to perform their assigned response functions.

5.2 Arrangements shall be made to inform all employees and all other persons present on the site of the actions to be taken in the event of an emergency.

5.3 Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that emergency response personnel meet the training obligations.

5.4 The site emergency plan shall be exercised at least annually. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned.

5.5 Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the plan shall be subject to review and updating in the light of experience gained.

Issue S: Protection against internal fires	
Document status: Final	Safety area: Emergency Preparedness

Reference levels

1. Fire safety objectives

1.1 The licensee shall implement the defence in depth principle to fire protection, providing measures to prevent fires from starting, to detect and extinguish quickly any fires that do start and to prevent the spread of fires and their effects in or to any area that may affect safety[62].

2. Basic design principles

2.1 SSCs important to safety shall be designed and located so as to minimize the frequency and the effects of fire and to maintain capability for shutdown, residual heat removal, confinement of radioactive material and monitoring of plant state during and after a fire event.

2.2 Buildings that contain SSCs important to safety shall be suitably [63], fire resistant.

2.3 Buildings that contain equipment that is important to safety shall be subdivided into compartments that segregate such items from fire loads and segregate redundant safety systems from each other[64]. When a fire compartment approach is not practicable, fire cells shall be used[65], providing a balance between passive and active means, as justified by fire hazard analysis.

2.4 Buildings that contain radioactive materials that could cause radioactive releases in case of fire shall be designed to minimize such releases.

2.5 Access and escape routes for fire fighting and operating personnel shall be available.

3. Fire hazard analysis

3.1 A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire design principles are satisfied, that the fire protection measures are appropriately designed and that any necessary administrative provisions are properly identified.

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[62] In this context, safety refers to all sources of nuclear safety risk, including radioactive waste facilities.

[63] In accordance with the results of the fire hazard analysis.

[64] A fire compartment is a building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total combustion of the fire load can occur without breaching the barriers. (Barriers comprise doors, walls, floors and ceilings.) The fire resistance rating of the barriers must be sufficiently high so that the total combustion of the fire load in the compartment can occur without breaching the barriers.

[65] In the fire cell approach the spread of fire is avoided by substituting the fire resistant barriers primarily with other passive provisions (e.g. distance, thermal insulation, etc.), that take into account all physical and chemical phenomena that can lead to propagation. Provision of active measures (e.g. fire extinguishing systems) may also be needed in order to achieve a satisfactory level of protection. The achievement of a satisfactory level of protection is demonstrated by the results of the fire hazard analysis.

3.2 The fire hazard analysis shall be developed on a deterministic basis, covering at least:

- For all normal operating and shutdown states, a single fire and consequential spread, anywhere that there is fixed or transient combustible material;
- Consideration of credible combination of fire and other PIEs likely to occur independently of a fire.

3.3 The fire hazard analysis shall demonstrate how the possible consequential effects of fire and extinguishing systems operation have been taken into account.

3.4 The fire hazard analysis shall be complemented by probabilistic fire analysis. In PSA level 1, the fires shall be assessed in order to evaluate the fire protection arrangements and to identify risks caused by fires.

4. Fire protection systems

4.1 Each fire compartment or fire cell shall be equipped with fire detection and alarm features, with detailed annunciation for the control room staff of the location of a fire. These features shall be provided with non-interruptible emergency power supplies and appropriate fire resistant supply cables.

4.2 Fixed or mobile, automated or manual extinguishing systems shall be installed. They shall be designed and located so that their rupture, spurious or inadvertent operation does not significantly impair the capability of SSCs important to safety to carry out their safety functions.

4.3 The distribution loop for fire hydrants outside building and the internal standpipes shall provide adequate coverage of areas of the plant relevant to safety. The coverage shall be justified by the fire hazard analysis.

4.4 Ventilation systems shall be arranged such that each fire compartment fully fulfils its segregation purpose in case of fire.

4.5 Parts of ventilation systems (such as connecting ducts, fan rooms and filters) that are located outside fire compartments shall have the same fire resistance as the compartment or be capable of isolation from it by appropriately rated fire dampers.

5. Administrative controls and maintenance

5.1 In order to prevent fires, procedures shall be established to control and minimize the amount of combustible materials and minimize the potential ignition sources that may affect items important to safety. In order to ensure the operability of the fire protection measures, procedures shall be established and implemented. They shall include inspection, maintenance and testing of fire barriers, fire detection and extinguishing systems.

6. Fire fighting organization

6.1 The licensee shall implement adequate arrangements for controlling and ensuring fire safety, as identified by the fire hazard analysis[66]

[66] Such arrangements must include nominating persons to be responsible for or have duties with respect to fire protection. The arrangements must set out the requirements for control of all activities that can have impact on fire safety, e.g. Maintenance; control of materials; training; tests and drills; modifications to layouts and systems – such as fire detection, fire extinguishing, ventilation, electrical and control systems.

6.2 Written emergency procedures that clearly define the responsibility and actions of staff in responding to any fire in the plant shall be established and kept up to date. A fire fighting strategy shall be developed, kept up-to date, and trained for, to cover each area in which a fire might affect items important to safety and protection of radioactive materials.

6.3 When reliance for manual fire fighting capability is placed on an offsite resource, there shall be proper coordination between the plant personnel and the off site response group, in order to ensure that the latter is familiar with the hazards of the plant.

6.4 If plant personnel are required to be involved in fire fighting, their organization, minimum staffing level, equipment, fitness requirements, and training shall be documented and their adequacy shall be confirmed by a competent person.

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Reactor Safety Reference Levels

Revision of March 2007

March 2007

WENRA Reactor Safety Reference Levels

Revision of March 2007

After some feedback and discussions with Stakeholders, based on the set of reference levels of January 2007, WENRA decided to introduce slight modifications of some reference levels to clarify their intent or avoid misunderstanding.

Appendix E	**Issue: Design Basis Envelope for Existing Reactors**
Document status: Final	Safety area: Design

Reference level :

8.2 Modified as follows :

The worst single failure[1] shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and its function remains unaffected by the PIE.

Appendix J	**Issue: System for Investigation of Events and Operational Experience Feedback**
Document status: Final	Safety area: operation

Reference level

3.1 Modified as follows :
The licensee shall report events of significance to safety in accordance with established procedures and criteria.

Appendix K	**Issue: Maintenance, in-service inspection and functional testing**
Document status: Final	Safety area: operation

Reference level

3.7 Modified as follows :
Following any event due to which the safety functions and functional integrity of any component or system may have been challenged, the licensee shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

[1] A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.

Appendix O	Issue: Probabilistic Safety Analysis (PSA)
Document status: Final	Safety area: Safety Verification

Reference level

1.3 Modified as follows :

The basic Level 1 PSA shall contain sensitivity and uncertainty analyses. The basic Level 2 PSA shall contain sensitivity analyses and, as appropriate, uncertainty analyses.

Appendix S	Issue: Protection against internal fires
Document status: Final	Safety area: Emergency Preparedness

Reference level

2 Basic design principles

2.2 The former 2.2 has been divided into 2 different reference levels :

Buildings that contain SSCs important to safety shall be suitably [2] fire resistant.

2.3 Buildings that contain equipment that is important to safety shall be subdivided into compartments that segregate such items from fire loads and segregate redundant safety systems from each other[3]. When a fire compartment approach is not practicable, fire cells shall be used[4], providing a balance between passive and active means, as justified by fire hazard analysis.

2.4 New numbering (former 2.3)

Buildings that contain radioactive materials that could cause radioactive releases in case of fire shall be designed to minimize such releases.

2.5 New numbering (former 2.4)

Access and escape routes for fire fighting and operating personnel shall be available.

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[2] In accordance with the results of the fire hazard analysis.

[3] A fire compartment is a building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total combustion of the fire load can occur without breaching the barriers. (Barriers comprise doors, walls, floors and ceilings.) The fire resistance rating of the barriers must be sufficiently high so that the total combustion of the fire load in the compartment can occur without breaching the barriers.

[4] In the fire cell approach the spread of fire is avoided by substituting the fire resistant barriers primarily with other passive provisions (e.g. distance, thermal insulation, etc.), that take into account all physical and chemical phenomena that can lead to propagation. Provision of active measures (e.g. fire extinguishing systems) may also be needed in order to achieve a satisfactory level of protection. The achievement of a satisfactory level of protection is demonstrated by the results of the fire hazard analysis.

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Reactor Safety Reference Levels

January 2007

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

WENRA Reactor Safety Reference Levels

January 2007

Issue A: Safety Policy	
Document status: Final	Safety area: Safety Management

Reference levels

1. Issuing and communication of a safety policy

1.1 A written safety policy[1] shall be issued by the licensee.

1.2 The safety policy shall be clear about giving safety an overriding priority in all plant activities.

1.3 The safety policy shall include a commitment to continuously develop safety.

1.4 The safety policy shall be communicated to all site personnel with tasks important to safety, in such a way that the policy is understood and applied.

1.5 Key elements of the safety policy shall be communicated to contractors, in such a way that licensee's expectations and requirements are understood and applied in their activities.

2. Implementation of the safety policy and monitoring safety performance

2.1 The safety policy shall require directives for implementing the policy and monitoring safety performance.

2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the plant management.

3. Evaluation of the safety policy

3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, more frequent than the periodic safety reviews.

[1] A safety policy is understood as a documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as a visible part of an integrated organisational policy.

Issue B: Operating Organisation	
Document status: Final	Safety area: Safety Management

Reference levels

1. Organisational structure

1.1 The organisational structure for safe and reliable operation of the plant, and for ensuring an appropriate response in emergencies, shall be justified[2] and documented.

1.2 The adequacy of the organisational structure, for its purposes according to 1.1, shall be assessed when organisational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated[3] after implementation.

1.3 Responsibilities, authorities, and lines of communication shall be clearly defined and documented for all staff with duties important to safety.

2. Management of safety and quality

2.1 The licensee shall ensure that the plant is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

2.2 The licensee shall ensure that decisions on safety matters are preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

2.3 The licensee shall ensure that the staff is provided with the necessary facilities and working conditions to carry out work in a safe manner.

2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are analysed in a systematic way and continuously used to improve the plant and the licensee's activities.

2.6 The licensee shall ensure that plant activities (processes) are controlled through a documented quality management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

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[2] The arguments shall be provided that the organisational structure supports safety and an appropriate response in emergencies.

[3] A verification that the implementation of the organisational change has accomplished its safety objectives.

3. Sufficiency and competency of staff

3.1 The required number of staff for safe operation[4], and their competence, shall be analysed in a systematic and documented way.

3.2 The sufficiency of staff for safe operation, their competence, and suitability for safety work shall be verified on a regular basis and documented.

3.3 A long-term staffing plan[5] shall exist for activities that are important to safety.

3.4 Changes to the number of staff, which might be significant for safety, shall be justified in advance, carefully planned and evaluated after implementation.

3.5 The licensee shall always have in house, sufficient, and competent staff and resources to understand the licensing basis of the plant (e.g. Safety Analysis Report or Safety Case and other documents based thereon), as well as to understand the actual design and operation of the plant in all plant states.

3.6 The licensee shall maintain, in house, sufficient and competent staff and resources to specify, set standards manage and evaluate safety work carried out by contractors.

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[4] Operation is defined as all activities performed to achieve the purpose for which a nuclear power plant was constructed (according to the IAEA Glossary).

[5] Long term is understood as 3-5 years for detailed planning and at least 10 years for prediction of retirements etc.

Issue C: Quality Management	
Document status: Final	Safety area: Safety Management

Reference levels

1. Objectives

1.1 Throughout the life of a nuclear power plant the licensee shall develop, implement, and maintain a documented quality management system[6] that defines the required quality and safety objectives applicable to work that is important to safety and is carried out by any organization[7], unit, or individual who can affect nuclear safety.

1.2 The quality management system shall grade the requirements set out in it to reflect their relative importance to nuclear safety with respect to each item, service, or process covered.

1.3 The quality management system shall enable the licensee to evaluate compliance with applicable nuclear safety requirements and to identify potential safety improvements.

2. Scope

2.1 Nuclear safety shall be the overriding consideration in the identification of the items, services, and processes to which the quality management system applies.

2.2 The quality management system shall ensure that the organizational structure, functional responsibilities, levels of authority, and interfaces for all organizations[8], units, and individuals who can affect nuclear safety are clearly documented and assigned.

2.3 The quality management system shall ensure that any organizational change that may affect safety is evaluated, classified with regard to its importance to safety, and justified.

3. Implementation

3.1 The most senior person representing the licensee on site shall be responsible and accountable for ensuring that an effective quality management system is being

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[6] In some IAEA Member States, the quality assurance programme is referred to as the quality assurance system or the quality system. A more recent term is "Quality Management System". IAEA is revising its main Reference SS document 50-C-SG-Q Code on QA for safety in NPPs etc to align it more with ISO 9001:2000. In Para 1.2 of the 4th draft of DS 338 it explains the new terminology it is proposing to adopt:

> *The term "Management System" has been adopted instead of "Quality Assurance". The term "Management System" reflects the evolution in the approach from the initial concept of "Quality Control" (controlling the quality of products) through "Quality Assurance" (the system to assure the quality of products) and "Quality Management" (the system to manage quality). The "Management System" is a set of interrelated or interacting elements (system) to establish policy and objectives and to achieve those objectives.*

In this Reference Level document, "quality management system" has been used in anticipation of that change whilst adhering largely to related standards from fully endorsed, rather than draft, IAEA standards.

[7] Such organizations include all those within the licensee's company as well as designers, vendors, contractors, suppliers, and service providers employed directly or indirectly on work for the licensee.

[8] Such organizations include all those within the licensee's company as well as designers, vendors, contractors, suppliers, and service providers employed directly or indirectly on work for the licensee.

implemented on site and that the senior management team is committed to and meeting its responsibility for reviewing and ensuring the success of the system.

3.2 The licensee shall establish and maintain sufficient resources and processes to define, achieve, analyse, and preserve the quality of items that are important to safety, and to take timely and effective corrective or preventive action to respond to deviations from required specifications.

3.3 The licensee shall ensure that procured items and services meet established requirements and perform as specified and that selected suppliers continue to provide acceptable items and services during the fulfilment of their procurement obligations. Licensees may delegate procurement activities to other organizations, but shall remain responsible for the overall effectiveness of these activities.

3.4 Products and processes that do not conform to specified requirements shall be identified and reported to an appropriate level of management within the organization. The safety implications of the non-conformances shall be evaluated and the actions taken shall be recorded, where appropriate.

3.5 The quality management system shall be implemented by management in collaboration[9] with those performing the work, and those assessing the work.

3.6 Work that is important to safety shall be controlled and performed using easily understood, approved current instructions, procedures, drawings, or other means, that have been appropriately validated before first use and are periodically reviewed to ensure adequacy and effectiveness.

3.7 Personnel shall be trained in requirements of the quality management system, so that they are competent to perform their assigned work and understand the safety consequences of their activities.

4. Assessment

4.1 The licensee shall assess the quality management system on a regular basis to ensure that it provides the required level of safety.

4.2 An organisational unit shall be established, or an outside agency assigned, that is responsible for independently assessing the adequacy of the quality management system and work performed. The organisational unit shall have sufficient authority and organisational freedom to carry out its responsibilities. People who conduct independent assessments shall not participate directly in the work being assessed[10].

4.3 All managers shall regularly carry out self-assessment by reviewing the processes for which they are responsible to determine their efficiency and effectiveness with establishing, promoting, and achieving nuclear safety objectives, and shall take any necessary corrective actions.

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[9] Collaboration is taken to mean that all groups are involved in the process.

[10] However, it is important that the audit team is familiar with the work being assessed. The aim of this requirement is to avoid any conflict of interest on the part of the assessor.

Issue D: Training and Authorization of NPP staff (jobs with safety importance)	
Document status: Final	Safety area: Safety Management

Reference levels

1. Policy

1.1 The licensee shall establish an overall training policy and a comprehensive training plan on the basis of long-term competency needs and training goals that acknowledges the critical role of safety. The plan shall be kept up to date.

1.2 A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

2. Competence and qualification

2.1 Only qualified persons that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The licensee shall ensure that all personnel performing safety-related duties including contractors have been adequately trained and qualified.

2.2 The Licensee shall define and document the necessary competence requirements for their staff.

2.3 Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

2.4 Staff qualifying for positions important to safety shall undergo a medical examination to ensure their fitness depending upon the duties and responsibilities assigned to them. The medical examination shall be repeated at specified intervals.

3. Training programmes and facilities

3.1 Performance based training programmes shall be established for all staff with tasks important to safety. The programmes shall cover basic training in order to qualify for a certain position and refresher training as needed.

3.2 All technical staff including on-site contractors shall have a basic understanding of nuclear safety, radiation safety, fire safety, the on-site emergency arrangements and industrial safety.

3.3 Representative simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures is effective. The simulator shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions[11].

3.4 For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator.[12]

3.5 Refresher training for control room operators shall include especially the following items as appropriate:

- Plant operation in normal operational states, selected transients and accidents;
- Shift crew teamwork;
- Operational experiences and modifications of plant and procedures.

3.6 Maintenance and technical support staff including contractors shall have practical training on the required safety critical activities.

4. Authorization

4.1 Staff controlling changes in the operational status of the plant shall be required to hold a authorization valid for a specified time period. The licensee shall establish procedures for their staff to achieve this authorization. In the assessment of an individual's competence and suitability as a basis for the authorization, documented criteria shall be used.

4.2 If an authorised individual:

- Moves to another position for which an authorization is required;
- Has been absent from the authorised position during an extended time period;

Re-authorisation shall be conducted after necessary individual preparations.

4.3 Work on safety related structures, systems, or components carried out by contractor personnel shall be approved and monitored by a suitably competent member of licensee's staff.

[11] This type of simulator is known as a full-scope simulator.
[12] Time includes the necessary briefings.

Appendix E	**Issue: Design Basis Envelope for Existing Reactors**
Document status: Final	Safety area: Design

Reference levels:

1. Objective

1.1 The design basis[13] shall have as an objective the prevention or, if this fails, the mitigation of consequences resulting from anticipated operational occurrences and design basis accident conditions. Design provisions shall be made to ensure that potential radiation doses to the public and the site personnel do not exceed prescribed limits and are as low as reasonably achievable.

2. Safety strategy

2.1 Defence-in-depth[14] shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases. The design shall therefore provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment, and an adequate protection of the barriers.

2.2 The design shall prevent as far as practicable:

- challenges to the integrity of the barriers;
- failure of a barrier when challenged;
- failure of a barrier as consequence of failure of another barrier.

3. Safety functions

3.1 The plant shall be able to fulfil the following fundamental safety functions[15]:

- control of reactivity,
- removal of heat from the core and
- confinement of radioactive material,

[13] The design basis shall be reviewed and updated during the lifetime of the plant (see ref level 11.1).
[14] Defined in the IAEA Safety Requirements NS-R-1, 2.9- 2.11. Further information is provided in INSAG-10.
[15] Under the conditions specified in the following paras.

in the plant states: normal operation, anticipated operational occurrences and design basis accident conditions.

4. Establishment of the design basis

4.1 The design basis shall specify the capabilities of the plant to cope with a specified range of plant states[16] within the defined radiation protection requirements. Therefore, the design basis shall include the specification for normal operation and transients/accident conditions from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis.

4.2 A list of PIEs shall be established to cover all events that could affect the safety of the plant. From this list, a set of design basis events shall be selected with deterministic or probabilistic methods or a combination of both, and used to set the boundary conditions according to which the structures, systems and components important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

4.3 The design basis shall be systematically defined and documented to reflect the actual plant.

5. Set of design basis events

5.1 Internal events such as loss of coolant accidents, equipment failures, maloperation and hazards, and their consequential events, shall be taken into account in the design of the plant. The list of events shall be plant specific. (see Appendix for assessment of implementation)

5.2 The following types of natural and man made external events shall as a minimum be taken into account in the design of the plant according to site specific conditions:

- extreme[17] wind loading
- extreme outside temperatures
- extreme rainfall, snow conditions and site flooding
- extreme cooling water temperatures and icing
- earthquake
- airplane crash
- other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the nuclear power plant

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[16] Normal operation, anticipated operational occurrences and design basis accident conditions.
[17] Definition of "extreme" is based on historical weather data for the site region

6. *Combination of events*

6.1 Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accident conditions, shall be considered in the design. Engineering judgement and probabilistic methods can be used for the selection of the event combinations.

7. Definition and application of technical acceptance criteria

7.1 Initiating events shall be grouped into a limited number of categories that correspond to plant states[18], according to their probability of occurrence. Radiological and technical acceptance criteria shall be assigned to each plant state such that frequent initiating events shall have only minor or no radiological consequences and that events that may result in severe consequences shall be of very low probability.

7.2 Criteria for protection of the fuel rod integrity, including fuel temperature, DNB, and cladding temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis event.

7.3 Criteria for the protection of the (primary) coolant pressure boundary shall be specified, including maximum pressure, maximum temperature, thermal- and pressure transients and stresses.

7.4 If applicable, criteria in 7.3 shall be specified as well for protection of the secondary coolant system.

7.5 Criteria shall be specified for protection of containment, including temperatures, pressures and leak rates.

8. Demonstration of reasonable conservatism and safety margins

8.1 The initial and boundary conditions shall be specified with conservatism.

8.2 The worst single failure[19] shall be assumed in the analyses of design basis events. However, it is not necessary to assume the failure of a passive component, provided it is justified that a failure of that component is very unlikely and it remains unaffected by the PIE.

8.3 Only safety systems shall be credited to carry out a safety function. Non-safety systems shall be assumed to operate only if they aggravate the effect of the initiating event[20].

8.4 A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis events[21].

8.5 The safety systems shall be assumed to operate at their performance level that is most penalising for the initiator.

[18] See footnote 16

[19] A failure and any consequential failure(s) shall be postulated to occur in any component of a safety function in connection with the initiating event or thereafter at the most unfavourable time and configuration.

[20] This means that non-safety systems are either supposed not to function after the initiator, either supposed to continue to function as before the initiator, depending on which of both cases is most penalising.

[21] This assumption is made to ensure the sufficiency of the shutdown margin. The stuck rod selected is the highest worth rod at Hot Zero Power and conservative values of reactor trip reactivity (conservative time delay and reactivity versus CR position dependence) are used. A stuck rod can be handled as single failure in the DBA-analysis if the stuck rod itself is the worst single failure.

8.6 Any failure, occurring as a consequence of a postulated initiating event, shall be regarded to be part of the original PIE.

8.7 The impact of uncertainties, which in specific cases are of importance for the results, shall be addressed in the analysis of design basis events.

9. Design of safety functions

General

9.1 The fail-safe principle shall be considered in the design of systems and components important to safety.

9.2 A failure in a system intended for normal operation shall not affect a safety function.

9.3 Activations and manoeuvring of the safety functions shall be automated or accomplished by passive means such that operator action is not necessary within 30 minutes after the initiating event. Any operator actions required by the design within 30 minutes after the initiating event shall be justified[22].

9.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components[23], redundancy, diversity[24], physical and functional separation and isolation.

Reactor shutdown functions

9.5 The means for shutting down the reactor shall consist of at least two diverse systems.

9.6 At least one of the two systems shall, on its own, be capable of quickly[25] rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.

Heat removal functions

9.7 Means for removing residual heat from the core after shutdown, and during and after anticipated operational occurrences and accident conditions, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Confinement functions

9.8 A containment system shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. This system shall include:

- leaktight structures covering all essential parts of the primary system;
- associated systems for control of pressures and temperatures;
- features for isolation;

[22] The control room staff has to be given sufficient time to understand the situation and take the correct actions. Operator actions required by the design within 30 min after the initiating event have to be justified and supported by clear documented procedures that are regularly exercised in a full scope simulator.

[23] Proven by experience under similar conditions or adequately tested and qualified.

[24] The potential for common cause failure shall be considered to determine where diversity should be applied to achieve the necessary reliability.

[25] Within 4-6 seconds, i.e. scram system.

- features for the management and removal of fission products, hydrogen, oxygen and other substances that could be released into the containment atmosphere.

9.9 Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

9.10 Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

10. Instrumentation and control systems

10.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems and the containment, and for obtaining any information on the plant necessary for its reliable and safe operation. Provision shall be made for automatic recording[26] of measurements of any derived parameters that are important to safety.

10.2 Instrumentation shall be adequate for measuring plant parameters and shall be environmentally qualified for the plant states concerned.

Control room

10.3 A control room shall be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences and design basis accidents.

10.4 Devices shall be provided to give in an efficient way visual and, if appropriate also audible indications of operational states and processes that have deviated from normal and could affect safety. Ergonomic factors shall be taken into account in the design of the control room. Appropriate information shall be available to the operator to monitor the effects of the automatic actions.

10.5 Special attention shall be given to identifying those events, both internal and external to the control room, which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

[26] By computer sampling and/or print outs.

10.6 For times when the main control room is not available, there shall be sufficient instrumentation and control equipment available, at a single location that is physically and electrically separated from the control room, so that the reactor can be placed and maintained in a shut down state, residual heat can be removed, and the essential plant parameters can be monitored.

Protection system

10.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:

- no single failure results in loss of protection function; and
- the removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

10.8 The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in operation. Exceptions shall be justified.

10.9 The design of the reactor protection system shall minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal operation and anticipated operational occurrences. Furthermore, the reactor protection system shall not prevent operators from taking correct actions if necessary in design basis accidents.

10.10 Computer based systems used in a protection system, shall fulfil the following requirements:

- the highest quality of and best practices for hardware and software shall be used;
- the whole development process, including control, testing and commissioning of design changes, shall be systematically documented and reviewed;
- in order to confirm confidence in the reliability of the computer based systems, an assessment of the computer based system by expert personnel independent of the designers and suppliers shall be undertaken; and
- where the necessary integrity of the system cannot be demonstrated with a high level of confidence, a diverse means of ensuring fulfilment of the protection functions shall be provided.

Emergency power

10.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

11. ***Review of the design basis***

11.1 The actual design basis shall regularly[27], and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach to identify needs and opportunities for improvement. Reasonably practicable measures shall be taken with respect to backfitting or other measures justified from a safety point of view.

Appendix

Interpretation of the reference level 5.1, for the purpose of benchmarking of implementation, in terms of types of internal events to be included in the safety analysis as a minimum:

The list mainly applies on PWR and BWR. For the other designs: AGR, CANDU and RBMK used in single WENRA countries, the list has to be adapted to the reactor type and implementation checked as self-assessment by the concerned country. The final list will in all cases be plant type specific.

Initiating events

- small, medium and large LOCA (break of the largest diameter piping of the Reactor Coolant Pressure Boundary)
- breaks in the main steam and main feed water systems
- forced decrease of reactor coolant flow
- forced increase or decrease of main feed water flow
- forced increase or decrease of main steam flow
- inadvertent opening of valves at the pressurizer (PWR)
- inadvertent operation of the emergency core cooling system ECCS
- inadvertent opening of valves at the steam generators (PWR)
- inadvertent opening of main steam relief/safety valves (BWR)
- inadvertent closure of main steam isolation valves
- steam generator tube rupture (PWR)
- uncontrolled movement of control rods
- uncontrolled withdrawal/ejection of control rod
- core instability (BWR)
- chemical and volume control system (CVCS) malfunction (PWR)
- pipe breaks or heat exchanger tube leaks in systems connected to the RCS and located partially outside containment (Interfacing System LOCA)
- fuel handling accidents

[27] Regularly is understood as an ongoing activity to analyse the plant and identify opportunities for improvement. The periodic safety reviews are complementary tools to verify and follow up on this activity in a longer perspective. Significant new safety information is understood as new insights gained from e.g. safety analyses and the development of safety standards and practices.

- loss of off-site power
- load drop by failure of lifting devices

Initiating events as well as consequential events (could be both types)

- fire
- explosion
- flooding

Consequential events

- missile generation, including turbine missiles
- release of fluid (oil etc) from failed systems
- vibration
- pipe whip
- jet impact

Appendix F	Issue: Design Extension of Existing Reactors
Document status: Final	Safety area: Design

Reference levels:

1. Objective

1.1 The design extension[28] analysis shall examine the performance of the plant in specified accidents beyond the design basis, including selected severe accidents, in order to minimise as far as reasonably practicable radioactive releases harmful to the public and the environment in cases of events with very low probability of occurrence.

2. Selection and analysis of Beyond Design Basis Events

2.1 Beyond design basis events shall be selected[29] and considered in the safety analysis to determine those sequences for which reasonable practicable preventive or mitigative measures can be identified and implemented (see Appendix for assessment of implementation).

2.2 Realistic assumptions and modified[30] acceptance criteria may be used for the analysis of the beyond design basis events.

3. Instrumentation for the management of beyond design basis accident conditions

3.1 Adequate instrumentation shall exist which can be used in severe accident environmental conditions in order to manage such accidents according to guidelines/procedures for severe accidents.

3.2 Necessary information from instruments shall be relayed to the control room as well as to a separately located supplementary control room/post and be presented in such a way

[28] Design extension is understood as measures taken to cope with additional events or combination of events, not foreseen in the design of the plant. Such measures need not involve application of conservative engineering practices but could be based on realistic, probabilistic or best estimate assumptions, methods and analytical criteria.
[29] Based on a combination of deterministic and probabilistic assessments as well as engineering judgement.
[30] Modified in relation to the conservative criteria used in the analysis of the design basis events.

to enable a timely assessment of the plant status and critical safety functions in severe accident conditions.

4. Protection of the containment against selected beyond design basis accidents[31]

4.1 Isolation of the containment shall be possible in a beyond design basis accident.[32] However, if an event leads to bypass of the containment, consequences shall be mitigated.

4.2 The leaktightness of the containment shall not degrade significantly for a reasonable time after a severe accident.

4.3 Pressure and temperature in the containment shall be managed in a severe accident.

4.4 Combustible gases shall be managed in a severe accident.

4.5 The containment shall be protected from overpressure in a severe accident[33].

4.6 High pressure core melt scenarios shall be prevented.

4.7 Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

Appendix

Interpretation of the reference level 2.1, for the purpose of benchmarking of implementation, in terms of types events to be analysed for design extension as a minimum, if not already considered in the design basis:

- anticipated transient without scram (ATWS)
- station black out
- total loss of feed water
- LOCA together with the complete loss of one emergency core cooling system[34]
- uncontrolled level drop during mid-loop operation (PWR) or during refuelling
- total loss of the component cooling water system
- loss of core cooling in the residual heat removal mode
- loss of fuel pool cooling
- loss of ultimate heat sink function
- uncontrolled boron dilution (PWR)
- multiple steam generator tube ruptures (PWR, PHWR)
- loss of required safety systems in the long term after a Postulated Initiating Event

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[31] These reference levels aim at providing protection at the level 4 of the defence-in-depth. Such protection could be provided by existing equipment that has been assessed, and if needed modified, to perform the relevant function in a severe accident condition or additional equipment on a best estimate basis.

[32] Special attention needs to be given for certain reactor types to the analysis of severe accident conditions with an open containment during certain shutdown states. Should such an accident occur, it should be possible to achieve timely containment isolation or implement equally effective compensatory measures. Therefore consideration has to be given to the time needed for the restoration of containment isolation and effective leaktightness, taking into account factors such as the progression of the accident sequences.

[33] This reference level could be seen as a special case of reference level 4.3. However, it is kept for clarity as a separate reference level since it might call for specific measures to protect against fast as well as slow containment overpressurization.

[34] Either the high pressure or the low pressure emergency core cooling system

Issue G: Safety Classification of Structures, Systems and Components	
Document status: Final	Safety area: Design

Reference levels

1. Objective

1.1 All SSCs[35] important for safety shall be identified and classified on the basis of their importance for safety.

2. Classification process

2.1 The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment.

2.2 The classification shall identify for each safety class:

- The appropriate codes and standards in design, manufacturing, construction and inspection;
- Need for emergency power supply, qualification to environmental conditions;
- The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;
- The quality management provisions.

3. Ensuring reliability

3.1 SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.

3.2 The failure of a SSC in one safety class shall not cause the failure of other SSCs in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.

4. Selection of materials and qualification of equipment

4.1 The design of SSCs important to safety and the materials used shall consider the effects of operational conditions over the plant lifetime and the effects of design basis accidents on their characteristics and performance.

[35] SSCs include software for I&C.

4.2 A qualification procedure shall be adopted to confirm that SSCs important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions[36] over the lifetime of the plant and when required in anticipated operational occurrences and accident conditions.

[36] Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof.

Issue H: Operational limits and conditions	
Document status: Final	Safety area: Operation

Reference levels

1. Purpose

1.1 OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intentions as documented in the SAR.

1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

2. Establishment and review of OLCs

2.1 Each established OLC shall be justified based on plant design, safety analysis and commissioning tests.

2.2 OLCs shall be kept updated and reviewed in the light of experience, developments in science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.

2.3 The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review.

3. Use of OLCs

3.1 The OLCs shall be readily accessible to control room personnel.

3.2 Control room operators shall be highly knowledgeable of the OLCs and their technical basis. Relevant operational decision makers shall be aware of their significance for the safety of the plant.

4. Scope of OLCs

4.1 OLCs shall cover all operational plant states including power operation, shutdown and refuelling, any intermediate conditions between these states and temporary situations arising due to maintenance & testing.

5. Safety limits, safety systems settings and operational limits

5.1 Adequate margins shall be ensured between operational limits and the established safety systems settings, to avoid undesirably frequent actuation of safety systems.

5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.

6. Unavailability limits

6.1 Limits and conditions for normal operation shall include limits on operating parameters, stipulation for minimum amount of operable equipment, actions to be taken by the operating staff in the event of deviations from the OLCs and time allowed to complete these actions.

6.2 Where operability requirements cannot be met, the actions to bring the plant to a safer state shall be specified, and the time allowed to complete the action shall be stated.

6.3 Operability requirements shall state for the various modes of normal operation the number of systems or components important to safety that should be in operating condition or standby condition.

7. Unconditional requirements

7.1 If operating personnel cannot ascertain that the power plant is operating within operating limits, or the plant behaves in an unexpected way, measures shall be taken without delay to bring the plant to a safe and stable state.

7.2 Plant shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

8. Staffing levels

8.1 Minimum staffing levels for shift staff shall be stated in the OLCs.

9. Surveillance

9.1 The licensee shall ensure that an appropriate surveillance[37] program is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

10. Non-compliance

10.1 In cases of non-compliance, remedial actions shall be taken immediately to re-establish OLC requirements.

10.2 Reports of non-compliance shall be investigated and corrective action shall be implemented in order to help prevent such non-compliance[38] in future.

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[37] The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the surveillance programme include not only deficiencies in SSCs and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the plant is deviating from the design intent. *(NS-G-2.6 Para 2.11)*

[38] If the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, plant shall be deemed to have operated in non-compliance with OLCs.

Issue I: Ageing Management	
Document status: Final	Safety area: Operation

Reference levels

1. Objective

1.1. The operating organisation shall have an Ageing Management Programme[39] to identify all ageing mechanisms important to safety related structures, systems and components (SSCs), determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.

2. Technical requirements, methods and procedures

2.1 The licensee shall assess structures, systems and components important to safety taking into account of relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.

2.3. The Periodic Safety Reviews shall be used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.

2.4. In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.

2.5. The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.

3. Major structures and components

3.1. Ageing management of the reactor pressure vessel[40] and its welds shall take all relevant factors including embrittlement, thermal ageing, and fatigue into account to compare their performance with prediction, throughout plant life.

3.2. Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

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[39] **Ageing** is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out).
An Ageing Management Programme (AMP) should be understood as an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components.
[40] Or its functional equivalent in other designs

Issue J: System for Investigation of Events and Operational Experience Feedback	
Document status: Final	Safety area: Operation

Reference levels

1. Programmes and Responsibilities

1.1 The licensee shall establish and conduct a programme to collect, screen, analyse, and document operating experience and events at the plant in a systematic way. Relevant operational experience and events reported by other plants shall also be considered.

1.2 Operating experience at the plant shall be evaluated to identify any latent safety relevant failures or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margin.

1.3 The licensee shall designate staff for carrying out these programmes, for the dissemination of findings important to safety and – where appropriate – for recommendations on actions to be taken. Significant findings and trends shall be reported to the licensee's top management.

1.4 Staff responsible for evaluation of operational experience and investigation into events shall receive adequate training, resources, and support from the line management.

1.5 The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

2. Collection and storage of information

2.1 The information relevant to experience from normal and abnormal operation and other important safety-related information shall be organized, documented, and stored in such a way that it can be easily retrieved and systematically searched, screened and assessed by the designated staff.

3. Reporting and dissemination of safety significant information

3.1 The licensee shall report incidents and abnormal events of significance to safety in accordance with established procedures and criteria.

3.2 Plant personnel shall be required to report abnormal events and be encouraged to report internally near misses relevant to the safety of the plant.

3.3 Information resulting from the operational experience shall be disseminated to relevant staff and shared with relevant national and international bodies.

3.4 A process shall be put in place to ensure that operating experience of events at the plant concerned as well as of relevant events at other plants is appropriately considered in the training programme for staff with tasks related to safety.

4. Assessment and investigation of events

4.1 An initial assessment of events important to safety shall be performed without delay to determine whether urgent actions are necessary.

4.2 The licensee shall have procedures specifying appropriate investigation methods, including methods of human performance analysis.

4.3 Event investigation shall be conducted on a time schedule consistent with the event significance. The investigation shall:

- Establish the complete event sequence;
- Determine the deviation;
- Include direct and root cause analysis;
- Assess the safety significance including potential consequences; and
- Identify corrective actions.

4.4 The operating organisation shall maintain liaison as appropriate with the organizations (manufacturer, research organization, designer) involved in design and construction, with the aims of feeding back information on operating experience and obtaining advice, if necessary, in case of equipment failures or abnormal events.

4.5 As a result of the analysis, timely corrective actions shall be taken such as technical modifications, administrative measures or personnel training to restore safety, to avoid event recurrence and where appropriate to improve safety.

5. Review and continuous improvement of the OEF process

5.1 Periodic reviews of the effectiveness of the OEF process based on performance criteria shall be undertaken and documented either within a self-assessment programme by the licensee or by a peer review team.

Issue K: Maintenance, in-service inspection and functional testing	
Document status: Final	Safety area: Operation

Reference levels

1. Scope and objectives

1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of SSCs important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. They shall take into account operational limits and conditions and be re-evaluated in the light of experience.

1.2 The programmes shall include periodic inspections and tests of SSCs important to safety in order to determine whether they are acceptable for continued safe operation of the plant or whether any remedial measures are necessary.

2. Programme establishment and review

2.1 The extent and frequency of preventive maintenance, testing, surveillance and inspection of SSCs shall be determined through a systematic approach on the basis of:

Their importance to safety;
Their inherent reliability;
Their potential for degradation (based on operating experience, research and vendor recommendation);
Operational and other relevant experience and results of condition monitoring.

2.2 In-service inspections of nuclear power plants shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

2.3 Data on maintenance, testing, surveillance, and inspection of SSCs shall be recorded, stored and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

2.4 The maintenance programme shall be periodically reviewed[41] in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on plant safety, and their conformance with applicable requirements.

2.5 The potential impact of maintenance upon plant safety shall be assessed.

[41] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

3. Implementation

3.1 SSCs important to safety shall be designed to be tested, maintained, repaired and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

3.2 Procedures shall be established, reviewed, and validated for maintenance, testing, surveillance and inspection tasks.

3.3 A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

3.4 Before equipment is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

3.5 The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks, shall be defined in the procedures.

3.6 Repairs to SSCs shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

3.7 Following any abnormal event due to which the safety functions and functional integrity of any component or system may have been challenged, the licensee shall identify and revalidate the safety functions and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

3.8 The reactor coolant pressure boundary shall be subject to a system leakage test before resuming operation after a reactor outage in the course of which its leak-tightness may been affected.

3.9 The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.

3.10 All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with the quality management system.

3.11 Any in-service inspection process shall be qualified[42], in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

3.12 When a detected flaw that exceeds the acceptance criteria is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall

[42] The ISI system qualification means to demonstrate that the combination of equipment, inspection procedure and personnel is appropriate for testing of a given inspection area according to a technical specification. It is recommended to uses as reference documents, eg the European Regulators Common Position on NDT Qualification, ENIQ methodology and/or IAEA – EBP-VVER-11 documents.

be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.

3.13 Surveillance measures to verify the containment integrity shall include: a) leak rate tests; b) tests of penetration seals and closure devices such as air locks and valves that are part of the boundaries, to demonstrate their leak-tightness and, where appropriate, their operability; c) inspections for structural integrity (such as those performed on liner and pre-stressing tendons).

Issue LM: Emergency Operating Procedures and Severe Accident Management Guidelines	
Document status: Final	Safety area: Operation

Reference levels

1. Objectives

1.1 A comprehensive set of emergency operating procedures (EOPs) for design basis accidents (DBAs) and beyond design basis accidents (BDBAs), and also guidelines for severe accident management (SAMG) shall be provided.

2. Scope

2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the plant state to a safe condition.

2.2 EOPs shall be provided to cover Beyond Design Basis Accidents up to, but not including, the onset of core damage. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent core damage.

2.3 SAMGs shall be provided to mitigate the consequences of severe accidents for the cases where the measures provided by EOPs have not been successful in the prevention of core damage.

2.4 EOPs for Design Basis Accidents shall be symptom-based or a combination of symptom based and event based[43] procedures. EOPs for Beyond Design Basis Accidents shall be only symptom based.

3. Format and Content of Procedures and Guidelines

3.1 EOPs shall be developed in a systematic way and shall be supported by realistic and plant specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, such as alarm response procedures and severe accident management guidelines.

3.2 EOPs shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate among EOPs and to proceed from EOPs to SAMGs.

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[43] Event-based EOPs enable the operator to identify the specific event and encompass:

- Information from significant plant parameters,
- Automatic actions that will probably be taken as a result of the event,
- Subsequent operator actions directed to returning the reactor to a normal condition or to provide for safe, extended and stable shutdown conditions.

Symptom-based EOPs enable the operator to respond to situations for which there are no procedures to identify accurately the event that has occurred. The decisions for measures to respond to such situations are specified in the procedures with respect to the symptoms and the state of systems of the plant (such as the values of safety parameters and critical safety functions).

3.3 SAMGs shall be developed in a systematic way using a plant specific approach. SAMGs shall address strategies to cope with scenarios identified by the severe accident analyses[44].

4. Verification and validation

4.1 EOPs and SAMGs shall be verified and validated in the form in which they will be used in the field, so far as practicable, to ensure that they are administratively and technically correct for the plant and are compatible with the environment in which they will be used.

4.2 The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations, using a simulator, where appropriate.

5. Review and updating of EOPs and SAMGs

5.1 EOPs and SAMGs shall be kept updated to ensure that they remain fit for their purpose.

6. Training

6.1 Shift personnel and on-site technical support shall be regularly trained and exercised, using simulators for the EOPs and, where practicable, for the SAMGs.

6.2 The transition from EOPs to SAMGs for management of severe accidents shall be exercised.

6.3 Interventions called for in SAMGs and needed to restore necessary safety functions shall be planned for and regularly exercised.

[44] Analysis aimed at identifying the plant vulnerabilities to severe accident phenomena, assessment of plant capabilities and development of accident management measures, including for containment protection as defined in Issue F (Design Extension of Existing Reactors) in RLs 4.1 to 4.7. It is understood that for these accident conditions also SAMGs shall be developed.

Issue N: Contents and updating of Safety Analysis Report (SAR)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Objective

1.1 The Licensee shall provide a SAR[45] and use it as a basis for continuous support of safe operation.

1.2 The Licensee shall use the SAR as a basis for assessing the safety implications of changes to the plant or to operating practices.

2. Content of the SAR

2.1 The SAR shall describe the site, the plant layout and normal operation; and demonstrate how safety is achieved.

2.2 The SAR shall contain detailed descriptions of the safety functions; all safety systems and safety-related structures, systems and components; their design basis and functioning in all operational states, including shut down and accident conditions.

2.3 The SAR shall identify applicable regulations codes and standards.

2.4 The SAR shall describe the relevant aspects of the plant organization and the management of safety.

2.5 The SAR shall contain the evaluation of the safety aspects related to the site.

2.6 The SAR shall outline the general design concept and the approach adopted to meet the fundamental safety objectives.

2.7 The SAR shall describe the safety analyses performed to assess the safety of the plant in response to postulated initiating events against safety criteria and radiological release limits.

2.8 The SAR shall describe the emergency operation procedures and accident management guidelines, the inspection and testing provisions, the qualification, and training of personnel, the operational experience feedback programme, and the management of ageing.

2.9 The SAR shall contain the technical bases for the operational limits and conditions.

2.10 The SAR shall describe the policy, strategy, methods, and provisions for radiation protection.

2.11 The SAR shall describe the on-site emergency preparedness arrangements and the liaison and co-ordination with off-site organizations involved in the response to an emergency.

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[45] A consistent safety document or integrated set of documents constituting the licensing basis of the plant and updated under control of the regulatory body

2.12 The SAR shall describe the on-site radioactive waste management provisions.

2.13 The SAR shall describe how the relevant decommissioning and end-of-life aspects are taken into account during operation.[46]

3. Review and update of the SAR

3.1 The licensee shall update the SAR to reflect modifications, new regulatory requirements, and relevant standards, as soon as practicable after the new information is available and applicable.

[46] Guidance on the specific aspects that need to be addressed in the SAR is given in Chapter XV of the IAEA Safety Guide GS-G-4.1.

Issue O: Probabilistic Safety Analysis (PSA)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Scope and content of PSA

1.1 For each plant design, a specific PSA shall be developed for level 1 and level 2 including all modes of operation and all relevant initiating events including internal fire and flooding. Severe weather conditions and seismic events shall be addressed[47].

1.2 PSA shall include relevant dependencies[48].

1.3 The basic Level 1 and Level 2 PSAs shall contain uncertainty and sensitivity analyses.

1.4 PSA shall be based on a realistic modelling of plant response, using data relevant for the design, and taking into account human action to the extent assumed in operating and accident procedures.

1.5 Human reliability analysis shall be performed, taking into account the factors which can influence the performance of the operators in all plant states.

2. Quality of PSA

2.1 PSA shall be performed, documented, and maintained according to the quality management system of the licensee.

2.2 PSA shall be performed according to an up to date proven methodology, taking into account international experience currently available.

3. Use of PSA

3.1 PSA shall be used to support safety management. The role of PSA in the decision making process shall be defined.

3.2 PSA shall be used[49] to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.

3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects"[50].

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[47] This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.

[48] Such as functional dependencies, area dependencies (based on the physical location of the components) and other common cause failures

[49] It is intended that such analyses will be done on a continuous basis, not just every ten years during the Periodic Safety Review.

3.4 PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.

3.5 Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.

3.6 The results of PSA shall be used to ensure that the items are included in the verification and test programmes if they contribute significantly to risk.

4. Demands and conditions on the use of PSA

4.1 The limitations of PSA shall be understood, recognized and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.

4.2 When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.

4.3 The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the SAR.

[50] Small deviations in the plant parameters that could give rise to severely abnormal plant behaviour.

Issue P: Periodic Safety Review (PSR)	
Document status: Final	Safety area: Safety Verification

Reference levels

1. Objective of the periodic safety review

1.1 The licensee shall have the prime responsibility for performing the Periodic Safety Review.

1.2 The review shall confirm the compliance of the plant with its licensing basis and any deviations shall be resolved.

1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available.

1.4 All reasonably practicable improvement measures shall be taken by the licensee as a result of the review.

1.5 An overall assessment of the safety of the plant shall be provided, and adequate confidence in plant safety for continued operation demonstrated, based on the results of the review in each area.

2. Scope of the periodic safety review

2.1 The review shall be made periodically, at least every ten years.

2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant and, as a minimum the following areas shall be covered by the review:

- Plant design as built and actual condition of systems, structures and components;
- Safety analyses and their use;
- Operating experience during the review period and the effectiveness of the system used for experience feed-back;
- Organisational arrangements;
- Staffing and qualification of staff;
- Emergency preparedness; and
- Radiological impact on the environment.

3. Methodology of the periodic safety review

3.1 The review shall use an up to date, systematic, and documented methodology, taking into account deterministic as well as probabilistic assessments.

3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices.

Issue Q: Plant modifications	
Document status: Final	Safety area: Operation

Reference levels

1. Purpose and scope

1.1 The licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely.[51]

1.2 The licensee shall control plant modifications using a graded approach with appropriate criteria for categorization according to their safety significance[52].

2. Procedure for dealing with plant modifications

2.1 The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met.

2.2 For modifications to SSC, this process shall include the following:

- Reason and justification for modification;
- Design;
- Safety assessment;
- Updating plant documentation and training;
- Fabrication, installation and testing; and
- Commissioning the modification.

3. Requirements on safety assessment and review of modifications

3.1 An initial safety assessment shall be carried out to determine any consequences for safety[53].

3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

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[51] RL 2.2 specifically addresses modifications to SSCs, all other reference levels relate to all type of modifications in the sense of IAEA NS-R-2, Para 7.1

[52] Para 4.5 of IAEA Guide NS-G-2.3 contains information about possible categories.

[53] This assessment is performed for the purpose of categorizing the intended modification according to its safety significance.

3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by personnel not immediately involved in their design or implementation.

4. Implementation of modifications

4.1 Implementation and testing of plant modifications shall be performed in accordance with the applicable work control and plant testing procedures.

4.2 The impact upon procedures, training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated.

4.3 Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated.

5. Temporary modifications[54]

5.1 All temporary modifications shall be clearly identified at the point of application and at any relevant control position[55]. Operating personnel shall be clearly informed of these modifications and of their consequences for the operation of the plant.

5.2 Temporary modifications shall be managed according to specific plant procedures.

5.3 The number of simultaneous temporary modifications shall be kept to a minimum. The duration of a temporary modification shall be limited.

5.4 The licensee shall periodically review outstanding temporary modifications to determine whether they are still needed.

[54] Examples of temporary modifications are temporary bypass lines, electrical jumpers, lifted electrical leads, temporary trip point settings, temporary blank flanges and temporary defeats of interlocks. This category of modifications also includes temporary constructions and installations used for maintenance of the design basis configuration of the plant in emergencies or other unanticipated situations. Temporary modifications in some cases may be made as an intermediate stage in making permanent modifications. IAEA Guide NS-G-2.3, Para 6.1

[55] By relevant control position it is meant any control point important for the modified system and also any administrative aspect related to the system in which the temporary modification has been implemented.

Issue R: On-site Emergency Preparedness	
Document status: Final	Safety area: Emergency Preparedness

Reference levels

1. Objective

1.1 The licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) Regaining control of any emergency arising at their site, including events related to combinations of non-nuclear and nuclear hazards;

(b) Preventing or mitigating the consequences at the scene of any such emergency: and

(c) Co-operating with external emergency response organizations in preventing adverse health effects in workers and the public.

2. Emergency Preparedness and Response Plan

2.1 The licensee shall prepare an on-site emergency plan and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating plant activities and co-operating with external response agencies throughout all phases of an emergency.

2.2 The licensee shall provide for:

(a) Prompt recognition and classification of emergencies;

(b) Timely notification and alerting of response personnel;

(c) Ensuring the safety of all persons present on the site, including the protection of the emergency workers;

(d) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(e) Performing assessments of the situation on the technical, & radiological points of view (on and off site);

(f) Monitoring radioactive releases;

(g) Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons on site; and

(h) Plant management and damage control[56].

2.3 The site emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall

[56] Understood as urgent mitigatory repairs, controls, and other actions that are carried out, primarily at the site, while the emergency is still in progress.

also be co-ordinated with all other involved bodies and capable of extension should more improbable, severe events occur.

3. Organization

3.1 The licensee shall have people on-site at all times with the authority and responsibilities to classify and declare an emergency and, upon classification, to initiate promptly the appropriate on-site response[57].

3.2 Sufficient numbers of qualified personnel shall be available at all times for staffing appropriate positions promptly following the declaration and notification of an emergency.

3.3 Arrangements shall be made to provide technical assistance to operational staff. Teams for mitigating the consequences of an emergency (e.g. radiation protection, damage control, fire fighting, etc) shall be available.

3.4 Arrangements shall be made to alert off-site responsible authorities promptly.

3.5 The licensee shall identify those who are authorized to carry out the response functions assigned in the emergency plan.

4. Facilities and equipment

4.1 Appropriate emergency facilities shall be designated for responding to events on site and that will provide co-ordination of off-site monitoring and assessment throughout different phases of an emergency response.

4.2 An "On-site Emergency Control Centre", separated from the plant control room, shall be provided for on-site emergency management staff. Important information shall be available in the control centre about the plant and radiological conditions on and around the site. The centre shall have means of communicating with the control room, any supplementary control room, other important points on site, and with the on-site and off-site emergency response organizations[58].

4.3 Emergency facilities shall be suitably located and protected to enable the exposure of emergency workers to be controlled. Appropriate measures shall be taken to protect those occupying emergency facilities for a protracted time from hazards resulting from accidents[59].

4.4 Instruments, tools, equipment, documentation, and communication systems for use in emergencies shall be kept available and tested sufficiently frequently to demonstrate that they are in good working condition where they are unlikely to be affected by postulated accidents.

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[57] The on duty shift supervisor could be among those authorised to declare an emergency and to initiate the appropriate on-site response.

[58] The *On-site Emergency Control Centre* is the office accommodation and associated office services set aside on or near to the site for staff who are brought together to provide technical support the Operations staff during an emergency. It may have plant information systems available, but is not expected to have any plant controls.

[59] This refers, primarily, to ensuring that the *On-site Emergency Control Centre* and other locations where staff are expected to spend a significant time are located somewhere that the staff can reach and work throughout an extended emergency with minimum risk to health. This will require location away from areas that are likely to be damaged of affected by radiation fields and, where appropriate, this will include provision of recirculatory air conditioning and continuous radiation monitoring systems.

5. Training, drills and exercises

5.1 Arrangements shall be made to identify the knowledge, skills, and abilities needed for personnel to perform their assigned response functions.

5.2 Arrangements shall be made to inform all employees and all other persons present on the site of the actions to be taken in the event of an emergency.

5.3 Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that emergency response personnel meet the training obligations.

5.4 The site emergency plan shall be exercised at least annually. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned.

5.5 Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the plan shall be subject to review and updating in the light of experience gained.

Issue S: Protection against internal fires	
Document status: Final	Safety area: Emergency Preparedness

Reference levels

1. Fire safety objectives

1.1 The licensee shall implement the defence in depth principle to fire protection, providing measures to prevent fires from starting, to detect and extinguish quickly any fires that do start and to prevent the spread of fires and their effects in or to any area that may affect safety[60].

2. Basic design principles

2.1 SSCs important to safety shall be designed and located so as to minimize the frequency and the effects of fire and to maintain capability for shutdown, residual heat removal, confinement of radioactive material and monitoring of plant state during and after a fire event.

2.2 Buildings that contain equipment that is important to safety shall be designed as fire resistant, subdivided into compartments that segregate such items from fire loads and segregate redundant safety systems from each other[61]. When a fire compartment approach is not practicable, fire cells shall be used[62], providing a balance between passive and active means, as justified by fire hazard analysis.

2.3 Buildings that contain radioactive materials that could cause radioactive releases in case of fire shall be designed to minimize such releases.

2.4 Access and escape routes for fire fighting and operating personnel shall be available.

3. Fire hazard analysis

3.1 A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire design principles are satisfied, that the fire protection measures are appropriately designed and that any necessary administrative provisions are properly identified.

3.2 The fire hazard analysis shall be developed on a deterministic basis, covering at least:

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[60] In this context, safety refers to all sources of nuclear safety risk, including radioactive waste facilities.

[61] A fire compartment is a building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total combustion of the fire load can occur without breaching the barriers. (Barriers comprise doors, walls, floors and ceilings.) The fire resistance rating of the barriers must be sufficiently high so that the total combustion of the fire load in the compartment can occur without breaching the barriers.

[62] In the fire cell approach the spread of fire is avoided by substituting the fire resistant barriers primarily with other passive provisions (e.g. distance, thermal insulation, etc.), that take into account all physical and chemical phenomena that can lead to propagation. Provision of active measures (e.g. fire extinguishing systems) may also be needed in order to achieve a satisfactory level of protection. The achievement of a satisfactory level of protection is demonstrated by the results of the fire hazard analysis.

- For all normal operating and shutdown states, a single fire and consequential spread, anywhere that there is fixed or transient combustible material;
- Consideration of credible combination of fire and other PIEs likely to occur independently of a fire.

3.3 The fire hazard analysis shall demonstrate how the possible consequential effects of fire and extinguishing systems operation have been taken into account.

3.4 The fire hazard analysis shall be complemented by probabilistic fire analysis. In PSA level 1, the fires shall be assessed in order to evaluate the fire protection arrangements and to identify risks caused by fires.

4. Fire protection systems

4.1 Each fire compartment or fire cell shall be equipped with fire detection and alarm features, with detailed annunciation for the control room staff of the location of a fire. These features shall be provided with non-interruptible emergency power supplies and appropriate fire resistant supply cables.

4.2 Fixed or mobile, automated or manual extinguishing systems shall be installed. They shall be designed and located so that their rupture, spurious or inadvertent operation does not significantly impair the capability of SSCs important to safety to carry out their safety functions.

4.3 The distribution loop for fire hydrants outside building and the internal standpipes shall provide adequate coverage of areas of the plant relevant to safety. The coverage shall be justified by the fire hazard analysis.

4.4 Ventilation systems shall be arranged such that each fire compartment fully fulfils its segregation purpose in case of fire.

4.5 Parts of ventilation systems (such as connecting ducts, fan rooms and filters) that are located outside fire compartments shall have the same fire resistance as the compartment or be capable of isolation from it by appropriately rated fire dampers.

5. Administrative controls and maintenance

5.1 In order to prevent fires, procedures shall be established to control and minimize the amount of combustible materials and minimize the potential ignition sources that may affect items important to safety. In order to ensure the operability of the fire protection measures, procedures shall be established and implemented. They shall include inspection, maintenance and testing of fire barriers, fire detection and extinguishing systems.

6. Fire fighting organization

6.1 The licensee shall implement adequate arrangements for controlling and ensuring fire safety, as identified by the fire hazard analysis[63]

6.2 Written emergency procedures that clearly define the responsibility and actions of staff in responding to any fire in the plant shall be established and kept up to date. A fire fighting strategy shall be developed, kept up-to date, and trained for, to cover each area

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[63] Such arrangements must include nominating persons to be responsible for or have duties with respect to fire protection. The arrangements must set out the requirements for control of all activities that can have impact on fire safety, e.g. Maintenance; control of materials; training; tests and drills; modifications to layouts and systems – such as fire detection, fire extinguishing, ventilation, electrical and control systems.

in which a fire might affect items important to safety and protection of radioactive materials.

6.3 When reliance for manual fire fighting capability is placed on an offsite resource, there shall be proper coordination between the plant personnel and the off site response group, in order to ensure that the latter is familiar with the hazards of the plant.

6.4 If plant personnel are required to be involved in fire fighting, their organization, minimum staffing level, equipment, fitness requirements, and training shall be documented and their adequacy shall be confirmed by a competent person.

Western European
WENRA
Nuclear Regulators Association

WENRA

Report on Harmonization of Reactor Safety in WENRA Countries

January 2006

Western European
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Nuclear Regulators' Association

Harmonization of Reactor Safety in WENRA Countries

Report by

WENRA Reactor Harmonization Working Group

January 2006

RHWG
Reactor Harmonization Working Group

Harmonization of Reactor Safety in WENRA Countries

Contents

WENRA Policy Statement

We, the heads of the national Nuclear Safety Authorities, members of WENRA, commit ourselves to a continuous improvement of nuclear safety in our respective countries.

Nuclear safety and radiation protection are based on the principle of the prime responsibility of the operators. The role of national regulators is to ensure that this responsibility is fully secured, in compliance with the regulatory requirements.

In order to work together, we created the Western European Nuclear Regulators' Association (WENRA) with the following main objectives:

- *to build and maintain a network of chief nuclear safety regulators in Europe;*
- *to promote exchange of experience and learning from each other's best practices;*
- *to develop a harmonized approach to selected nuclear safety and radiation protection issues and their regulation, in particular within the European Union;*
- *to provide the European Union Institutions with an independent capability to examine nuclear safety and its regulation in Applicant Countries.*

In order to develop a harmonized approach, we are:

- *sharing our experience feedback and our vision;*
- *making efforts to further exchange of personnel, allowing an in-depth knowledge of working methods of each other;*
- *developing common reference safety levels in the fields of reactor safety, decommissioning safety, radioactive waste and spent fuel management facilities in order to benchmark our national practices.*

We recognise the IAEA standards form a good basis for the continuous improvement of national nuclear regulatory systems and nuclear safety.

The reference levels that we have developed represent good practices in our counties from which we can also seek to learn from each other to further improve nuclear safety and its regulation. Hence, we are committed:

- *by the year of 2010 to improve and harmonize our nuclear regulatory systems, using as a minimum, the reference levels;*
- *to influence the revision of the IAEA standards when appropriate;*
- *to regularly revise the reference levels when new knowledge and experience are available.*

We strive for openness and improvement of our work. For that purpose we will:

- *keep the European Nuclear Safety and Radiation Protection Bodies not belonging to WENRA, and the EU Institutions, informed of the progress made in our work;*
- *make our public reports available on the Internet (www.wenra.org);*
- *invite stakeholders to make comments and suggestions on these reports.*

Signed in Stockholm December 2005

Executive Summary

One of the aims of the Western European Nuclear Regulators' Association (WENRA) that now comprises the Chief Regulators of 17 European Nuclear Regulatory Authorities is to develop a harmonized approach to reactor safety. To achieve this objective the Reactor Harmonization Working Group (RHWG) was set up and undertook a pilot study to develop a methodology[1], which was then used by WENRA to establish terms of reference for RHWG's main study reported here.

The RHWG used the following understanding of harmonization:

> *No substantial differences between countries from the safety point of view in generic, formally issued, national safety requirements and in their resulting implementation on nuclear power plants.*

The safety areas and issues included in the study were selected to cover important aspects of reactor safety where differences in substance between WENRA countries might be expected. They did not seek to cover everything that could have an impact upon safety or to judge the overall level of safety in existing plants.

A methodology was developed in five main steps:

1. A set of Reference Levels identifying the main relevant requirements on reactor safety was developed for 18 safety issues. These Reference Levels were primarily based on IAEA safety standards;
2. Countries assessed themselves against the Reference Levels on both the legal[2] and implementation side and documented their national position;
3. The national positions were scrutinized in peer review panel sessions to validate the self-assessments;
4. Where judged necessary, changes were made to national assessments and, in some cases, Reference Levels were modified;
5. Areas where harmonization was considered necessary on the implementation and/or legal side in each country were identified.

The study indicates that the majority of the Reference Levels are implemented in nuclear power plants in WENRA countries; however, the implementation results need to be further validated. The study also shows that there is a significant amount of work to do to align the national requirements with the Reference Levels, in view of the very strict harmonization definition. It appears that for full harmonization, all countries have some work to do both on their regulations and on the implementation of the Reference Levels.

The group's work signifies a considerable amount of effort and commitment by the participating organizations over a period of two and a half years. This work is considered to be an important input for reactor safety harmonization on existing nuclear power plants, and some participating countries are already using it to develop or revise their regulatory requirements. The results should be seen as a step towards meeting WENRA's commitment to achieving continuous improvement of reactor safety within Europe through mutual learning, and they offer the potential for further developments.

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[1] Pilot Study on Harmonization of Reactor Safety in WENRA Countries Abstract. WENRA Working Group on Reactor Harmonization, March 2003.

[2] References to the 'legal side' requirements relate to application of the qualification given on Page 3 for national requirements to have a formal basis and be formally issued.

1. Introduction

In November 1999, the Western European Nuclear Regulators' Association (WENRA)[3] set up a group to stimulate discussion within the Association on how to harmonize reactor safety in participating countries. The objectives were:

- To create a common understanding among WENRA members on any significant differences in substance that may exist between countries with regard to safety requirements for existing reactors of different design generations; and
- To suggest appropriate steps, if necessary, to move towards a harmonized approach to reactor safety.

Between 2000 and 2002, a pilot study was performed to develop and test a methodology for systematic comparison of national requirements on selected reactor safety issues[4]. Nine countries were involved in this study: Belgium, Finland, France, Germany, Italy, Netherlands, Spain, Sweden, and UK. The objectives of the pilot study were met, and the methodology proved to be suitable for its purpose.

WENRA issued a mandate to extend the work for the safety issues relevant for harmonization of reactor safety and, in 2003, extended its membership to the Chief Regulators of the regulatory authorities of Bulgaria, Czech Republic, Hungary, Lithuania, Romania, Slovakia, and Slovenia. All countries represented in WENRA, i.e. all aforementioned countries plus Switzerland, took part in this study.

To achieve the task and to co-ordinate the necessary actions within the participating organizations, a working group was formed (Reactor Harmonization Working Group, RHWG) with representatives from all 17 countries (Annex 4). It has taken the RHWG over two and a half years to complete its assigned task.

This report concludes the work performed according to the above mandate. It includes terms of reference, scope covered, methodology used, and the overall results.

2. Terms of Reference

The following boundary conditions were applied for the study[5]:

- Given WENRA members' responsibilities, the study should cover nuclear power reactor safety, excluding radiation protection and physical protection;
- The study should address existing operating power reactors;
- The study should address principal differences and similarities in substance of safety requirements in the areas of deterministic and probabilistic requirements, as well as safety management and safety culture;
- The study should not go into legal and technical details; and

[3] The Chief Regulators of the national nuclear safety authorities that have at least one nuclear power reactor in operation or being decommissioned.

[4] Pilot Study on Harmonization of Reactor Safety in WENRA Countries Abstract. WENRA Working Group on Reactor Harmonization, March 2003.

[5] The Terms of Reference are based on the Pilot Study and on proposals from the Working Group, both endorsed by WENRA in 2002.

- As a first step towards harmonization, the study should concentrate on safety requirements that are placed by the regulatory regime upon the licensee, but should not deal with regulatory practices, such as regulatory assessment and review criteria for safety cases.

The following understanding of harmonization was agreed for the study:

> ***No substantial differences between countries from the safety point of view in generic, formally issued, national safety requirements, and in their resulting implementation on Nuclear Power Plants.***

This implies that both the legal and implementation sides have to be considered by the study. It is not judged sufficient to harmonize safety based on voluntary or other less formal agreements with the industry, because there are no guarantees that such agreements will withstand an environment with changing organizational and economic challenges.

Based on the methodology developed for the pilot study, the following definition of a 'national requirement' was used:

To qualify, a national requirement must be part of the legal regulatory system and be formally issued. It must be documented in an official, open document/publication. These requirements are of two types, both of which provide a basis for regulators to exercise their powers and duties, but at different levels:

- ***A legally binding requirement****, such as a law, ordinance or regulation that is mandated and enforced, if necessary with the use of legal sanctions. These requirements are issued by the parliament, government, or regulatory body as authorized; and*
- ***A general recommendation*** *(rule, condition, guideline, principle, standard, etc) that the regulatory body issues formally with reference to a legally binding document, decision, permission, or other formal authorization. These are not legally binding and enforced like regulations; however, they are used for granting licences and regulating licensees' activities.*

Documents issued by licensees were not accepted as valid national requirements. Specific regulatory decisions that are legally binding and documented but do not address all licensees equally are also excluded.

National requirements that are more demanding and implementation that is more extensive than Reference Levels specify will be regarded as harmonized. It is not anticipated that countries would relax them to 'improve' harmonization[6]. Annex 3 gives details about the legal systems and corresponding 'national requirements' used for countries' self-assessments.

3. Safety areas and issues

The safety issues have been selected with the objective of covering the most important aspects where differences in substance between WENRA countries might be expected. The list of safety issues does not define reactor safety: its objective is to provide guidance on what is currently significant for harmonization in the context of this study. This means that issues that are not important for harmonization have

[6] It is not WENRA's primary goal to develop safety standards for the European Union.

been excluded, since few safety-significant differences between participating countries are expected with them.

On this basis, the list of issues shown in **Table 1** was endorsed by WENRA for the main study. Several reference sources were used to generate it, such as:

- The Convention on Nuclear Safety;
- Existing national safety regulations;
- The IAEA Safety Requirements on Design and Operation;
- Studies of national practices within the framework of OECD/NEA, and EU;
- Current or forthcoming regulatory challenges, such as risk-informed applications; and
- A proposal from the industry group behind the European Utility Requirements[7].

Safety area		Safety issue
Safety Management	A	Safety policy
	B	Operating organization
	C	Quality management
	D	Training and Authorization of NPP staff (jobs with safety importance)
Design	E	Verification and improvement of the design
	F	Design basis envelope for existing reactors
	G	Safety classification of structures, systems and components
Operation	H	Operational limits and conditions
	I	Ageing management
	J	System for investigation of events and operational experience feedback
	K	Maintenance, in-service inspection and functional testing
	LM	Emergency Operating procedures and severe accident management guidelines
Safety verification	N	Contents and updating of safety analysis report (SAR)
	O	Probabilistic safety analysis (PSA)
	P	Periodic safety review (PSR)
	Q	Plant modifications
Emergency preparedness	R	On-site emergency preparedness
	S	Fire protection against internal fires

Table 1: Safety areas and issues

For clarity, the "safety issues" have been structured into five "safety areas" that correspond closely to the Convention on Nuclear Safety and the structure used by IAEA, as well as to the structure of many national regulations.

The pilot study covered Issues A, B, E, O, P, and the severe accident management aspects of Issue LM; the extended group revised and benchmarked all these issues again.

4. Process

The study used a process that had evolved from the pilot study:

1. A set of Reference Levels identifying the main relevant requirements for harmonization of reactor safety was developed for the 18 safety issues in Table 1. These Reference Levels were primarily based on IAEA safety standards;

[7] Meeting between representatives from EUR and WENRA 03/2002.

2. Countries assessed themselves against these Reference Levels on both the legal and implementation side and documented their position;
3. The national positions were scrutinized in peer review panel sessions to validate the self-assessments;
4. Where judged necessary, changes were made to national assessments and, in some cases, Reference Levels were modified;
5. Areas where harmonization was considered necessary on the legal and/or implementation side in each country were identified.

Figure 1 provides an overview of the process described in subsequent sections.

4.1 Establishment of "Reference Levels"

As a basis for the Reference Levels, the most recent publicly available edition of the IAEA safety standards was used. For each issue, a lead country was appointed to propose a first draft by:

- Checking the IAEA safety standards associated with the safety issue under study;
- Selecting the significant (key) IAEA requirements (from the Requirements documents); and
- Adding any significant (key) IAEA recommendations (from the Safety Guides).

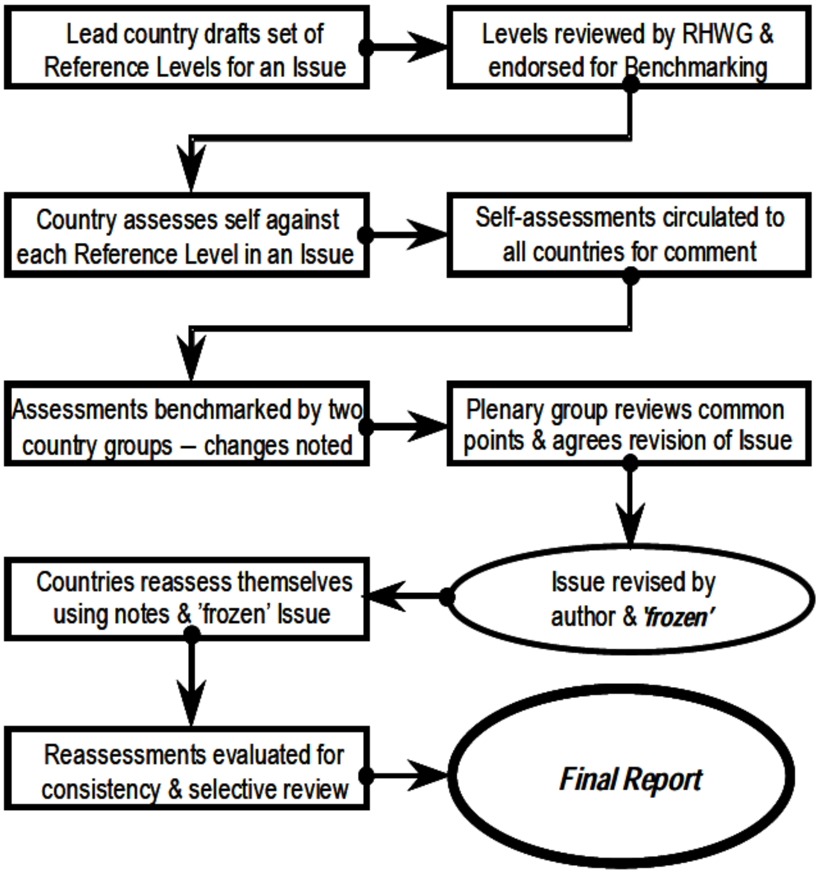

Figure 1: Process summary

The lead country circulated a draft set of Reference Levels for an issue to members for comment before the meeting at which the working group discussed it. In particular, the working group focused on the need for common understanding and the practicality of benchmarking the Reference Levels, the amount of detail in them, and their value for harmonization.

The Reference Levels are all formulated as "shall" statements. This was decided during the pilot study, to ensure that all Reference Levels were treated the same, irrespective of the underlying source of the material (IAEA Requirements or Guides). The use of "shall" indicates that all levels have to be benchmarked, and action will need to be taken for harmonization if necessary. A legally binding generic national requirement or a formally issued general recommendation, as defined in Section 2, is equally acceptable evidence for meeting the levels. Thus, the levels themselves are not regulations[8], but provide a means for judging whether there are national requirements in WENRA member countries to meet them and whether countries have implemented them.

Each country also had an opportunity to propose additional Reference Levels, not based on the IAEA, but on their national regulations (so called "European deltas"). The criteria agreed for accepting a "European delta" were that:

- The proposed "delta" was supported by a national requirement that applies to existing reactors;
- The proposed "delta" required more than, or was more extensive than, the corresponding IAEA requirements or recommendations; and
- The working group agreed, either by consensus or by majority vote (one vote per country), that the addition of the proposed "delta" benefited safety and harmonization.

In practice, there have been very few such additions, because international safety standards in IAEA documents are rising and many WENRA countries have participated in the work of IAEA committees, which has contributed to convergence.

Each Reference Level document underwent various redrafts before endorsement for benchmarking. Some Reference Levels are quite general, others more detailed, but the aim was to focus on substantial differences and similarities, in the interest of harmonization, and to avoid technical detail.

4.2 National self-assessments

Each country completed a self-assessment table for each of the 18 issues by responding to two questions:

(i) Is there an equivalent national requirement that meets the substance of the Reference Level?

(ii) Have all operating Nuclear Power Plants in the country implemented the Reference Level?

It is possible to answer 'no' to the first question and 'yes' to the second or vice versa, and there are three possible coded results for each question:

A. Yes – already harmonized in substance;

B. No – a difference exists, but can be justified from a safety point of view; or

C. No – a difference exists, and should be addressed for harmonization.

Countries documented their national position by highlighting keywords and coding their response to each question '**A**', '**B**', or '**C**'. Thus, two letters summarize the answers for each Reference Level: the first gives the result for question (i) above, relating to the legal requirement, and the second for question (ii) relating to implementation.

[8] As mentioned in Section 2, it is not WENRA's primary goal to develop safety standards for the European Union. The Reference Levels are to be seen as a tool to improve safety. In addition, Reference Levels should not be interpreted as a means of weakening the deterministic approach and the defence-in-depth principle. For instance, PSA should always be used to supplement a combined approach.

It should be noted that an ‘**A**’ assessment can be achieved for implementation on NPPs, even when there are no formally issued, public, generic, national requirements, and this can be for many reasons.

Code ‘**B**’ reflects a safety-related difference that has been justified and does not need to be addressed further for harmonization. For consistency, the following criteria were agreed for justification of a ‘**B**’:

- Regulations are under development or revision and will cover the Reference Level(s) by the end of 2005 at the latest;
- The Reference Level is covered sufficiently by alternative national requirements;
- Specifically for issue N: The Reference Level is covered by another controlled safety document than the SAR, but which has a similar status to the SAR, i.e. it is a document that is approved by the regulatory body and included in the licensing documentation;
- Implementation is lacking with respect to a Reference Level in an older plant for which a shut down decision has been taken;
- Implementation with respect to a Reference Level is in progress and will be completed before the end of 2005; or
- Implementation has been exempted with respect to a Reference Level on the basis of a technical justification that has been accepted by the regulatory body.

Code ‘**C**’ can have quite different meanings. It can result from failure to match a single keyword from a Reference Level or failure to meet an entire Reference Level.

Assessment results that are coded ‘**C**’ will form the basis of national action plans.

4.3 Review and validation of national positions

Self assessment

Each country tabulated their own data and ‘validated’ their own assessments in the light of both their own knowledge and the nature of the questioning that would take place in both the subsequent panel validation process carried out by the working group and the quality checks described below.

Panel benchmarking

At working group meetings, countries divided into two groups and worked in parallel, scrutinizing and questioning each other’s justifications of national positions. The objectives of the panel benchmarking sessions were:

- To achieve consistency between the countries and, in particular, to check that the Reference Levels had been interpreted the same way and equally rigorously;
- To increase the overall reliability of the result of the study by ensuring that national positions met the agreed justifications; and
- To provide guidance for countries on how they could revise their assessments.

If the group judged that the justification provided by the country was not sufficient for the proposed coding, the country was asked to provide more evidence or change its coding. (In the meantime, countries applied a '**C**'.)

After the panel benchmarking session, a plenary meeting took place to consider any problems with respect to the wording of the Reference Levels and agree a common understanding on them and/or the allocation of codes. Document authors then used these to amend and 'finalize' the corresponding Reference Levels. Countries then used the amended versions and comments on their evidence from panel sessions to update their self-assessments. (Annex 2 gives more detail.)

All countries have been benchmarked against all Reference Levels, and all Reference Levels have been reviewed and scrutinized thoroughly by these processes.

Quality checks

Because benchmarking had been carried out over 18 months and later self-assessment updates used revised Reference Level documents, various quality checks were carried out to help confirm that panel comments had been included and that the results presented were self-consistent. It had also been noticed towards the end of the benchmarking cycle that standards for evidence had become stricter, and statements needed improvement in many cases, so several final checks were carried out and consequential changes made:

1. Each country revised their assessments using more specific written guidance;
2. Updated assessments were checked for consistency independently by nominated countries;
3. The secretariat crosschecked final assessments before finalizing the report; and
4. Members used the final meeting of the working group to carry out a final review of the Reference Levels and to carry out spot-checks of their evidence.

5. Overview of results

Results have been analysed and presented graphically to illustrate the proportion of '**A**'s, '**B**'s, and '**C**'s. As described in Section 4.2, code '**A**' means that national legal requirements are substantially equivalent to a Reference Level; code '**B**' means that any differences have been justified; and code '**C**' identifies differences that need addressing for harmonization.

The next two sub-sections and the Appendix provide pairs of graphs; the first (sub-referenced (a)) all relate to the existence of formal national legal requirements and the second (sub-referenced (b)) to implementation on all of the associated country's NPPs. The graphs all use the following colour scheme throughout:

Code A – Already harmonized in substance;

Code B – A difference exists, but can be justified from a safety point of view; and

Code C – A difference exists, and should be addressed for harmonization.

5.1 General overview

Fig 2(a) summarizes the legal requirement data from all 17 countries for all 18 issues and Fig 2(b) the complementary results for the implementation status in NPPs.

Figure 2(a) indicates that the Reference Levels are formally required for over half of the assessments, and 4% have differences that can be justified. Hence, for

Figure 2(a):
Overview - All legal requirement results

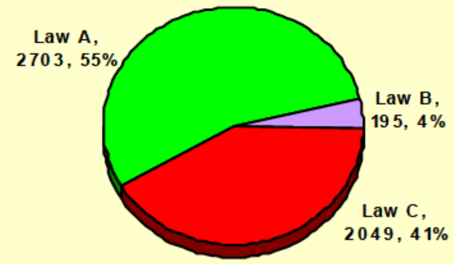

Figure 2(b):
Overview - All implementation results

harmonization on the legal side, there is a need for a significant number of additional, formally issued, generic national requirements.

For implementation, Fig 2(b) shows that the situation is different: Reference Levels are implemented for 88% of the assessments and 6% could be justified, leaving just 6% needing to be implemented for harmonization.

Thus, the results indicate that a significantly higher number of Reference Levels are implemented than are currently explicitly legally required by the criteria for the study. The reasons could include licensees responding to regulatory decisions or practices that do not meet the definition of national requirements, as explained in Section 2, or responses to other legal requirements and less formal agreements between the regulator and the licensee. Licensees also act on their own initiative under their responsibility for safety.

5.2 Grand totals by topic

Figures 3(a) and 3(b) give the overall situation for all countries on the 18 safety

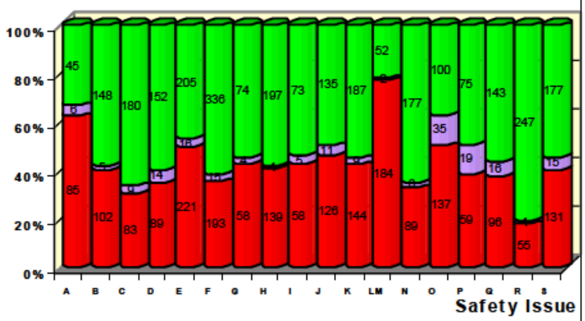

Figure 3(a): **Law: Grand totals by topic (%)**

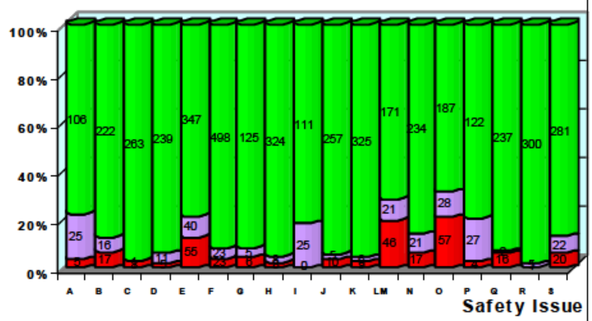

Figure 3(b): **Imp: Grand totals by topic (%)**

issues. At one extreme, for the two issues Safety Policy and Emergency Operating

Procedures/Severe Accident Management Guidelines, only about 20-30% of the assessments are coded '**A**'. On the other hand, for On-site Emergency Preparedness 80% of the assessments are '**A**'. Other issues show that 40 to 60% of the assessments are '**A**'. The largest numbers of justified differences ('**B**') occur with *PSA* (Issue O) and *PSR* (Issue P), each with about 15% of the assessments.

For implementation, the situation is quite different. For the *Emergency Operating Procedures/Severe Accident Management Guidelines* (Issue LM), *PSA* (Issue O) and *Verification and Improvement of Design* (Issue E), about 80% of the assessments showed implementation. Larger percentages of justified differences (10-20%) are found for *Safety Policy* (Issue A), *Ageing Management* (Issue I), *PSA* (Issue O), *PSR* (Issue P), and *Fire Protection* (Issue S). Implementation was generally found to be very good for *Quality Management* (Issue C), *Training and Authorization* (Issue D), *Design Basis Envelope* (Issue F), *Safety Classification* (Issue G), *OLCs* (Issue H), and *Maintenance and In-service Inspection* (Issue K). For *On-site Emergency Preparedness* (Issue R), all assessments show that the Reference Levels are implemented, with one exception and some justified differences.

Further details are provided in the Appendix.

6. Comments on the process

The Working Group has identified some limitations and uncertainties, but believes that these do not undermine the credibility of the study as a whole, and considers that the study meets the aims of the Terms of Reference.

6.1 Scope

Despite the Terms of Reference constraints to avoid covering radiation protection, physical protection, and legal and technical detail, some radiological aspects have been included with regard to radiological criteria for design, periodic safety review, and on-site emergency preparedness because they relate so closely to nuclear safety.

6.2 Agreement of Reference Levels

Participants used expert judgement from many years of regulatory experience to select and refine Reference Levels. The levels themselves were only finalized after several iterations. A challenge for the group was to define sufficient detail for Reference Levels without making them too complicated.

6.3 Panel procedure and validation

Legal Requirements acceptance criteria

Benchmarking has taken place over a period of 18 months and the acceptance criteria for an '**A**' on the legal side, i.e. no substantial differences between a Reference Level and a national requirement, have changed slightly over time. Earlier benchmarks looked for substantial similarities, and accepted detailed wording differences and national requirement that did not cover all aspects of the Reference Level. Over time, it proved difficult to be consistent in these judgements, so panels sought key words from the Reference Levels in the corresponding national requirements. If some keywords were missing, the '**A**' became a '**B**' if the shortfall could be justified or, more often, a '**C**' if not. This change has led to an increased number of '**C**'s, because many Reference Levels contain several aspects which are looked for now in national requirements. In the final stage of the project, all participants revised their earlier benchmarks to apply the more rigorous criteria and ensure that the best evaluation possible had been made.

Implementation status

The Pilot Study reported that assessments of Reference Level implementation were quite uncertain because there was little time or resource for verification. In principle, the same applies to the Main Study, which relies on regulatory knowledge. In most cases, this knowledge comes from inspection and review of safety documentation, although the Main Study allowed regulators to approach operators for additional information about any specific Reference Level, if appropriate: some members have consulted to some extent. To improve the reliability of the implementation justification, more evidence was called for in the self-assessments than had been accepted initially. This took the form of references to site documentation or regulatory interfaces that confirmed the position; however, it was not possible in the framework of the study for panels to verify implementation with the same rigour as for legal requirements.

Regulatory practice

The Working Group did not evaluate national approaches to regulations per se. The Terms of Reference excluded regulatory practices such as enforcement strategies and acceptance criteria for inspections and safety reviews, which also adds to uncertainty of implementation status because regulators can enforce and accept implementation of the same requirements in different ways. It has not been possible to evaluate the impact of such factors on implementation.

The criteria for acceptable national requirements used for interpreting harmonization in the study excluded voluntary agreements with the licensees and other demands that did not meet the generic, publicly available basis, even though regulatory powers may have been used to achieve them. This does not mean that such approaches are less successful, or that the resulting safety levels are lower than in countries with more extensive, formally issued, generic requirements.

6.4 Classification system

The '**A**, **B**, **C**' coding system is a powerful tool with great communication value. It summarizes complicated comparisons. However, caution is necessary when using it. It is a working tool, and not a comparison between countries. Therefore, the study does not allow judgements or comparisons of the overall safety levels on existing plants. The '**A**'s are more likely to be consistent between countries, along with the justified '**B**' differences, which have clear criteria, but a '**C**' can refer to several conditions. It can mean that the whole Reference Level is unsubstantiated, or only one key word within it, so it is necessary to check precisely what is missing: The '**B**' and '**C**' assessments describe the substance of the differences that are usually specific to the country and should help them to draw up their national plans.

7. Conclusions

The RHWG considers that the study has fulfilled WENRA's mandate. The group has produced a set of Reference Levels on relevant issues for harmonization and has identified the substantial differences that need addressing by each participating country to reach harmonization. This has been done in a systematic manner, using agreed criteria. The methodology used has proved to be a good tool for conducting the study and permits a thorough analysis of national positions and validation of the results.

The study indicates that most WENRA countries have already implemented most of the Reference Levels, but also that each country will have to address implementation of others for harmonization. For legal requirements, the situation is significantly different: many Reference Levels are not formally required within countries, according to the strict definitions of the study, so harmonization will require a large regulatory and legal effort. Even so, some participating organizations are already using the results of the study to develop or revise their regulations.

The finalized Reference Levels, benchmark results, and this report are major parts of, but not the only common achievements. In addition, it has given a better understanding of the nuclear safety approaches used within Europe and each member state's legal requirements. During the development of the Reference Levels and the subsequent benchmarking process, group members gained a deep understanding of the strengths and comprehensiveness of current IAEA Safety Standards. This contributed essentially to a common understanding of the attached Reference Levels between members.

The whole process demonstrates that European Regulators are able to co-operate effectively on a broad spectrum of important safety issues through a common commitment to achieve high levels of safety. The active engagement of the participating organizations has generated momentum towards harmonization of reactor safety. There is a potential for further developments, either by broadening the scope of the study, exploring some issues in more detail, or by taking further improvements in safety standards into account. The results achieved should be seen as part of an ongoing process and an intermediate step towards the continuous improvement of reactor safety within Europe.

Appendix – Overview of assessments of Safety Issues

As described in Section 5, the following sub-sections provide pairs of graphs; the first (sub-referenced (a)) all relate to the existence of formal national legal requirements and the second (sub-referenced (b)) to implementation on all of the associated country's NPPs. The graphs all use the following colour scheme throughout:

- **Code A** – Already harmonized in substance;
- **Code B** – A difference exists, but can be justified from a safety point of view; and
- **Code C** – A difference exists, and should be addressed for harmonization.

The following sections consider each safety issue in turn. The vertical axis represents exactly how the 17 countries coded each Reference Level in the document (one code per country).

A – Safety policy

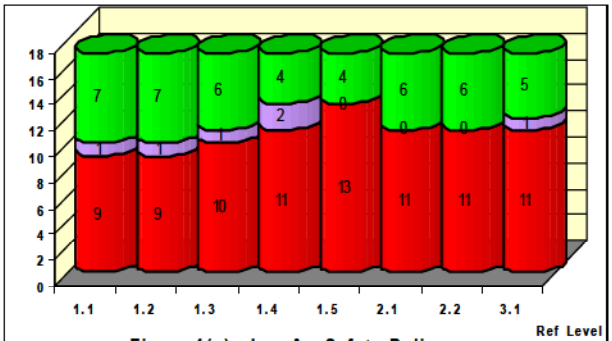

Figure 4(a): Law A - Safety Policy

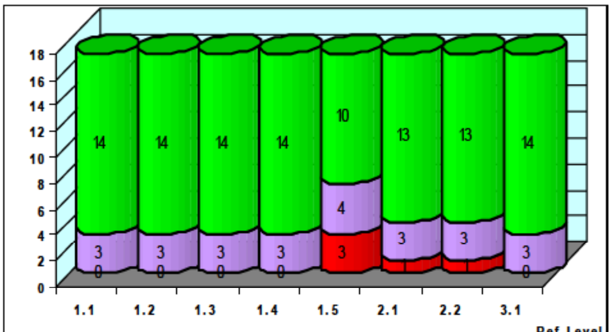

Figure 4(b): Imp A - Safety Policy

For *Safety Policy*, Figure 4(a) shows that only about 30% of countries have formal legal requirements for the Reference Levels, whereas Figure 4(b) indicates that NPPs in all countries have implemented virtually all of them. The exception is Reference Level 1.5, which calls for communication of safety policy to subcontractors.

B – Operating organization

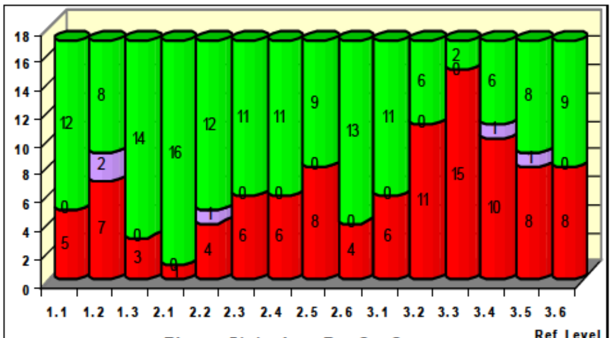

Figure 5(a): Law B - Op Org

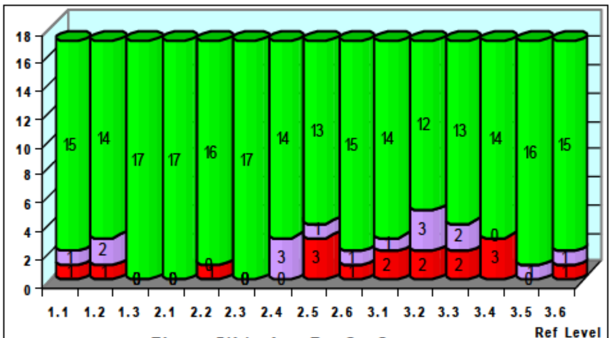

Figure 5(b): Imp B - Op Org

Figure 5(a) shows that Reference Levels for *Operating Organization* are formally required to varying degrees. Ninety percent of countries formally require Reference Level 2.1, relating to ensuring that plants operate safely. Reference Levels with lower percentages of legal requirements relate to staffing, verification of sufficiency of staff, justification of changes in the level of staffing, and, particularly, the requirement for long-term staffing plans, with only two countries satisfying the requirement.

Figure 5(b) illustrates that a large majority of the countries implements Reference Levels. Levels relating to analysis of operational experience, including knowledge gained through R&D projects, and justification of changes in the level of staffing, show lowest levels of implementation.

C – Quality Management System

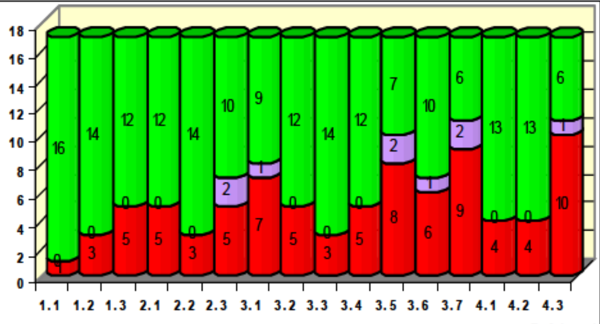

Figure 6(a): Law C - QMS

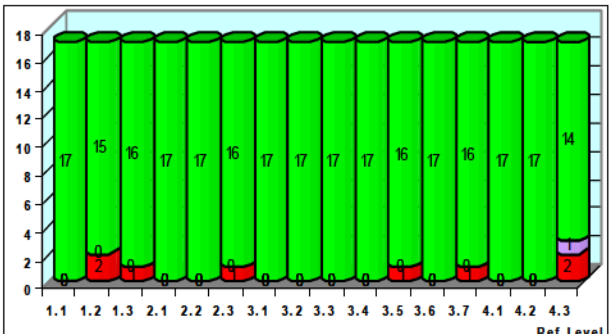

Figure 6(b): Imp C - QMS

Overall, formal requirements are in place in the majority of countries for all but a few *Quality Management System* Reference Levels (Figure 6(a)). Those that are not, relate to responsibilities for implementing the QMS and self-assessment and review by managers.

Most countries have implemented all Levels (Figure 6(b)): the lowest results being for application of a graded approach and managers' self-assessments.

D – Training and authorization

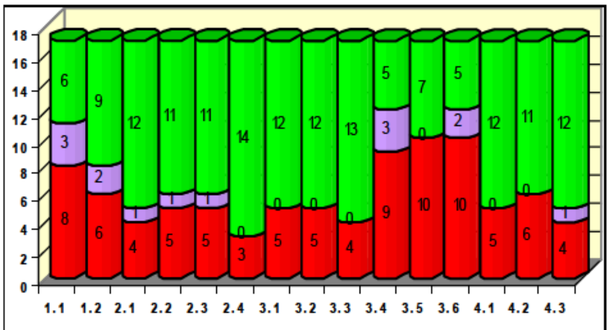

Figure 7(a): Law D - Training

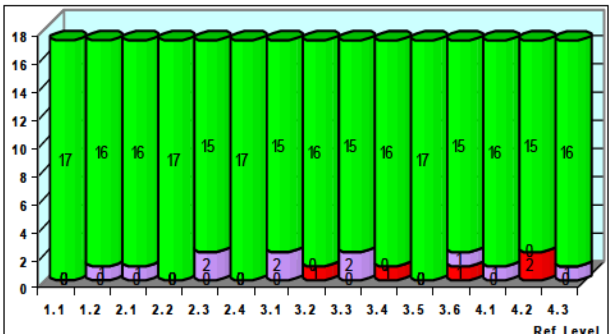

Figure 7(b): Imp D - Training

The numbers of countries with formal requirements for *Training* Reference Levels vary quite widely, with medical examination and authorization of operations staff highest, and annual simulator retraining for at least 5 days, specified items to be included in retraining, and hands-on training for maintenance staff lowest (Figure 7(a)).

Figure 7(b) shows that the majority of countries have implemented the Levels, for all but reauthorisation of operations staff after moving or a period of absence, which has the lowest level of implementation.

E – Verification and improvement of the design

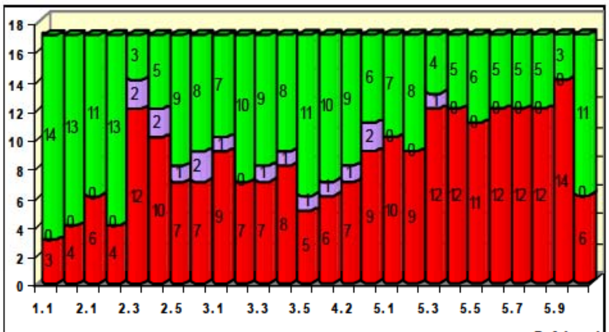

Figure 8(a): Law E - Verification

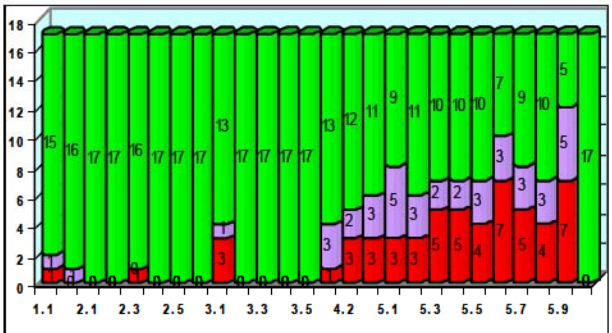

Figure 8(b): Imp E - Verification

Provision of legal requirements gives a mixed picture for *Verification of Design* (Figure 8(a)); the majority of the countries requires about half of the Reference Levels. Only about 50% of countries require a cluster of Reference Levels relating to conservative assumptions and radiological and other technical acceptance criteria for plant conditions, while only 30% or so require Levels relating to instrumentation and provision of hardware for severe accident management.

Figure 8(b) reflects some aspects of Figure 8(a): all countries have implemented most of the Levels dealing with the design basis, but only 80% have implemented radiological and other technical acceptance criteria assigned to plant conditions. Hardware provisions for management of severe accident conditions are only implemented by about 60% of countries.

F – Design basis envelope

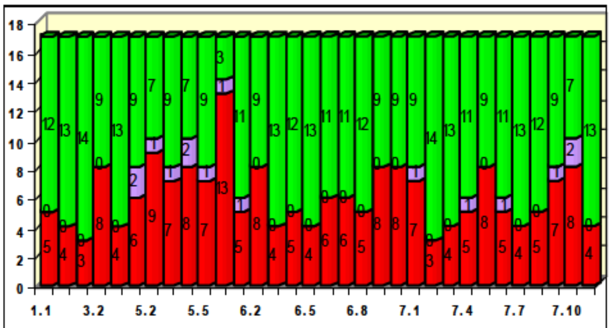

Figure 9(a): Law F - DBE

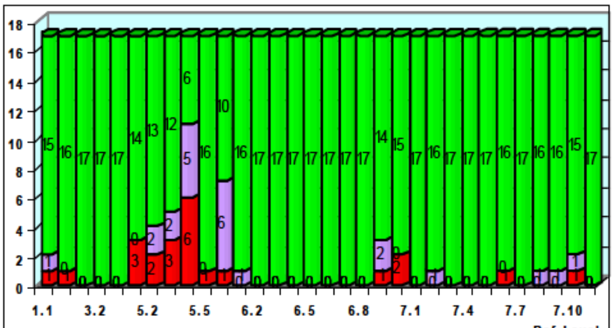

Figure 9(b): Imp App F - DBE

The majority of countries require most of the *Design Basis Envelope* Reference Levels formally (Figure 9(a)). About 60% require three of the Reference Levels listing internal events for analyses, considering beyond design basis events in safety analyses, and listing safety analysis rules.

Most countries have implemented most of the Reference Levels, as can be seen in Figure 9(b). The cluster of Reference Levels that fewer countries have implemented, relates to lists of PIEs to be included in safety analyses, external types of events for analysis, and beyond design basis events for design extension. There are also many justified differences in this group. It is important to remember that failure to match a single item in these lists counts as a difference that requires justification, if this is possible (Code B), or further action (Code C).

G – Safety classification

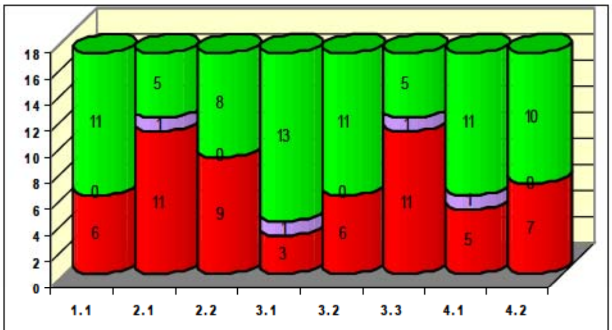

Figure 10(a): Law G - SSCs

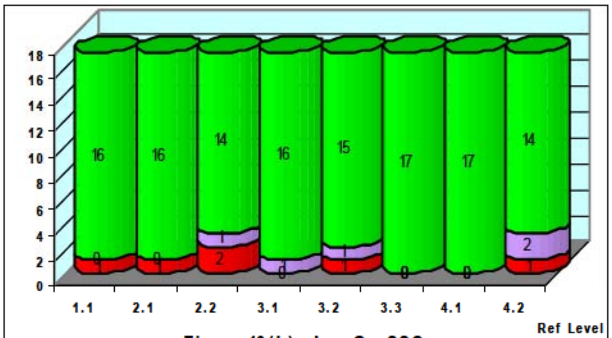

Figure 10(b): Imp G - SSCs

About half the countries have legal requirements for the Reference Levels relating to *Classification of SSCs* in Figure 10(a). Two levels are required least. These have to do with use of deterministic methods as the basis for the classification and that SSCs and auxiliary systems in one class shall not cause failure of other SSCs in another class. Almost all countries have implemented all Reference Levels.

H – Operational Limits and Conditions

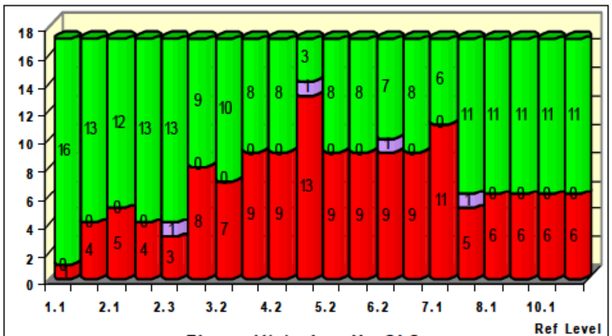

Figure 11(a): Law H - OLCs

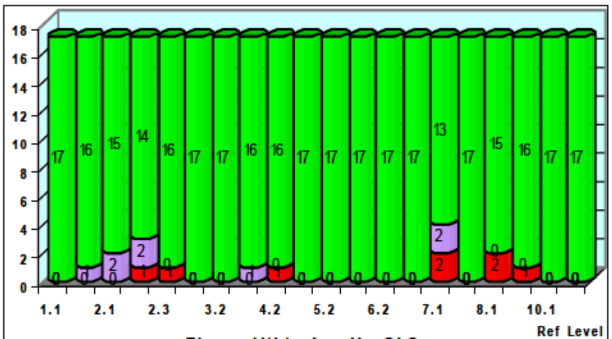

Figure 11(b): Imp H - OLCs

Albeit with a few exceptions, at least 50% of countries formally require *Operational Limits and Conditions* Reference Levels, although Figure 11(a) shows that the picture is quite mixed. Only 30% of countries require the two levels relating to the need to avoid repeated actuation of safety systems, and the general directive for operators to bring the plant to a safer state if they cannot confirm that it is within operating limits or if it behaves unpredictably.

Figure 11(b) shows that most countries implement most of the Reference Levels, with the lowest rate of implementation for the general directive above, and the need for OLCs to contain minimum staffing levels for shift staff.

I – Ageing management

About 50% of countries formally require most of the *Ageing Management* Reference Levels, although more than 80% of the countries require that for the reactor pressure vessel and associated welds (Figure 12(a)).

Figure 12(b) reveals that several Reference Levels have been justified (Coded **B**), so countries have equivalent requirements, or an agreed implementation programme is in place.

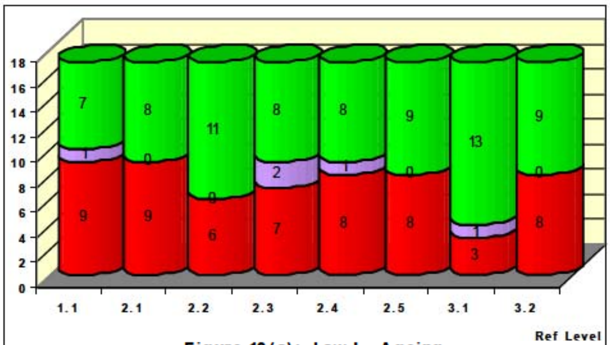

Figure 12(a): Law I - Ageing

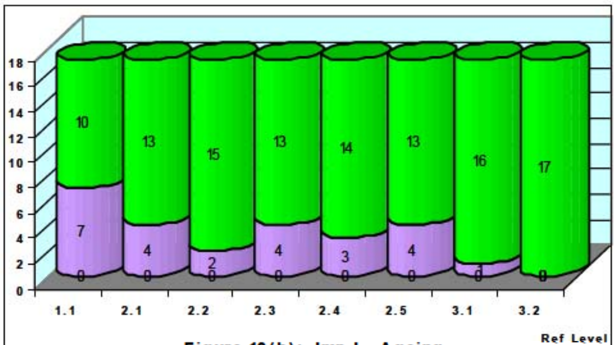

Figure 12(b): Imp I - Ageing

J – System for investigation of events and Operational Experience Feedback

The legal side for *Event Investigation and OEF* reveals another mixed picture in Figure 13(a). Over half of the countries require the Reference Levels, although the proportion is lower for three levels requiring: internal reporting of abnormal events and near misses by staff, procedures for specifying investigation methods, and liaison with organisations involved in design and construction. A high proportion of countries have requirements for establishing OEF programmes and reporting incidents and abnormal events in accordance with procedures.

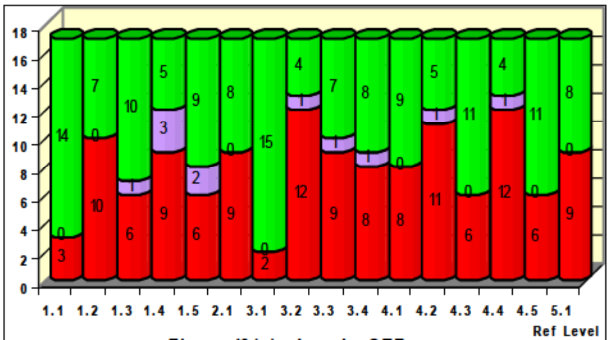

Figure 13(a): Law J - OEF

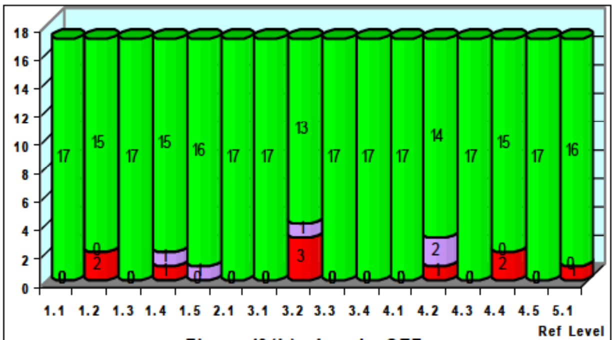

Figure 13(b): Imp J - OEF

Most countries have implemented most of the Reference Levels, as shown in Figure 13(b). Three levels, relating to precursor identification and internal reporting of abnormal events and near misses, show the lowest level of implementation.

K – Maintenance, in-service inspection and functional testing

In Figure 14(a), about 50% of the countries require the *Maintenance and Inspection* Reference Levels.

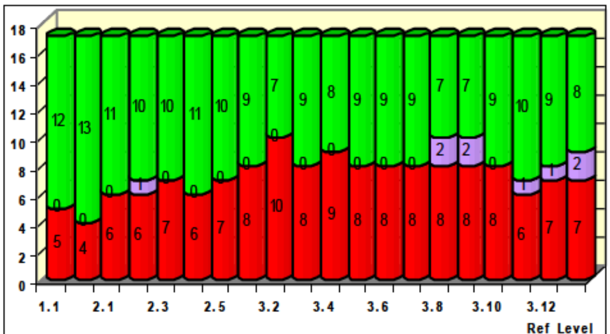

Figure 14(a): Law K - Maintenance

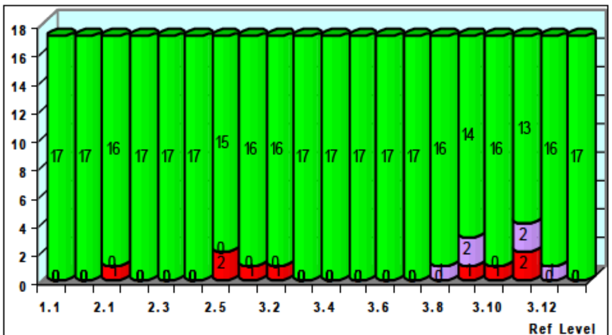

Figure 14(b): Imp K - Maintenance

Figure 14(b) shows most of the Levels implemented: the two least so, relate to assessing the impact of maintenance on plant safety and performing additional investigations in cases where a detected flaw exceeds acceptance standards.

LM – Emergency Operating Procedures and Severe Accident Management Guidelines

Figure 15(a) indicates that few countries have formal legal requirements for *EOP and SAMG* Reference Levels, and none requires that transition from EOPs to SAMGs be exercised (6.2), while systematic preparation of EOPs (3.1) and planning and exercising SAMG interventions (6.3) each have only one '**A**' assessment. Systematic development of EOPs in conjunction with plant-specific analyses is only required by one country.

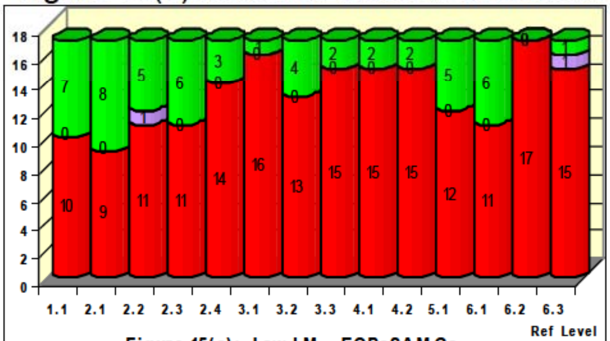

Figure 15(a): Law LM - EOPsSAMGs

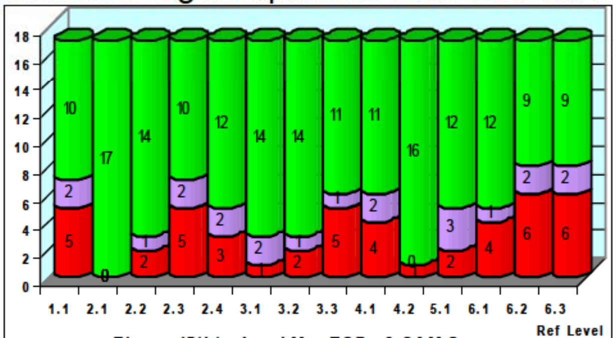

Figure 15(b): Imp LM - EOPs & SAMGs

With implementation, the situation varies (Figure 15(b)). Most countries provide EOPs for Design Basis Events and develop EOPs in conjunction with plant-specific analyses. The latter is an example of the situation where most countries have implemented the Reference Level, but almost no one has any legal requirement to support it. Only about 60% of countries have implemented six Levels covering provision of EOPs, SAMGs, and symptom-based procedures, for exercising the transition from EOPs to SAMGs, and for planning and exercising interventions called for in SAMGs.

N – Contents and updating of SAR

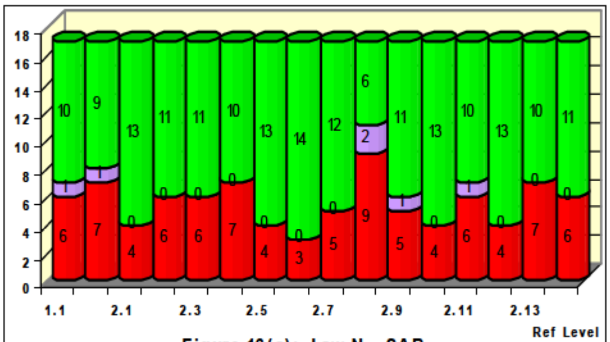

Figure 16(a): Law N - SAR

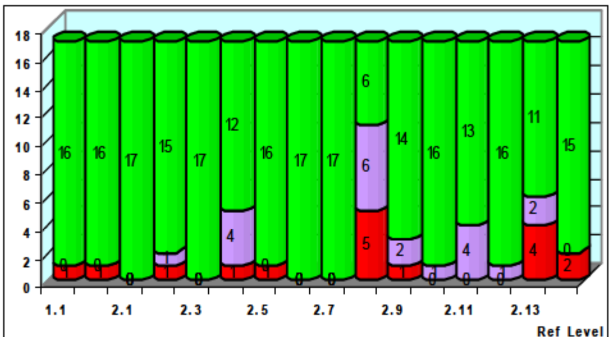

Figure 16(b): Imp App N - SAR

The majority of countries have legal requirements for *SAR* Reference Levels (Figure 16(a)), with one calling for description of plant safety programmes in the SAR, and another for how decommissioning and end-of-life aspects are taken into account during operation, required the least.

Most countries have implemented most Levels; however, Figure 16(b) shows that several countries have justified differences. The same two Levels that had fewest legal requirements also have fewest countries implementing them.

O – Probabilistic Safety Analysis

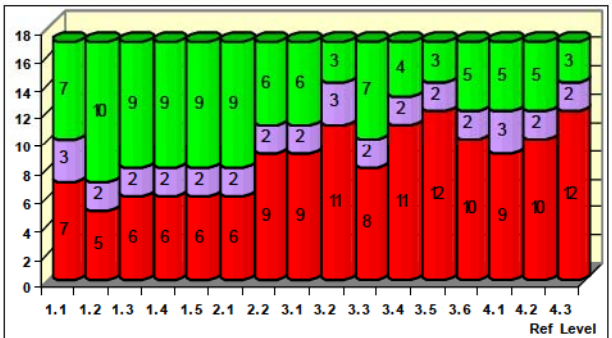

Figure 17(a): Law O - PSA

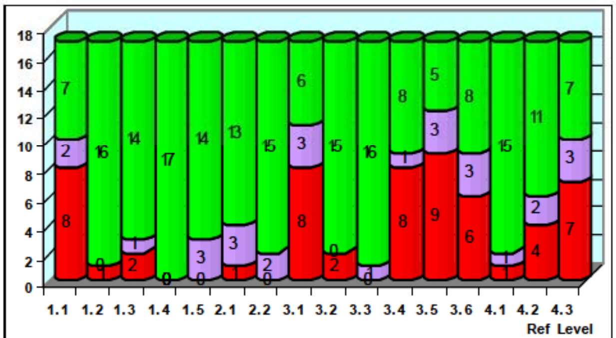

Figure 17(b): Imp O - PSA

About half of the countries have legal requirements for half of the *PSA* Reference Levels. Figure 17(a) shows that a minority of countries requires the rest of the Levels. In general, there are many justified differences. Levels required by fewest countries relate to use of PSA insights to assess plant modifications and changes of procedures, developing of training programmes, and validating inspection programmes. The least required level relates to ensuring operability of components that PSA has shown to be important to safety.

Figure 17(b) shows a mixed picture for implementation: at least half of the countries have implemented the levels – with justified differences in several cases. Sixty percent of countries (some with justified differences) have implemented Reference Level 1.1, which requires specific Level 1 and Level 2 PSAs for all modes of operation and all relevant events for each plant design. All countries have implemented four levels (three with '**B**' justifications) on realistic modelling, Human Reliability Analysis, use of best international practice, and using PSA to identify the need for plant modifications.

P – Periodic Safety Review

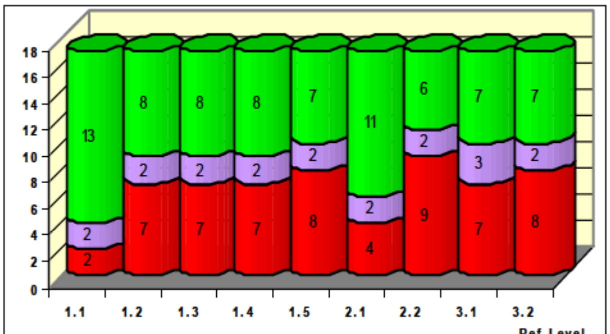

Figure 18(a): Law P - PSR

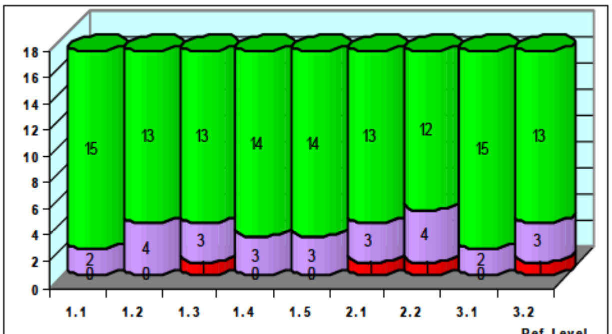

Figure 18(b): Imp P - PSR

At least half of the countries have legal requirements for *PSR* Reference Levels, several with justified differences in several cases, however (Figure 18(a)). Two Levels, dealing with the responsibility for the review and the 10-year interval, are formally required, not by all, but by a large majority of countries.

All but one country have implemented four Levels; the rest are implemented with some justifications (Figure 18(b)).

Q – Plant Modifications

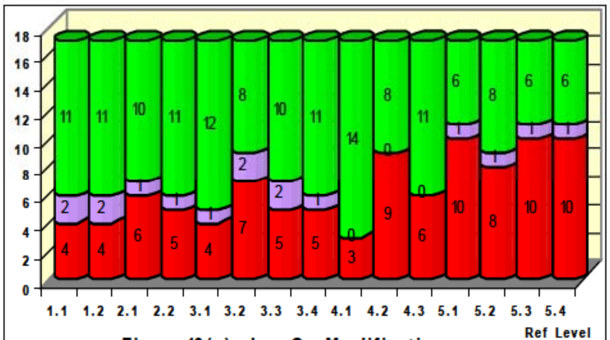

Figure 19(a): Law Q - Modifications

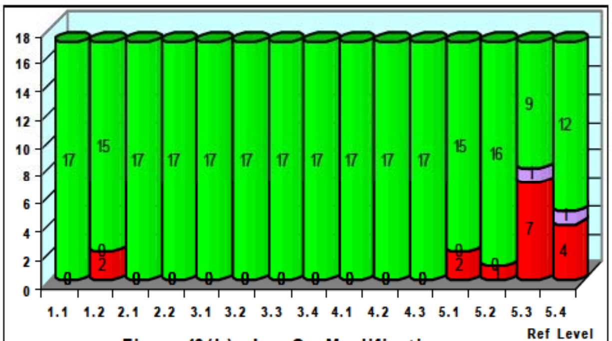

Figure 19(b): Imp Q - Modifications

The picture for *Modifications* (Figure 19(a)) is quite mixed. Over half of the countries formally require the majority of Reference Levels. A minority of the countries satisfies a cluster of levels relating to temporary modifications.

All countries implement most Levels (Figure 19(b)). Those implemented least relate to use of a graded approach to controlling plant modifications and temporary modifications.

R – Emergency Preparedness

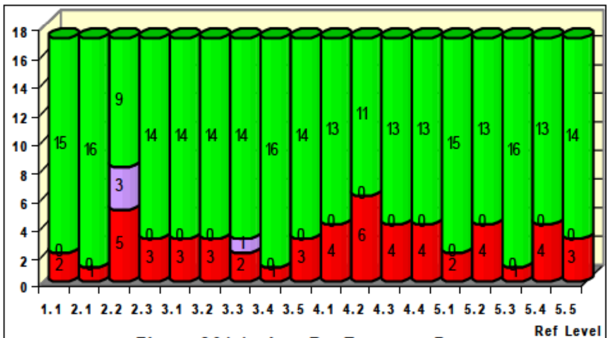

Figure 20(a): Law R - Emerency Prep

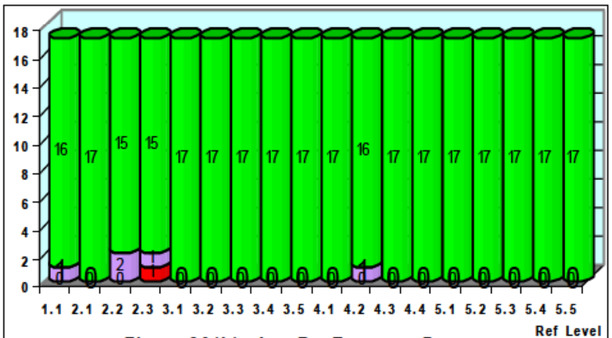

Figure 20(b): Imp R - Emerency Prep

A large majority of the countries formally requires all the *Emergency Preparedness* Reference Levels (Figure 20(a)); those required least concern staffing of the emergency organization and emergency facilities and equipment. In addition, emergency exercises are not required in about 20% of countries.

Figure 20(b) shows that all countries, with one minor exception and a few justified differences, implement all Reference Levels.

S – Protection against internal fires

Figure 21(a) reveals that over half the countries require most of the *Fire Protection* Reference Levels, and that over 80% of countries require some. These levels have to do with implementation of defence-in-depth for fire protection, design, and location of SSCs to minimize the probabilities of fire, escape routes, and provisions for manual fire fighting. Those required by fewest countries relate to features of the fire hazards analysis, active fire protection systems, and the need for procedures to minimize combustible material and ignition sources.

Most of the countries have implemented most levels (Figure 21(b)). Least implemented levels relate to fire hazard analysis and coverage of water based fire protection systems.

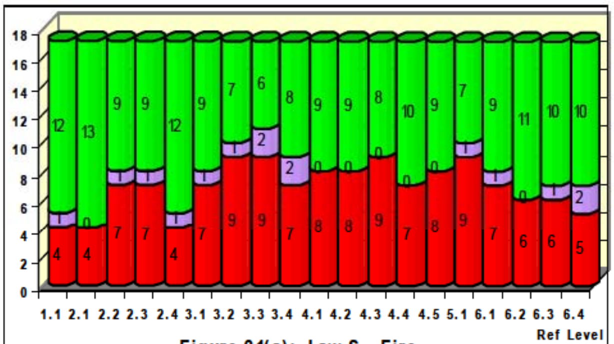

Figure 21(a): Law S - Fire

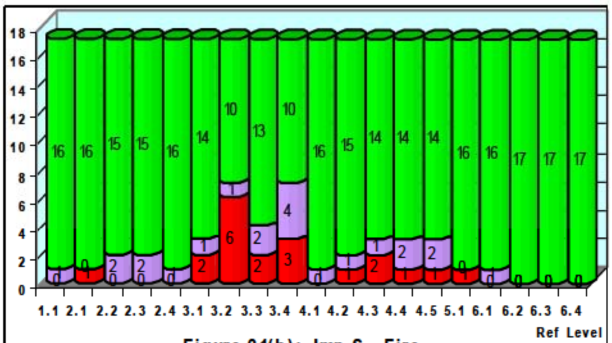

Figure 21(b): Imp S - Fire

Annex 1 – RHWG Reference Levels by Issue

Issue A: Safety Policy	
Safety area: Safety Management	Document status: Final

1. Issuing and communication of a safety policy

1.1 A written safety policy[9] shall be issued by the licensee.

1.2 The safety policy shall be clear about giving safety first priority in all plant activities.

1.3 The safety policy shall include a commitment to continuously develop safety.

1.4 The safety policy shall be communicated to all staff[10], with tasks important to safety, in such a way that the policy is understood and applied.

1.5 Key elements of the safety policy shall be communicated to subcontractors, in such a way that the policy is understood and applied in their on-site activities.

2. Implementation of the safety policy and monitoring safety performance

2.1 The safety policy shall include directives for implementing the policy and monitoring safety performance.

2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the plant management.

3. Evaluation of the safety policy

3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, more frequent than the periodic safety reviews.

[9] A safety policy is understood as a documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as a visible part of an integrated organisational policy.

[10] This is understood as the licensee's own staff.

Issue B: Operating Organization	
Safety area: Safety Management	Document status: Final

1. Organizational structure

1.1 The organ***iz***ational structure for safe and reliable operation of the plant, and for ensuring an appropriate response in emergencies, shall be justified[11] and documented.

1.2 The adequacy of the organ***iz***ational structure, for its purposes according to 1.1, shall be assessed when organ***iz***ational changes are made which might be significant for safety. Such changes shall be justified in advance, carefully planned, and evaluated[12] after implementation.

1.3 Responsibilities, authorities, and lines of communication shall be clearly defined and documented for all staff with duties important to safety.

2. Management of safety and quality

2.1 The licensee shall ensure that the plant is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

2.2 The licensee shall ensure that decisions on safety matters are preceded by appropriate investigation and consultation so that all relevant safety aspects are considered. Safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function.

2.3 The licensee shall ensure that the staff is provided with the necessary facilities and working conditions to carry out work in a safe manner.

2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are analysed in a systematic way and continuously used to improve plant activities.

2.6 The licensee shall ensure that plant activities (processes) are controlled through a documented quality management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

[11] The arguments shall be provided that the organisational structure supports safety and an appropriate response in emergencies.

[12] A verification that the implementation of the organisational change has accomplished its safety objectives.

3. Sufficiency and competency of staff

3.1 The required number of staff for safe operation[13], and their competence, shall be analysed in a systematic and documented way.

3.2 The sufficiency of staff for safe operation, their competence, and suitability for safety work shall be verified on a regular basis and documented.

3.3 A long-term staffing plan[14] shall exist for activities that are important to safety.

3.4 Changes to the number of staff, which might be significant for safety, shall be justified in advance, carefully planned and evaluated after implementation.

3.5 The licensee shall always have in house, sufficient, and competent staff and resources to understand the licensing basis of the plant (e.g. Safety Analysis Report or Safety Case and other documents based thereon), as well as to understand the actual design and operation of the plant in all plant states.

3.6 The licensee shall maintain, in house, sufficient, and competent staff and resources to specify, set standards manage and evaluate safety work carried out by contractors.

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[13] Operation is defined as all activities performed to achieve the purpose for which a nuclear power plant was constructed (according to the IAEA Glossary).

[14] Long term is understood as 3-5 years for detailed planning and at least 10 years for prediction of retirements etc.

Issue C: Quality Management	
Safety area: Safety Management	Document status: Final

1. Objectives

1.1 Throughout the life of a nuclear power plant the licensee shall develop, implement, and maintain a documented quality management system[15] that defines the required quality and safety objectives applicable to work that is important to safety and is carried out by any organization[16], unit, or individual who can affect nuclear safety.

1.2 The quality management system shall grade the requirements set out in it to reflect their relative importance to nuclear safety with respect to each item, service, or process covered.

1.3 The quality management system shall enable the licensee to evaluate compliance with applicable nuclear safety requirements and to identify potential safety improvements.

2. Scope

2.1 Nuclear safety shall be the fundamental consideration in the identification of the items, services, and processes to which the quality management system applies.

2.2 The quality management system shall ensure that the organizational structure, functional responsibilities, levels of authority, and interfaces for all organizations[17], units, and individuals who can affect nuclear safety are clearly documented and assigned.

2.3 The quality management system shall ensure that any organizational change that may affect safety is evaluated, classified with regard to its importance to safety, and justified.

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[15] In some IAEA Member States, the quality assurance programme is referred to as the quality assurance system or the quality system. A more recent term is "Quality Management System". IAEA is revising its main Reference SS document 50-C-SG-Q Code on QA for safety in NPPs etc to align it more with ISO 9001:2000. In Para 1.2 of the 4th draft of DS 338 it explains the new terminology it is proposing to adopt:

> *The term "Management System" has been adopted instead of "Quality Assurance". The term "Management System" reflects the evolution in the approach from the initial concept of"Quality Control" (controlling the quality of products) through "Quality Assurance" (the system to assure the quality of products) and "Quality Management" (the system to manage quality). The "Management System" is a set of interrelated or interacting elements (system) to establish policy and objectives and to achieve those objectives.*

In this Reference Level document, "quality management system" has been used in anticipation of that change whilst adhering largely to related standards from fully endorsed, rather than draft, IAEA standards.

[16] Such organizations include all those within the licensee's company as well as designers, vendors, contractors, suppliers, and service providers employed directly or indirectly on work for the licensee.

[17] Such organizations include all those within the licensee's company as well as designers, vendors, contractors, suppliers, and service providers employed directly or indirectly on work for the licensee.

3. Implementation

3.1 The most senior person representing the licensee on site shall be responsible and accountable for ensuring that an effective quality management system is being implemented and that the senior management team is committed to and meeting its responsibility for reviewing and ensuring the success of the programme.

3.2 The licensee shall establish and maintain sufficient resources and processes to define, achieve, analyse, and preserve the quality of items that are important to safety, and to take timely and effective corrective or preventive action to respond to deviations from required specifications.

3.3 The licensee shall ensure that procured items and services meet established requirements and perform as specified and that selected suppliers continue to provide acceptable items and services during the fulfilment of their procurement obligations. Licensees may delegate procurement activities to other organizations, but shall remain responsible for the overall effectiveness of these activities.

3.4 Products and processes that do not conform to specified requirements shall be identified and reported to an appropriate level of management within the organization. The safety implications of the non-conformances shall be evaluated and the actions taken shall be recorded, where appropriate.

3.5 The quality management system shall be implemented in collaboration[18] with management, those performing the work, and those assessing the work.

3.6 Work that is important to safety shall be controlled and performed using easily understood, approved current instructions, procedures, drawings, or other means, that have been appropriately validated before first use and are periodically reviewed to ensure adequacy and effectiveness.

3.7 Personnel shall be trained in requirements of the quality management system, so that they are competent to perform their assigned work and understand the safety consequences of their activities.

4. Assessment

4.1 The licensee shall assess the quality management system on a regular basis to ensure that it provides the required level of safety.

4.2 An organizational unit or group shall be established, or an outside agency assigned, that is responsible for independently assessing the adequacy of management processes and work performed that has sufficient authority and organizational freedom to carry out its responsibilities. People who conduct independent assessments shall not participate directly in the work being assessed[19].

4.3 All managers shall regularly carry out self-assessment and review of the processes for which they are responsible to determine their efficiency and effectiveness with establishing, promoting, and achieving nuclear safety objectives, and shall take any necessary corrective actions.

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[18] Collaboration is taken to mean that all groups are involved in the process.

[19] However, it is important that the audit team is familiar with the work being assessed. The aim of this requirement is to avoid any conflict of interest on the part of the assessor.

Issue D: Training and Authorization of NPP staff (jobs with safety importance)	
Safety area: Safety Management	Document status: Final

1. Policy

1.1 The licensee shall establish an overall training policy and a comprehensive training plan on the basis of long-term training needs and goals that acknowledges the critical role of safety. The plan shall be kept up to date.

1.2 A systematic approach to training shall be used to provide a logical progression, from identification of the competences required for performing a job, to the development and implementation of training programmes including respective training materials for achieving these competences, and to the subsequent evaluation of this training.

2. Competence and qualification

2.1 Only qualified persons that have the necessary knowledge, skills, and safety attitudes shall be allowed to carry out tasks important to safety. The licensee shall ensure that all personnel performing safety-related duties including contractors have been adequately trained and qualified.

2.2 The Licensee shall define and document the necessary competence requirements for their staff.

2.3 Appropriate training records and records of assessments against competence requirements shall be established and maintained for each individual with tasks important to safety.

2.4 Staff qualifying for positions important to safety shall undergo a medical examination to ensure their fitness for the duties and responsibilities assigned to them. The medical examination shall be repeated at specified intervals.

3. Training programmes and facilities

3.1 Performance based training programmes shall be established for all staff with tasks important to safety. The programmes shall cover basic training in order to qualify for a certain position and refresher training as needed.

3.2 All technical staff including contractors shall have a basic understanding of nuclear safety, radiation safety, personal safety, and the on-site emergency arrangements.

3.3 Representative simulator facilities shall be used for the training of control room operators to such an extent that the hands-on-training of normal and emergency operating procedures is effective, and shall be equipped with software to cover normal operation, anticipated operational occurrences, and a range of accident conditions.

3.4 For control room operators, initial and annual refresher training shall include training on a representative full-scope simulator. Annual refresher training shall include at least 5 days on the simulator[20].

3.5 Refresher training for control room operators shall include especially the following items as appropriate

- Plant operation in all normal operational states, transients, and accidents
- Shift crew teamwork
- Operational experiences and modifications of plant and procedures.

3.6 Maintenance and technical support staff including contractors shall have practical hands-on-training on the required safety critical activities.

4. Authorization

4.1 Staff controlling changes in the operational status of the plant shall be required to hold an authorization valid for a specified time period. The licensee shall establish procedures for their staff to achieve this authorization. In the assessment of an individual's competence and suitability as a basis for the authorization, documented criteria shall be used.

4.2 If an authorized individual:

- Moves to another position for which an authorization is required;
- Has been absent from the authorized position during an extended time period;

Re-authorization shall be conducted after necessary individual preparations.

4.4 Work on safety related structures, systems, or components carried out by contractor personnel shall be approved and monitored by a suitably competent member of licensee's staff.

[20] Time includes the necessary briefings.

Issue E: Verification and Improvement of the Design	
Safety area: Design	Document status: Final

1. Selection of design basis events and hazards[21]

1.1 The current design basis shall be clearly and systematically defined and documented.

1.2 The design basis shall include a set of postulated initiating events, with consideration of failures and hazards (internal and external, natural and man-induced), selected with deterministic or probabilistic methods or a combination of both, to demonstrate that the necessary safety functions are accomplished and the safety objectives met

2. Demonstration of reasonable conservatism and safety margins of the design basis

2.1 The initial and boundary conditions shall be specified in a conservative way.

2.2 The single failure criterion shall be applied in all design basis analyses of postulated initiating events.

2.3 Non-safety systems, including off-site power, shall be assumed to operate only if they aggravate the effect of the initiating event[22].

2.4 The safety systems shall be assumed to operate at their performance level[23] that is most penalizing for the initiator.

2.5 Any failure, occurring as a consequence of a postulated initiating event, shall be included in the design basis analysis.

2.6 The impact of uncertainties, which are of importance for the results, shall be addressed in the design basis analyses.

3. Definition and application of technical acceptance criteria

3.1 Radiological and other technical acceptance criteria shall be assigned to each plant condition (typically normal operation, anticipated operational occurrences, design basis accidents, additional failure assumptions, and severe accidents), according to its probability of occurrence.

3.2 Criteria for protection of the fuel cladding shall be specified, including fuel temperature, DNB, cladding temperature, fuel rod integrity, and maximum allowable fuel damage during any design basis accident.

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[21] Only deterministic analyses are to be considered here. Probabilistic analyses are treated in Issue O (on PSA)

[22] This means that non-safety systems are either supposed not to function after the initiator, either supposed to continue to function as before the initiator, depending on which of both cases is most penalising.

[23] The performance level can be at the minimum or the maximum, depending on which of both cases is most penalising.

3.3 Criteria for the protection of the (primary) coolant pressure boundary shall be specified, including maximum pressure, maximum temperature, thermal- and pressure transients, and stresses.

3.4 For PWR only: Criteria in 3.3 shall be specified as well for protection of the secondary coolant system.

3.5 Criteria shall be specified for protection of the containment, including temperatures, pressure and leak rates.

4. Accidents beyond design basis[24]

4.1 Consideration shall be given to the performance of the plant in specified accidents beyond the design basis, including a selection of severe accidents, to determine those sequences for which reasonable practicable preventive or mitigatory measures can be identified (accident vulnerability study). For this study a combination of engineering judgement and probabilistic methods can be used and evaluations be made on a best estimate basis.

4.2 Consideration shall be given, in the same manner as in 4.1, to combination of postulated initiating events with internal and external hazards.

4.3 The specified accidents beyond the design basis shall include station blackout, ATWS, multiple SG tube rupture, loss of main heat sink, and loss of required safety systems in the long term after a postulated initiating event.

5. Instrumentation and hardware provisions for the management of severe accident conditions

5.1 Adequate instrumentation shall exist which can survive severe accident environmental conditions in order to manage such accidents according to guidelines/procedures for severe accidents.

5.2 Necessary information from instruments shall be relayed to the control room and presented in such a way to enable a timely assessment of the plant status and critical safety functions in severe accident conditions.

5.3 Means shall exist for containment isolation in a severe accident, including bypass prevention[25].

5.4 The containment leak-tightness shall be ensured for a reasonable time after a severe accident.

5.5 Means shall be provided to manage pressure and temperature in the containment during a severe accident.

5.6 Means shall be provided to control combustible gases in a severe accident.

5.7 Means shall be provided for containment overpressure protection in a severe accident.

5.8 Means shall be provided for prevention of high-pressure core-melt scenarios.

5.9 Means shall be provided to prevent containment melt through.

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[24] Only deterministic analyses are to be considered here. Probabilistic analyses are treated in Issue O (on PSA)

[25] It is understood that the means mentioned in 5.3-5.9 shall be able to perform its functions in relevant severe accident conditions, although not formally qualified.

6. Improvement of the design

6.1 The current design shall on a regular basis, and when needed as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach, against current requirements and practices to identify deviations. The safety significance of identified deviations shall be determined with respect to possible design improvements or back-fitting or other measures justified from a safety point of view.

Issue F: Design Basis Envelope for existing reactors	
Safety area: Design	Document status: Final

1. Objective

1.1 The design shall have as an objective the prevention or, if this fails, the mitigation of radiation exposures resulting from design basis accidents and selected beyond design basis accidents. Design provisions shall be made to ensure that potential radiation doses to the public and the site personnel do not exceed acceptable limits and are as low as reasonably achievable.

2. Scope

2.1 The design basis shall specify the necessary capabilities of the plant to cope with a specified range of plant states within the defined radiological protection requirements. The design basis shall include normal operation and transients/accident conditions from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis.

3. Safety strategy

3.1 Defence-in-depth shall be applied in order to prevent releases harmful to the public and the environment during normal operation, operational occurrences, and design basis accident conditions. The design shall therefore provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment.

3.2 The design shall prevent as far as practicable:

- Challenges to the integrity of the barriers;
- Failure of a barrier when challenged by a PIE;
- Failure of a barrier as consequence of a failure of another barrier.

4. Safety functions

4.1 The plant shall be able to fulfil the fundamental safety functions[26]:

- Control of reactivity;
- Removal of heat from the core; and
- Confinement of radioactive material;

during normal operation, anticipated operational occurrences, and design basis accident conditions.

5. General design basis

5.1 [For benchmarking of requirements only]: A set of design basis accidents shall be derived from the listing of all relevant PIEs for the purpose of setting

[26] Under the conditions specified in section 6.

boundary conditions according to which the structures, systems and components important to safety shall be designed. Structures, systems, and components important to safety shall be designed to be capable of withstanding all identified PIEs with sufficient reliability.

[For benchmarking of implementation only]: The following types of PIEs shall, as a minimum be included in the safety analysis for the design of safety systems with sufficient reliability:

- Small, medium, large LOCA;
- Breaks in the main steam and main feed water systems;
- Forced decrease of reactor coolant flow;
- Forced increase or decrease of main feed water flow;
- Forced increase or decrease of main steam flow;
- Inadvertent opening of valves at the pressurizer (PWR);
- Inadvertent operation of ECCSs;
- Inadvertent opening of valves at the steam generators (PWR);
- Inadvertent opening of main steam relief/safety valves (BWR);
- Inadvertent closure of main steam isolation valves;
- SGT rupture (PWR);
- Uncontrolled movement of control rods;
- Ejection of control rods;
- Loss of off-site power;
- Chemical and volume control system (CVCS) malfunction (PWR);
- Pipe breaks or heat exchanger tube leaks in systems connected to the RCS and located partially outside containment (Interfacing System LOCA);
- Fuel handling accidents;
- Loss of core cooling in the RHR mode;
- Loss of Fuel Pool Cooling.

5.2 The design shall take into consideration specific loads and environmental conditions imposed on structures, systems and components by internal events. The following types of internal events shall as a minimum be included in the safety analysis:

- Pipe whipping;
- Internal flooding;
- Internal missiles;
- Load drop;
- Internal explosion;
- Fire.

5.3 The design shall take into consideration specific loads and environmental conditions imposed on structures, systems, and components by natural and man made external events specific for the site. The following types of external events shall, as a minimum, be included in the safety analysis according to site specific criteria:

- Extreme[27] wind loading;
- Extreme outside temperatures;
- Extreme rainfall and site flooding;

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[27] In comparison with historical weather data for the site region.

- Extreme cooling water temperatures and icing;
- Earthquake;
- Aircraft crash;
- Nearby transportation and industrial activities.

5.4 [For benchmarking of requirements only]: Selected beyond design basis events shall be considered in the safety analysis to determine those sequences for which reasonable practicable preventive or mitigative measures can be identified and implemented. For these events, realistic analysis assumptions and modified acceptance criteria may be used:

[For benchmarking of implementation only]: The following types of events shall be considered in the safety analysis for the design extension[28]. Realistic analysis assumptions and modified acceptance criteria may be used:

- ATWS;
- Station blackout;
- Total loss of feed water;
- LOCA together with the complete loss of one emergency core cooling system;
- Uncontrolled level drop during mid-loop operation (PWR) or during refuelling;
- Total loss of the Component Cooling Water System;
- Loss of Ultimate Heat Sink;
- Uncontrolled boron dilution (PWR);
- Multiple SGT ruptures (PWR);
- A steam line break together with a SG tube rupture.

5.5 Plant states shall be identified and PIEs shall be grouped into a limited number of categories according to their probability of occurrence. The categories typically cover normal operation, anticipated operational occurrences, design basis accidents and beyond design basis accidents. Acceptance criteria shall be assigned to each category that take account of the requirement that frequent PIEs shall have only minor or no radiological consequences, and that events that may result in severe consequences shall be of very low probability.

5.6 The following safety analysis rules shall normally be observed, any deviations shall be justified:

- Only safety classified systems shall be used in order to reach and to maintain the safe shutdown state;
- The most penalizing single failure shall be applied to an equipment used to achieve the safety function;
- Manual action from the main control room shall be assumed to take place, at the earliest, 30 minutes after the first significant information is given to the operator;

[28] Design extension is understood as measures taken to cope with additional PIEs, not covered by earlier defined design basis events. Design extension analyses may be done with realistic assumptions.

- A stuck rod shall be considered as an additional aggravating failure for events during anticipated operational occurrences, events of moderate frequency and infrequent events;
- Loss of offsite power shall be considered as an additional aggravating failure for events of moderate frequency and infrequent events.

6. Design of safety functions

General

6.1 The fail-safe principle shall be considered in the design of systems and components important to safety.

6.2 A failure in a system intended for normal operation shall not affect a safety function.

6.3 Design features and suitable redundancy and diversity in components shall be provided in order to fulfil the requirements with sufficient reliability for each PIE, on the assumption of a single failure.

6.4 The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components[29], redundancy, diversity, physical and functional separation, and isolation.

Reactor shutdown functions

6.5 The means for shutting down the reactor shall consist of at least two diverse systems.

6.6 At least one of the two systems shall, on its own, be capable of quickly[30] rendering the nuclear reactor sub critical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure.

Heat removal functions

6.7 Means for removing residual heat from the core after shutdown, and during and after anticipated operational occurrences and accident conditions, shall be provided taking into account the assumptions of a single failure and the loss of off-site power.

Confinement functions

6.8 A containment system shall be provided in order to ensure that any release of radioactive material to the environment in a design basis accident would be below prescribed limits. This system shall include:

- Leak-tight structures covering all essential parts of the primary system;
- Associated systems for control of pressures and temperatures;
- Features for isolation, management, and removal of fission products, hydrogen, oxygen, and other substances that could be released into the containment atmosphere.

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[29] Proven by experience under similar conditions or adequately tested and qualified.

[30] Within 4-6 seconds, e.g. Scram system.

6.9 Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two adequate containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

6.10 Each line that penetrates the primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one adequate containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

7. Instrumentation and control systems

7.1 Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, and the containment, and for obtaining any information on the plant necessary for its reliable and safe operation. Provision shall be made for automatic recording[31] of measurements of any derived parameters that are important to safety.

7.2 Instrumentation shall be environmentally qualified for the plant states concerned and shall be adequate for measuring plant parameters and thus classifying events for the purposes of emergency response.

Control room

7.3 A control room shall be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences and design basis accidents.

7.4 Devices shall be provided to give in an efficient way visual and, if appropriate, audible indications of operational states and processes that have deviated from normal and could affect safety. Ergonomic factors shall be taken into account in the design of the control room. Appropriate information shall be available to the operator to monitor the effects of the automatic actions.

7.5 Special attention shall be given to identifying those events, both internal and external to the control room, which may pose a direct threat to its continued operation, and the design shall provide for reasonably practicable measures to minimize the effects of such events.

7.6 For times when the main control room is not available, there shall be sufficient instrumentation and control equipment available to shut down the reactor, maintain it in a safe shut down state and remove residual heat from a supplementary control room/post, which is physically and electrically separated from the main control room. It shall also be possible to monitor the essential reactor parameters from the supplementary control room/post.

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[31] By computer sampling and/or print outs

Protection system

7.7 Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:

- No single failure results in loss of protection function; and
- The removal from service of any component or channel does not result in loss of the necessary minimum redundancy.

7.8 The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in operation.

7.9 The design of the reactor protection system shall be such as to minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal operations and expected operational occurrences, but not to negate correct operator actions in design basis accidents.

7.10 Computer based systems used in a protection system, shall fulfil the following requirements:

- The highest quality of and best practices for hardware and software shall be used;
- The whole development process, including control, testing and commissioning of design changes, shall be systematically documented and reviewed;
- In order to confirm confidence in the reliability of the computer based systems, an assessment of the computer based system by expert personnel independent of the designers and suppliers shall be undertaken; and
- Where the necessary integrity of the system cannot be demonstrated with a high level of confidence, a diverse means of ensuring fulfilment of the protection functions shall be provided.

Emergency power

7.11 It shall be ensured that the emergency power supply is able to supply the necessary power to systems and components important to safety, in any operational state or in a design basis accident, on the assumption of a single failure and the coincidental loss of off-site power.

Issue G: Safety Classification of Structures, Systems and Components	
Safety area: Design	Document status: Final

1. Principle

1.1 All SSCs[32] important for safety shall be identified and classified on the basis of their importance for safety. They shall be designed, constructed, and maintained such that their quality and reliability is commensurate with this classification.

2. Classification process

2.1 The classification of SSCs shall be based on deterministic methods, complemented where appropriate by engineering judgment.

2.2 The classification shall identify for each safety class:

- The appropriate codes and standards in design, manufacturing, construction and inspection;
- Need for emergency power supply, qualification to environmental conditions;
- The availability or unavailability status of systems for PIEs[33] to be considered in deterministic safety analysis;
- The QA provisions.

3. Design for reliability

3.1 SSCs important to safety shall be designed to withstand PIEs with sufficient reliability.

3.2 The potential for common cause failure shall be considered to determine where diversity, redundancy, and independence should be applied to achieve the necessary reliability.

3.3 The failure of a SSC in one safety class shall not cause the failure of other SSCs in a higher safety class. Auxiliary systems supporting equipment important to safety shall be classified accordingly.

4. Selection of materials and qualification of equipment

4.1 The design of SSCs important to safety and the materials used shall consider the effects of operational conditions over the plant lifetime and the effects of design basis accidents on their characteristics and performance.

4.2 A qualification procedure shall be adopted to confirm that SSCs important to safety meet throughout their design operational lives the demands for performing their function, taking into account environmental conditions[34] over the lifetime of the plant and when required[35].

[32] SSCs include software for I&C.

[33] Postulated Initiating Event – as defined by IAEA – includes consequences following on from event.

[34] Environmental conditions include as appropriate vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, humidity, and combinations thereof.

[35] When required includes as appropriate the consequences of PIEs and hazards.

Issue H: Operational Limits and Conditions	
Safety area: Operation	Document status: Final

1. Purpose

1.1 OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intentions as documented in the SAR.

1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur.

2. Establishment and review of OLCs

2.1 Each established OLC shall have detailed justification based on plant design, safety analysis and commissioning tests.

2.2 OLCs shall be kept updated and reviewed in the light of experience, developments in science and technology, and every time modifications in the plant or in the safety analysis warrant it, and changed if necessary.

2.3 The process for making modifications or temporary modifications of OLCs shall be defined. Such modifications shall be adequately justified by safety analysis and independent safety review.

3. Use of OLCs

3.1 The OLCs shall be readily accessible to control room personnel.

3.2 Control room operators shall be highly knowledgeable of the OLCs and their technical basis and relevant operational decision makers shall be aware of their significance for the safety of the plant.

4. Scope of OLCs

4.1 OLCs shall cover all operational plant states including power operation, shutdown and refuelling, transitions between these states and temporary situations arising due to maintenance & testing.

4.2 OLCs shall include:

- Safety limits;
- Safety systems settings;
- Equipment required; and
- Action to be taken in the case of deviations from OLCs.

5. Safety limits, safety systems settings, and operational limits

5.1 Adequate margins shall be provided between safety limits, safety systems settings, alarms, and operational limits to avoid activating safety systems too frequently.

5.2 Safety limits shall be established using a conservative approach to take uncertainties in the safety analyses into account.

6. Unavailability limits

6.1 Limits and conditions for normal operation shall include limits on operating parameters, stipulation for minimum amount of operable equipment, actions to be taken by the operating staff in the event of deviations from the OLCs and time allowed to complete these actions.

6.2 Where operability requirements cannot be met, the actions to bring the plant to a safer state, such as power reduction or reactor shutdown, shall be specified, and the time allowed to complete the action shall be stated.

6.3 Operability requirements shall state for the various modes of normal operation the number of systems or components important to safety that should be in operating condition or standby condition.

7. Unconditional requirements

7.1 If operating personnel cannot ascertain that the power plant is operating within operating limits, or the plant behaves in an unpredicted way, measures shall be taken without delay to bring the plant to a safer state.

7.2 Plant shall not be returned to service following unplanned shutdown until it has been shown to be safe to do so.

8. Staffing levels

8.1 Minimum staffing levels for shift staff shall be stated in the OLCs.

9. Surveillance

9.1 The licensee shall ensure that an appropriate surveillance[36] program is established and implemented to ensure compliance with OLCs and shall ensure that results are evaluated and retained.

10. Non-compliance

10.1 In cases of non-compliance, remedial actions shall be taken immediately to re-establish OLC requirements.

10.2 Reports of non-compliance shall be investigated and corrective action shall be implemented in order to help prevent such non-compliance[37] in future.

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[f] The objectives of the surveillance programme are: to maintain and improve equipment availability, to confirm compliance with operational limits and conditions, and to detect and correct any abnormal condition before it can give rise to significant consequences for safety. The abnormal conditions which are of relevance to the surveillance programme include not only deficiencies in SSCs and software performance, procedural errors and human errors, but also trends within the accepted limits, an analysis of which may indicate that the plant is deviating from the design intent. *(NS-G-2.6 Para 2.11)*

[37] If the actions taken to correct a deviation from OLCs are not as prescribed, including those times when they have not been completed successfully in the allowable outage time, plant shall be deemed to have operated in non-compliance with OLCs.

Issue I: Ageing Management	
Safety area: Operation	Document status: Final

1. Content of an Ageing Management Programme

1.1. In addition to the maintenance, surveillance, and inspection programmes, the operating organization shall have an Ageing Management Programme[38] to identify all ageing mechanisms important to safety related structures, systems and components (SSCs), determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.

2. Technical requirements, methods, and procedures

2.1 The licensee shall assess structures, systems, and components important to safety taking into account of relevant ageing and wear-out mechanisms and potential age related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.

2.2 The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.

2.3. The Periodic Safety Reviews shall be used determine whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.

2.4. In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.

2.5. The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.

3. Major structures and components

3.1. Ageing management of the reactor pressure vessel and its weldments shall take all relevant factors including embrittlement, thermal ageing, and fatigue

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[38] **Ageing** is considered as a process by which the physical characteristics of a structure, system, or component (SSC) change with time (ageing) or use (wear-out).

An Ageing Management Programme (AMP) should be understood as an integrated approach to identifying analyzing monitoring and taking corrective actions and document the ageing degradation of structures, systems, and components.

into account to compare their performance with prediction, throughout plant life.

3.2. Monitoring of major structures and components shall be carried out to timely identify preventive and remedial actions such as changes to water chemistry, to periodic in-service inspection.

Issue J: System for Investigation of Events and Operational Experience Feedback	
Safety area: Operation	Document status: Final

1. Programmes and Responsibilities

1.1 The licensee shall establish and conduct a programme to collect, screen, analyse, and document operating experience and events at the plant in a systematic way. Relevant operational experience and events reported by other plants shall also be considered.

1.2 Operating experience at the plant shall be evaluated to identify any undetected safety relevant events or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margin.

1.3 The licensee shall designate staff for carrying out these programmes, for the dissemination of findings important to safety and – where appropriate – for recommendations on actions to be taken. Significant findings and trends shall be reported to the licensee's top management.

1.4 Staff responsible for evaluation of operational experience and investigation into events shall receive adequate training, resources, and support from the line management.

1.5 The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered, and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

2. Collection, documentation and storage of events

2.1 Experience from normal and abnormal operation and other important safety-related information shall be organized, documented, and stored in such a way that it can be easily retrieved and systematically searched, screened and assessed by the designated staff

3. Reporting and dissemination of safety significant information

3.1 The licensee shall report incidents and abnormal events of significance to safety in accordance with established procedures and criteria.

3.2 Plant personnel shall be required to report abnormal events and be encouraged to report internally near misses relevant to the safety of the plant.

3.3 Information resulting from the operational experience shall be disseminated to relevant staff and shared with relevant national and international bodies.

3.4 A process shall be put in place to ensure that operating experience of events at the plant concerned as well as of relevant events at other plants is appropriately considered in the training programme for staff with tasks related to safety.

4. Assessment and investigation of events

4.1 An initial assessment of events important to safety shall be performed without delay to determine whether urgent actions are necessary.

4.2 The licensee shall have procedures specifying appropriate investigation methods. Methods of human performance analysis shall be used to investigate human performance related events.

4.3 Event investigation shall be conducted on a time schedule consistent with the event significance. The investigation shall:

- Establish the complete event sequence;
- Determine the deviation;
- Include direct and root cause analysis;
- Assess the safety significance including potential consequences; and
- Identify corrective actions.

4.4 The operating organization shall maintain liaison as appropriate with the organizations (manufacturer, research organization, designer) involved in design and construction, with the aims of feeding back information on operating experience and obtaining advice, if necessary, in case of equipment failures or abnormal events.

4.5 As a result of the analysis, timely corrective actions shall be taken such as technical modifications, administrative measures, or personnel training to restore safety, to avoid event recurrence and to improve safety margins and trends.

5. Review and continuous improvement of the OEF process

5.1 Periodic reviews of the effectiveness of the OEF process based on performance criteria shall be undertaken and documented either within a self-assessment programme by the licensee or by a peer review team.

Issue K: Maintenance, In-Service Inspection, and Functional Testing	
Safety area: Operation	Document status: Final

1. Scope and objectives

1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of SSCs important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. They shall take into account operational limits and conditions and be re-evaluated in the light of experience.

1.2 The programme shall include periodic inspections or tests of SSCs important to safety in order to demonstrate their reliability and to determine whether they are acceptable for continued safe operation of the plant or whether any remedial measures are necessary.

2. Programme establishment and review

2.1 The extent and frequency of preventive maintenance, testing, surveillance and inspection of SSCs shall be determined through a systematic approach on the basis of:

- Their importance to safety;
- Their inherent reliability;
- Their potential for degradation (based on operating experience, research and vendor recommendation);
- Operational and other relevant experience and results of condition monitoring.

2.2 In-service inspections of nuclear power plants shall be carried out at intervals whose length shall be chosen in order to ensure that any deterioration of the most exposed component is detected before it can lead to failure.

2.3 Data on maintenance, testing, surveillance, and inspection of SSCs shall be recorded, stored, and analysed. Such records shall be reviewed to look for evidence of incipient and recurring failures, to initiate corrective maintenance and review the preventive maintenance programme accordingly.

2.4 The maintenance programme shall be periodically reviewed[39] in light of operating experience, and any proposed changes to the programme shall be assessed to analyse their effects on system availability, their impact on plant safety, and their conformance with applicable requirements.

2.5 The potential impact of maintenance upon plant safety shall be assessed.

[39] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

3. Implementation

3.1 SSCs important to safety shall be designed to be tested, maintained, repaired, and inspected or monitored periodically in terms of integrity and functional capability over the lifetime of the plant, without undue risk to workers and significant reduction in system availability. Where such provisions cannot be attained, proven alternative or indirect methods shall be specified and adequate safety precautions taken to compensate for potential undiscovered failures.

3.2 Procedures shall be established, reviewed, and validated for all maintenance, testing, surveillance and inspection tasks.

3.3 A comprehensive work planning and control system shall be implemented to ensure that maintenance, testing, surveillance and inspection work is properly authorized and carried out according to the procedures.

3.4 Before equipment is removed from or returned to service, full consideration and approval of the proposed reconfiguration shall be ensured, followed by a documented confirmation of its correct configuration and, where appropriate, functional testing.

3.5 The actions to be taken in response to deviations from the acceptance criteria in the maintenance, testing, surveillance and inspection tasks shall be defined in the procedures.

3.6 Repairs to SSCs shall be devised, authorized, and carried out as promptly as practicable. Priorities shall be established with account taken first of the relative importance to safety of the defective structure, system, or component.

3.7 Following any abnormal event, the licensee shall revalidate the safety functions and functional integrity of any component or system that may have been challenged by the event and carry out any necessary remedial actions, including inspection, testing, maintenance, and repair, as appropriate.

3.8 The reactor coolant pressure boundary shall be subject to a system leakage test before resuming operation after a reactor outage in the course of which its leaktightness may been affected.

3.9 The reactor coolant pressure boundary shall be subject to a system pressure test at or near the end of each major inspection interval.

3.10 All items of equipment used for examinations and tests together with their accessories shall be qualified and calibrated before they are used. All equipment shall be properly identified in the calibration records, and the validity of the calibration shall be regularly verified by the licensee in accordance with the quality management system.

3.11 Any in-service inspection process shall be qualified[40], in terms of required inspection area(s), method(s) of non-destructive testing, defects being sought and required effectiveness of inspections.

[40] The ISI system qualification means to demonstrate that the combination of equipment, inspection procedure and personnel is appropriate for testing of a given inspection area according to a technical specification. It is recommended to uses as reference documents, eg the European Regulators Common Position on NDT Qualification, ENIQ methodology and/or IAEA – EBP-VVER-11 documents

3.12 When a detected flaw that exceeds the acceptance standards is found in a sample, additional examinations shall be performed to investigate the specific problem area in the analysis of additional analogous components (or areas). The extent of further examinations shall be decided with due regard for the nature of the flaw and degree to which it affects the nuclear safety assessments for the plant or component and the potential consequences.

3.13 Surveillance measures to verify the containment integrity shall include: a) leak rate tests; b) tests of penetration seals and closure devices such as air locks and valves that are part of the boundaries, to demonstrate their leaktightness and, where appropriate, their operability; c) inspections for structural integrity (such as those performed on liner and pre-stressing tendons).

Issue LM: Emergency Operating Procedures and Severe Accident Management Guidelines	
Safety area: Operation	Document status: Final

1. *Objectives*

1.1 A comprehensive set of emergency operating procedures (EOPs) for design basis accidents (DBAs) and beyond design basis accidents (BDBAs), and also guidelines for severe accident management (SAMG) shall be provided.

2. *Scope*

2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the plant state to a safe condition.

2.2 EOPs shall be provided to cover Beyond Design Basis Accidents up to, but not including, the onset of core damage. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent core damage.

2.3 SAMGs shall be provided to mitigate the consequences of severe accidents in case that the measures to re-establish or compensate for lost safety functions are not successful.

2.4 EOPs for Design Basis Accidents shall be symptom-based or a combination of symptom based and event based[41] procedures. EOPs for Beyond Design Basis Accidents shall be only symptom based.

3. *Format and Content of Procedures and Guidelines*

3.1 EOPs shall be developed in a systematic way and shall be supported by plant specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, especially alarm response procedures and severe accident management guidelines.

3.2 EOPs shall enable the operator quickly to recognise the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate between EOPs and to proceed from EOPs to SAMGs.

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[41] Event-based EOPs enable the operator to identify the specific event and encompass:
- Information from significant plant parameters,
- Automatic actions that will probably be taken as a result of the event,
- Subsequent operator actions directed to returning the reactor to a normal condition or to provide for safe, extended, and stable shutdown conditions.

Symptom-based EOPs enable the operator to respond to situations for which there are no procedures to identify accurately the event that has occurred. The decisions for measures to respond to such situations are specified in the procedures with respect to the symptoms and the state of systems of the plant (such as the values of safety parameters and critical safety functions).

3.3 SAMGs shall be developed in a systematic way using a plant specific approach. SAMGs shall address strategies to cope with scenarios identified by the severe accident analyses[42].

4. Verification and validation

4.1 EOPs and SAMGs shall be verified and validated in the form in which they will be used in the field, so far as practicable, to ensure that they are administratively and technically correct for the plant and are compatible with the environment in which they will be used.

4.2 The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations, using a simulator, where appropriate.

5. Review and updating of EOPs and SAMGs

5.1 EOPs and SAMGs shall be kept updated to ensure that they remain fit for their purpose.

6. Training

6.1 Shift personnel and on-site technical support shall be regularly trained and exercised, using simulators for the EOPs and, where practicable, for the SAMGs.

6.2 The transition from EOPs to SAMGs for management of severe accidents shall be exercised.

6.3 Interventions called for in SAMGs and needed to restore necessary safety functions shall be planned for and regularly exercised.

[42] Severe accident conditions, for which means shall be provided, are defined in issue E (Verification and Improvement of the Design) in Reference Levels 5.3 to 5.9. It is understood that for these accident conditions also SAMGs shall be developed.

Issue N: Contents and updating of Safety Analysis Report (SAR)	
Safety area: Safety Verification	Document status: Final

1. Objective

1.1 The Licensee shall provide a SAR[43] and use it as a basis for continuous support of safe operation.

1.2 The Licensee shall use the SAR as a basis for assessing the safety implications of changes to the plant or to operating practices.

2. Content of the SAR

2.1 The SAR shall describe the site, the plant layout and normal operation; and demonstrate how safety is achieved.

2.2 The SAR shall contain detailed descriptions of the safety functions; all safety systems and safety-related structures, systems and components; their design basis and functioning in all operational states, including shut down and accident conditions.

2.3 The SAR shall identify applicable regulations codes and standards.

2.4 The SAR shall describe the relevant aspects of the plant organization and the management of safety.

2.5 The SAR shall contain the evaluation of the safety aspects related to the site.

2.6 The SAR shall outline the general design concept and the approach adopted to meet the fundamental safety objectives.

2.7 The SAR shall describe the safety analyses performed to assess the safety of the plant in response to postulated initiating events against safety criteria and radiological release limits.

2.8 The SAR shall describe the emergency operation procedures and accident management guidelines, the inspection and testing provisions, the qualification, and training of personnel, the operational experience feedback programme, and the management of ageing.

2.9 The SAR shall contain the technical bases for the operational limits and conditions.

2.10 The SAR shall describe the policy, strategy, methods, and provisions for radiation protection.

2.11 The SAR shall describe the emergency preparedness arrangements.

2.12 The SAR shall describe the on-site radioactive waste management provisions.

[43] A consistent safety document or integrated set of documents constituting the licensing basis of the plant and updated under control of the regulatory body.

2.13 The SAR shall describe how the relevant decommissioning and end-of-life aspects are taken into account during operation.

3. Review and update of the SAR

3.1 The licensee shall update the SAR to reflect modifications, new regulatory requirements, and relevant standards, as soon as practicable after the new information is available and applicable.

Issue O: Probabilistic Safety Analysis (PSA)	
Safety area: Safety Verification	Document status: Final

1. Scope and content of PSA

1.1 For each plant design, a specific PSA shall be developed for levels 1 and 2, including all modes of operation, all relevant initiating events, and hazards, including internal fire, internal flooding, severe weather conditions, and seismic events.

1.2 PSA shall include relevant dependencies[44].

1.3 The basic Level 1 PSA shall contain uncertainty and sensitivity analyses; the basic Level 2 PSA shall contain uncertainty or sensitivity analyses.

1.4 PSA shall be based on a realistic modelling of plant response, using data relevant for the design, and taking into account human action to the extent assumed in operating and accident procedures.

1.5 Human reliability analysis shall be performed, taking into account the factors that can influence the performance of the operators in all plant states.

2. Quality of PSA

2.1 PSA shall be performed, documented, and maintained according to the quality management system of the licensee.

2.2 PSA shall be performed according to best international practice.

3. Use of PSA

3.1 PSA shall be used to support safety management. Its role in the decision making process shall be defined.

3.2 PSA shall be used[45] to identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant.

3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects"[46].

3.4 PSA shall be used to assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational occurrences.

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[44] Such as functional dependencies, area dependencies (based on the physical location of the components) and other common cause failures

[45] It is intended that such analyses will be done on a continuous basis, not just every ten years during the Periodic Safety Review.

[46] Small deviations in the plant parameters that could give rise to severely abnormal plant behaviour.

3.5 Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.

3.6 The results of PSA shall be used to check that the items with greatest risk are included in the inspection programmes.

4. Demands and conditions on the use of PSA

4.1 The limitations of PSA shall be understood, recognized, and taken into account in all its use. The adequacy of a particular PSA application shall always be checked with respect to these limitations.

4.2 When PSA is used, for evaluating or changing the requirements on periodic testing and allowed outage time for a system or a component, all relevant items, including states of systems and components and safety functions they participate in, shall be included in the analysis.

4.3 The operability of components that have been found by PSA to be important to safety shall be ensured and their role shall be recorded in the SAR.

Issue P: Periodic Safety Review (PSR)	
Safety area: Safety Verification	Document status: Final

1. Objective of the Periodic Safety Review

1.1 The licensee shall have the prime responsibility for performing the Periodic Safety Review.

1.2 The review shall confirm the compliance of the plant with its licensing basis and any deviations shall be resolved.

1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and best international practices.

1.4 All reasonably practicable improvement measures shall be taken by the licensee as a result of the review.

1.5 An overall assessment of the safety of the plant shall be provided, and adequate confidence in plant safety for continued operation demonstrated, as a result of the full scope review.

2. Scope of the Periodic Safety Review

2.1 The review shall be made periodically, at least every ten years.

2.2 The scope of the review shall be clearly defined and justified. The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant and, as a minimum the following areas shall be covered by the review:

- Plant design as built and actual condition of systems, structures and components;
- Current safety analyses and their use;
- Operating experience during the review period and the effectiveness of the system used for experience feed-back;
- Organizational arrangements;
- Staffing and qualification of staff;
- Emergency preparedness; and
- Radiological impact on the environment.

3. Methodology of the Periodic Safety Review

3.1 The review shall use an up to date, systematic, and documented methodology, taking into account deterministic as well as probabilistic assessments.

3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices.

Issue Q: Plant modifications	
Safety area: Operation	Document status: Final

1. Purpose and scope

1.1 The licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely.

1.2 The licensee shall control plant modifications using a graded approach with appropriate criteria for categorization according to their safety significance[47].

2. Procedure for dealing with plant modifications

2.1 The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met.

2.2 For modifications to SSC, this process shall include the following:

- Reason and justification for modification;
- Design;
- Safety assessment;
- Updating plant documentation and training;
- Fabrication, installation and testing; and
- Commissioning the modification.

3. Requirements on safety assessment and review of modifications

3.1 Before starting a modification, an initial safety assessment shall be carried out to determine any consequences for safety.

3.2 A detailed, comprehensive safety assessment shall be undertaken, unless the results of the initial safety assessment show that the scope of this assessment can be reduced.

3.3 Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met.

3.4 The scope, safety implications, and consequences of proposed modifications shall be reviewed by personnel not immediately involved in their design or implementation.

4. Implementation of modifications

4.1 Implementation and testing of plant modifications shall be performed in accordance with relevant work control and plant testing procedures.

[47] Para 4.4 of IAEA Guide NS-G-2.3 contains information about possible categories.

4.2 The impact upon procedures, training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated.

4.3 Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated.

5. Temporary modifications[48]

5.1 All temporary modifications shall be clearly identified at the point of application and at any relevant control position. Operating personnel shall be clearly informed of these modifications and of their consequences for the operation of the plant.

5.2 Temporary modifications shall be managed according to specific plant procedures.

5.3 The number of simultaneous temporary modifications shall be kept to a minimum. The period of a temporary modification shall be limited.

5.4 The licensee shall periodically review outstanding temporary modifications to determine whether they are still needed.

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[48] Examples of temporary modifications are temporary bypass lines, electrical jumpers, lifted electrical leads, temporary trip point settings, temporary blank flanges and temporary defeats of interlocks. This category of modifications also includes temporary constructions and installations used for maintenance of the design basis configuration of the plant in emergencies or other unanticipated situations. Temporary modifications in some cases may be made as an intermediate stage in making permanent modifications. IAEA Guide NS-G-2.3, Para 6.1

Issue R: On-site Emergency Preparedness	
Safety area: Emergency Preparedness	Document status: Final

1. Objective

1.1 The licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) Regaining control of any emergency arising at their site, including events related to combinations of non-nuclear and nuclear hazards;

(b) Preventing or mitigating the consequences at the scene of any such emergency; and

(c) Co-operating with external emergency response organizations in preventing adverse health effects in workers and the public.

2. Emergency Preparedness and Response Plan

2.1 The licensee shall prepare a site emergency plan and establish the necessary organizational structure for clear allocation of responsibilities, authorities, and arrangements for co-ordinating plant activities and co-operating with external response agencies throughout all phases of an emergency.

2.2 The licensee shall provide for:

(a) Prompt recognition and classification of emergencies;

(b) Timely notification and alerting of response personnel;

(c) Ensuring the safety of all persons present on the site, including the protection of the emergency workers;

(d) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(e) Performing assessments of the situation on the technical, & radiological points of view (on and off site);

(f) Monitoring radioactive releases;

(g) Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons; and

(h) Plant management and damage control[49].

2.3 The site emergency plan shall be based upon an assessment of reasonably foreseeable events and situations that may require protective measures on- or off-site. The plan shall also be co-ordinated with all other involved bodies and capable of extension should more improbable, severe events occur.

[49] Understood as urgent mitigatory repairs, controls, and other actions that are carried out, primarily at the site, while the emergency is still in progress.

3. Organization

3.1 The licensee shall have people on-site at all times with the authority and responsibilities to classify and declare an emergency and, upon classification, to initiate promptly the appropriate on-site response.

3.2 Sufficient numbers of qualified personnel shall be available at all times for staffing appropriate positions promptly following the declaration and notification of an emergency.

3.3 Arrangements shall be made to provide technical assistance to operational staff. Teams for mitigating the consequences of an emergency (eg radiation protection, damage control, fire fighting, etc) shall be available.

3.4 Arrangements shall be made to alert police, medical, and off-site fire-fighting services promptly.

3.5 The licensee shall identify those who are authorized to carry out the response functions assigned in the emergency plan.

4. Facilities and equipment

4.1 Appropriate emergency facilities shall be designated for responding to events on site and that will provide off-site monitoring and assessment throughout different phases of an emergency response.

4.2 An "On-site Emergency Control Centre", separated from the plant control room, shall be provided for on-site emergency management staff. Important information shall be available in the control centre about the plant and radiological conditions on and around the site. The centre shall have means of communicating with the control room, any supplementary control room, other important points on site, and with the on-site and off-site emergency response organizations[50].

4.3 Emergency facilities shall be suitably located and protected to enable the exposure of emergency workers to be controlled. Appropriate measures shall be taken to protect those occupying emergency facilities for a protracted time from hazards resulting from accidents[51].

4.4 Instruments, tools, equipment, documentation, and communication systems for use in emergencies shall be kept available and tested sufficiently frequently to demonstrate that they are in good working condition where they are unlikely to be affected by postulated accidents.

5. Training, drills and exercises

5.1 Arrangements shall be made to identify the knowledge, skills, and abilities needed for personnel to perform their assigned response functions.

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[50] The *On-site Emergency Control Centre* is the office accommodation and associated office services set aside on or near to the site for staff who are brought together to provide technical support the Operations staff during an emergency. It may have plant information systems available, but is not expected to have any plant controls.

[51] This refers, primarily, to ensuring that the *On-site Emergency Control Centre* and other locations where staff are expected to spend a significant time are located somewhere that the staff can reach and work throughout an extended emergency with minimum risk to health. This will require location away from areas that are likely to be damaged of affected by radiation fields and, where appropriate, this will include provision of recirculatory air conditioning and continuous radiation monitoring systems.

5.2 Arrangements shall be made to inform all employees and all other persons present on the site of the actions to be taken in the event of an emergency.

5.3 Training arrangements shall include basic emergency training and ongoing refresher training on an appropriate schedule and shall ensure that emergency response personnel meet the training obligations.

5.4 The site emergency plan shall be exercised at least annually. Some exercises shall be integrated to include as many as possible of the off-site organizations concerned.

5.5 Emergency exercises shall be evaluated systematically, and the emergency preparedness arrangements and the plan shall be subject to review and updating in the light of experience gained.

Issue S: Protection against internal fires	
Safety area: Emergency Preparedness	Document status: Final

1. Fire safety objectives

1.1 The licensee shall implement the defence in depth principle to fire protection, providing measures to prevent fires from starting, to detect and extinguish quickly any fires that do start and to prevent the spread of fires in or to any area that may affect safety[52].

2. Basic design principles

2.1 SSCs important to safety shall be designed and located so as to minimize the probabilities and the effects of fire and to maintain capability for shutdown, residual heat removal, confinement of radioactive material and monitoring of plant state during and after a fire event.

2.2 Buildings that contain equipment that is important to safety shall be designed as fire resistant, subdivided into compartments that segregate such items from fire loads and segregate redundant safety systems from each other[53]. When a fire compartment approach is not practicable, fire cells shall be used[54], providing a balance between passive and active means, as justified by fire hazard analysis.

2.3 Buildings that contain radioactive materials, or that could affect the safety of plant in the event of a fire, shall be fire resistant.

2.4 Access and escape routes for fire fighting and emergency operating personnel shall be available.

3. Fire hazard analysis

3.1 A fire hazard analysis shall be carried out and kept updated to demonstrate that the fire safety objectives are met, that the fire safety principles are satisfied, that the fire protection systems are appropriately designed and that any necessary administrative provisions are properly implemented.

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[52] In this context, safety refers to all sources of nuclear safety risk, including radioactive waste facilities.

[53] A fire compartment is a building or part of building that is completely surrounded by fire resistant barriers of sufficient rating so that a total combustion of the fire load can occur without breaching the barriers. (Barriers comprise doors, walls, floors and ceilings.) The fire resistance rating of the barriers must be sufficiently high so that the total combustion of the fire load in the compartment can occur without breaching the barriers.

[54] Fire cells limit the spread of fire through the restriction of combustible material; separation of items by distance; and by the provision of extinguishing systems and passive fire protection (e.g. Shields, wraps).

3.2 The fire hazard analysis shall be developed on a deterministic basis, covering at least:

- For all normal operating and shutdown states, a single fire and consequential spread anywhere that there is fixed or transient combustible material;
- Consideration of appropriate combination of fire and other PIEs likely to occur independently of a fire.

3.3 The fire hazard analysis shall demonstrate how the possible consequential effects of fire and extinguishing systems operation have been taken into account.

3.4 The fire hazard analysis shall be complemented by probabilistic fire analysis. Together with initiating events analysed in PSA level 1, the fires shall be assessed in order to evaluate the fire protection arrangements and to identify risks caused by fires.

4. Fire protection systems

4.1 Each fire compartment or fire cell shall be equipped with fire detection and alarm features, with detailed annunciation for the control room staff of the location of a fire. These features shall be provided with non-interruptible emergency power supplies and appropriate fire resistant supply cables.

4.2 Fixed or mobile, automated or manual extinguishing systems shall be installed. They shall be designed and located so that their rupture, spurious or inadvertent operation does not significantly impair the capability of SSCs important to safety to carry out their safety functions.

4.3 The distribution loop for fire hydrants shall provide exterior coverage of the buildings. Internal standpipes shall provide complete coverage of the interior areas of the plant.

4.4 Ventilation systems shall be arranged such that each fire compartment fully fulfils its segregation purpose in case of fire.

4.5 Parts of ventilation systems (such as connecting ducts, fan rooms and filters) that are located outside fire compartments shall have the same fire resistance as the compartment or be capable of isolation from it by appropriately rated fire dampers.

5. Administrative controls and maintenance

5.1 Procedures shall be established to control and minimize the amount of combustible material and minimize ignition sources that may affect items important to safety, and to establish inspection, maintenance and testing of fire barriers, fire detection and extinguishing systems.

6. Fire fighting organization

6.1 The licensee shall implement adequate arrangements for controlling and ensuring fire safety, as identified by the fire hazard analysis[55]

[55] Such arrangements must include nominating persons to be responsible for or have duties with respect to fire protection. The arrangements must set out the requirements for control of all activities that can have impact on

6.2 Written emergency procedures that clearly define the responsibility and actions of staff in responding to any fire in the plant shall be established and kept up to date. A fire fighting strategy shall be developed, kept up-to date, and trained for, to cover each area in which a fire might affect items important to safety and protection of radioactive materials.

6.3 When reliance for manual fire fighting capability is placed on an offsite resource, there shall be proper coordination between the plant personnel and the off site response group, in order to ensure that the latter is familiar with the hazards of the plant.

6.4 If plant personnel are required to be involved in fire fighting, their organization, minimum staffing level, equipment, fitness requirements, and training shall be documented and their adequacy shall be confirmed by a competent person.

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fire safety, e.g. Maintenance; control of materials; training; tests and drills; modifications to layouts and systems – such as fire detection, fire extinguishing, ventilation, electrical and control systems.

Annex 2 – Panel procedure

Introduction

The objectives of the panel procedure were:

- To ensure consistency between the countries and, in particular, to check that the Reference Levels have been interpreted in the same manner and with the same stringency;
- To give the opportunity to clarify the interpretation of some Reference Levels if they appear to be misunderstood;
- To give the opportunity to modify or even delete some Reference Levels with overlapping information; and
- Overall, to establish a peer review process that would increase the reliability of the study results by making sure that the national positions rely on accepted justifications.

Because the working group had increased to 17 countries, it was necessary to modify the panel procedure in relation to that used in the Pilot Study. Therefore, the panel sessions were conducted in parallel for two groups of countries, comprising nine and eight countries respectively. During these parallel sessions, notes were taken of all feedback from the panel to individual countries and of any problems with the Reference Levels that were encountered during the benchmarking. After the parallel sessions, a plenary session was held to discuss the issues raised in the parallel groups and to decide on any changes of the Reference Levels. After these plenary sessions, the Reference Levels were amended by authors and declared "frozen", and each country revised their self-assessments in accordance with the specific comments from the panel and the final version of the Reference Levels. The composition of the parallel country groups was changed between the meetings according to a rotating schedule, to ensure cross-fertilization of experience.

For the panel sessions, the following procedure was used:

- Each of the 18 safety issues (see Table 1 in Section 3 of the report) was examined for each of the 17 participating countries during a full panel session (parallel groups plus plenary). This means that all countries were benchmarked in this study for the issues that had already been benchmarked in the Pilot Study, even those that participated in the pilot study;
- National self-assessments for the upcoming meeting were sent in advance to all participants, ideally two weeks before the meeting, to allow participants to prepare for the meeting, including sending comments before it;
- At the panel meeting, a computer and projector was used to display the assessment under review onto a screen, so that the group of countries could follow progress through the document.
- Each country presented their assessment briefly and commented on their national position. Also, comments on the Reference Levels were given;

- After the presentation, the case was open for questioning and comments by the other participants; the justifications were scrutinized; and the panel agreed or requested further justification for the case put forward. Comments from the panel were made directly in the assessment file to assist the country and keep a record using the computer. When the panel remained divided upon a position, a vote was conducted;
- Notes were also made on proposed revisions of the Reference Levels in the light of problems encountered during the benchmarking;
- The chair of the panel session summarized the comments given to the individual country;
- At the following plenary session, the chairs of the two parallel groups presented the outcome of the panel sessions and the issues raised in the two groups about any technical difficulties with assessing the Reference Levels. (It was noted that the problems raised by the two groups with respect to the Reference Levels and assessment process were very similar, indicating further that the peer review process gave a valid method for cross-checking each other and giving confidence that the country groups were large enough to cover all aspects.);
- The plenary group decided whether the benchmarking process had identified the need for any amendments to the Reference Levels. Changes were only made where levels were difficult to interpret, overlapped other levels, or where they addressed too many aspects in a single Reference Level;
- Reference Levels were amended by their lead authors and frozen; and
- All participants revised their self-assessments, as needed, and distributed them to the whole group.

Treatment of backlogs

Some countries accumulated a backlog during the process and were not able, for various reasons, to participate in all panel sessions. Therefore, a number of backlog panels were conducted at the end of the project. During these backlog sessions, the same rules for acceptance were used as for the original panels. In the end, all issues were benchmarked and peer-reviewed for all countries.

Quality checks

It was also necessary before the finalizing the benchmarking activity, to carry out a quality check, to ensure that all comments had been dealt with properly and that the assessments were consistent with regard to the information required in the '**B**' and '**C**' assessments. This was done in two steps:

1. A self-check using stricter written guidelines; and
2. An independent check done by an appointed group of participants.

Because of these checks, several final modifications and amendments were made.

Annex 3 – Descriptions of the national legal systems

Belgium

The legislative and regulatory framework has been put progressively in place since 1955. The law of 15 April 1994, replacing the law of 29 March 1958, very generally outlines the protection of the population and the environment against the dangers of ionising radiation. The detailed stipulations are given in the Royal Decree (R.D.) of 20 July 2001, replacing the R.D. of 28 February 1963, "providing the General Regulations regarding protection of the population, workers, and environment against the dangers of ionising radiation".

In 1975, when the decision was taken to build four more nuclear units (Doel 3-Tihange 2 and Doel 4-Tihange 3), the Belgian Nuclear Safety Commission decided that the American nuclear safety rules would be applied, and this according to a schedule consistent with their date of issue, and that a number of external accidents be considered in a deterministic manner (crash of civil or military aircraft, gas explosion, toxic cloud, large fire…). The whole safety analysis of these units was conducted on these bases, applying the USNRC regulation and guidance. Deviations, if accepted, were documented.

The licence of each nuclear power plant takes the form of a Royal Decree of Authorisation. It stipulates that the plant has to be in conformity with its Safety Analysis Report (SAR) and that only minor modifications are allowed without a formal licensing process (minor modifications are defined as those having either no impact on the safety, or that are safety improvements). This means that the SAR is legally binding, that no exemptions are allowed, but that the SAR can be modified if the modification is minor and if it is approved by the authorised inspection organisation (AVN). However, since the SAR is not a public document, it was not credited as national requirement according to the criteria of this study. Important modifications must go through the whole licensing process. No time limit is mentioned in the licence, but a periodic safety reassessment is required every ten years.

The law of 15 April 1994 has created the Federal Agency for Nuclear Control (FANC) and defines the missions entrusted to this agency, regrouping most of the activities previously held by the relevant Ministries. The various Articles of that law were gradually brought into force as needed, and the FANC became completely operational on 1 September 2001. According to the law of 15 April 1994, the FANC appoints the authorized inspection organisations in charge of the regulatory inspections of nuclear installations. AVN is the authorized inspection organisation for the nuclear power plants (as well as for a number of other nuclear installations).

Concerning emergency planning, a specific Royal Decree dated 17.10.2003 describes, amongst others, requirements of the licensees with respect to the nuclear and radiological emergency plan for the Belgian territory.

More information on the Belgian legislative and regulatory system can be found in the Belgian report to the Nuclear Safety Convention, available on the AVN web site: www.avn.be.

Bulgaria

In the Republic of Bulgaria, the Parliament has the authority to adopt legislative acts, while the Government adopts the secondary legislation for implementation of the laws. The rules and regulations are promulgated by a governmental decree. Each governmental authority issues instructions to provide directions and guidance concerning the implementation of the legislation.

The **Safe Use of Nuclear Energy Act** (Law), SUNEA, 2002, is the basic legislative act in the use of nuclear energy. It stipulates the state regulation of the safe use of nuclear energy and ionising radiation, and the safety of radioactive waste and spent fuel management. The responsibilities of the licensees for ensuring nuclear safety and radiation protection are specified there as well.

With regard to the safety of nuclear power plants (NPPs) and the sources of ionizing radiation, the secondary legislation comprises 19 regulations on the application of the SUNEA requirements. The following regulations relate to reactor safety and have been used in the benchmarking:

Regulation for providing the safety of nuclear power plants, promulgated in 2004, which settles provisions related to the basic criteria and rules for NPP safety based on the defence in-depth concept. Subject to regulation are the organizational measures and technical requirements for providing the safety during site selection, design, construction, commissioning, and operation of NPPs;

Regulation for the procedure for issuing licenses and permits for safe use of nuclear energy, promulgated in 2004, which defines all matters related to the procedures for issuing, changing, renewing, cancelling, revoking and controlling the licenses and permits;

Regulation of the conditions and procedure for notification of the NRA about events in nuclear facilities and sites with sources of ionizing radiation, promulgated in 2004, which specifies the responsibilities for creation of a system for collecting, registration, investigation, analysis and evaluation of events and identification of corrective measures;

Regulation for emergency planning and emergency preparedness in case of nuclear and radiation accident, promulgated in 2004, which defines the conditions and procedure for developing emergency plans, the responsibilities of persons and authorities, measures for mitigation of the consequences of nuclear or radiation accident, the decision making criteria;

Regulation of the conditions and procedure for acquiring professional qualification and for the procedure for issuing licenses for specialized training and certificates for qualification for use of nuclear energy, promulgated 2004, which sets the requirements for acquiring professional qualification for execution of activities in nuclear facilities, the positions and the procedure for issuing certificates for qualification;

Regulation for the safety of the decommissioning of nuclear facilities, promulgated in 2004, which comprises the requirements for decommissioning.

The regulatory body for nuclear safety in Bulgaria is the Nuclear Regulatory Agency (NRA). The NRA Chairman is an independent specialized authority of the executive power and is vested with competencies for state regulation of the safe use of nuclear energy and ionizing radiation, and the safety of radioactive waste management and spent fuel management as specified by SUNEA.

More information is available at the web site of the Bulgarian Nuclear Regulatory Agency:

www.bnsa.bas.bg.

Czech Republic

The development of the current State supervision is connected with the establishment of the independent state Czech Republic at the turn of 1992–1993. The Act No. 21/1992 Coll. established the State Office for Nuclear Safety (SÚJB), which started to develop new comprehensive nuclear legislation.

The Atomic Act (Act No. 18/1997 Coll., on peaceful utilization of nuclear energy and ionizing radiation) was approved in January 1997. The Atomic Act entrusted execution of the state administration and supervision of peaceful utilization of nuclear energy and radiation practices to SÚJB and redefined the scope of its competency and powers.

The Atomic Act has defined conditions for peaceful utilization of nuclear energy and ionizing radiation, including the activities requiring SÚJB license. An extensive list of obligations of the licensees includes, among other items, obligations relating to their preparedness for a radiation accident. Since 1997, the Atomic Act has been amended several times. The most significant amendment was performed by the Act No. 13/2002 Coll., which was particularly adopted in connection with the preparation of the Czech Republic for accession to the European Union, aimed at enabling the implementation of obligations arising from newly concluded international treaties.

The Atomic Act authorized the SÚJB to issue a set of related implementing regulations; the main ones are as follows:

- Regulation No. 214/1997 concerning the quality assurance in activities related to the utilization of nuclear energy and in radiation practices;
- Regulation No. 215/1997 for the siting of nuclear installations and very significant ionizing radiation sources;
- Regulation No. 106/1997 for the commissioning and operation of nuclear facilities;
- Regulation No. 195/1999 to the basic design criteria for nuclear installations;
- Regulation No. 185/2003 to the decommissioning of nuclear installation;
- Regulation No. 146/1997 as amended by SÚJB Regulation No. 315/2002 to the requirements on qualification and professional training of selected personnel;
- Regulation No. 307/2003 to the radiation protection criteria and methodology;
- Regulation No. 318/2002 to the details of emergency preparedness of nuclear installations and on-site emergency plans and emergency rules;
- Regulation No. 319/2002 to the performance and management of the national radiation monitoring network; and
- Regulation No. 240/2000 on the crisis management and the emergency planning zones.

SÚJB is authorized to require the inspected person to remedy the situation, to perform technical checks, inspections, or functional ability tests, to withdraw authorizations about special professional competence and to impose penalties for violating obligations established in the Atomic Act or to suspend operation of the nuclear installation.

A complete text of the Atomic Act, including its implementing decrees is available on the SÚJB web site www.sujb.cz.

Finland

In Finland, the legislation for the use of nuclear energy and for radiation protection was established in 1957. In 1987, a completely revised Nuclear Energy Act was issued, together with a supporting Nuclear Energy Decree (1988). The Radiation Act and Decree were revised in 1991. The acts and decrees are regularly updated, as necessary.

Based on the Nuclear Energy Act, the Government has issued the following decisions on:

- General Regulations for the Safety of Nuclear Power Plants (395/1991)
- General Regulations for Physical Protection of Nuclear Power Plants (396/1991)
- General Regulations for Emergency Response Arrangements at Nuclear Power Plants (397/1991)
- General Regulations for the Safety of a Disposal Facility for Reactor Waste (398/1991)
- Safety of Disposal of Spent Nuclear Fuel (478/1999).

The general regulations 395/1991, 396/1991 and 397/1991 are applied to a nuclear power plant, which is defined to be a nuclear facility equipped with a nuclear reactor and intended for electricity generation. The general regulations are also applied to other nuclear facilities to the extent applicable.

Detailed safety requirements are provided in YVL Guides. YVL Guides also provide administrative procedures for regulation of the use of nuclear energy. YVL Guides are issued by STUK, as stipulated in the Nuclear Energy Act. The publication of an YVL guide does not, as such, alter any decisions made by STUK before the publication of the guide. It is only after STUK has heard those concerned that STUK makes a separate decision on how a new or revised YVL guide is applied to operating nuclear power plants, or to those under construction, and to the licence-holders' activities. The guides apply as such to new nuclear facilities.

When STUK considers how new safety requirements presented in the YVL guides apply to operating nuclear power plants, or to those under construction, STUK takes into account the principle prescribed in section 27 of the Government Decision (395/1991), according to which for further safety enhancement, actions shall be taken which can be regarded as justified considering operating experience and the results of safety research as well as the advancement of science and technology.

If exemptions from the requirements of the YVL guides are needed, STUK shall be presented with an acceptable procedure or solution by which the safety level set forth in the YVL guides is attained.

More information about STUK and Finnish regulations can be obtained at:

http://www.stuk.fi/english

France

The organisation for nuclear safety in France relies on the principle of the prime responsibility of the operator. The legal basis regulating the safety of nuclear installations in France is the law 61-842 of 2 August 1961, which states that industrial premises shall be operated as to prevent pollutions of any type, which could compromise public health or security. Taken for the implementation of this law, the decree 63-1228 of 11 December 1963, as amended, concerning nuclear installations constitutes the basis of the nuclear safety regulations. Its article 2 defines the Basic Nuclear Installations (BNI), which are subject to the above-mentioned regulations, in particular which comprise all civilian nuclear power reactors. In addition, its article 10 sets the principle of the general technical regulation with respect to BNIs safety.

More recently, the Decree 2002-255 of 22 February 2002, modifying decree 93-1272 of 1 December 1993, created a new Directorate General for Nuclear Safety and Radiation Protection (DGSNR), which has taken the place of previously existing regulatory bodies, and whose duties have been extended to ensure that all users of ionising radiation fully comply with their responsibilities and obligations with regard to nuclear safety, as well as to radiation protection.

Basic Nuclear Installations (BNIs) are subject to two particular types of regulations: licensing procedures and general technical regulation.

BNIs are regulated by decree 63-1228 of 11 December 1963, which describes the procedure for the initial licensing, and all the authorisations required for all the lifetime of a plant. Each plant licensing is formulated in a specific decree at different stages of the plant lifetime: installation creation decree, plant's commissioning and decommissioning decrees, and authorisation for dismantling decree. BNIs must also comply with the requirements of decree 95-540 of 4 May 1995, implementing both the above-mentioned law of 2 August 1961 and law 92-3 of 3 January 1992 amended concerning water (articles L.210-1 to L.217-1 of the Environment Code). This decree stipulates the authorisation procedure for liquid and gaseous effluent release and water intake for these installations.

There are two levels that are summarised below: **legally binding regulation**, in the form of ministerial orders, and **general recommendations** such as ministerial letters, circular letters, and basic safety rules (BSR). Legally binding regulation currently covers three major subjects: pressure vessels, quality organisation, and protection of the environment. The ministerial order of 26 February 1974 applies to the particular case of the construction of the main primary system of EDFs PWRs. In service inspection of the main primary system and the main secondary systems of PWRs are covered by the interministerial order of 10 November 1999. The regulations for conventional pressure vessels apply to the other pressure vessels. The ministerial order of 10 August 1984 stipulates the general rules for quality assurance and organisation to be followed by operators at the BNI design, construction, and operating stages. The ministerial order of 31 December 1999 prescribes the general technical regulations for the prevention and limitation of external hazards and detrimental effects related to BNI operation, apart from water intake and effluent release issues

General recommendations are ranked in a series of texts. Firstly, there are ministerial letters, which were issued to the operator for each type of reactors before construction and aimed at defining the regulatory position on the main safety options. Then, come the circular letters, which are mostly written to explain and detail the above ministerial orders. Finally, the basic safety rules (BSR) are issued by the French nuclear safety authority on various technical subjects, concerning both PWRs and other BNIs. These rules constitute recommendations defining the safety aims to be achieved and describing accepted practice the DGSNR deems compatible with these aims. There are currently about forty Basic Safety Rules. Further information can be found on ASN web site: www.asn.gouv.fr

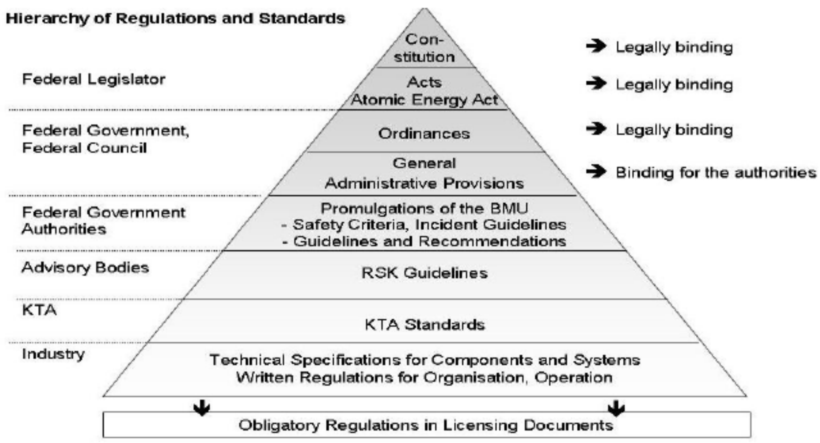
Germany

In accordance with the federal structure of the Federal Republic of Germany, its Constitution (Article 74 (1) 11a Basic Law) contains detailed provisions on the legislative and administrative competencies of the Federation and the individual Federal States (Länder). The Federation has enacted the "Federal Act on the Peaceful Uses of Atomic Energy and Protection against its Hazards" of 1959 – Atomgesetz (AtG) last amended in August 2005. The Act is implemented by the competent Regulatory Authorities of the Länder. The supreme regulatory authority of the Federation, the Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU), supervises the regulatory authorities of the Federal States with respect to compliance with the AtG and to expediency, including the right of instructions.

Under the AtG any person, who constructs operates or substantially alters or decommissions a nuclear power plant needs a license. The AtG contains fundamental licensing prerequisites as for example: necessary precautions against damage in the light of the state of the art in science and technology or trustworthiness and qualification of the responsible personnel. Today, these requirements for the licensing of nuclear power plants are only significant for modifications of existing plants because after the amendment of 2002 licenses for new facilities for the fission of nuclear fuel for the commercial production of electricity will no longer be granted.

Enabled by specific articles of the law legally binding ordinances that contain more detailed procedural or technical requirements can be released. They are drafted by the Federal Government and need the consent of the Federal Council. The following ordinances have been used for benchmarking: Radiation Protection Ordinance (StrlSchV), October 13, 1976, last Amendment 1997, Nuclear Licensing Procedure Ordinance (AtVfV), February 18, 1977, last Amended 1995, Nuclear Safety Officer and Reporting Ordinance (AtSMV), October 14, 1992, last Amendment 2002

The safety provisions and regulations of the Atomic Energy Act and of the associated ordinances are put into concrete terms by regulatory guidelines as "Safety Criteria" of 1977, "Checklist of Layout of a Standard Safety Analysis Report" 1976,), by safety standards of the Nuclear Safety Standards Commission as KTA 1201 "Requirements for the Operating Manual", All these documents are generic and published. Related requirements are not directly legally binding.



Respective generic regulations have been used for benchmarking if they have been enforced by valid licensing or supervisory actions or if the licensing is based on the generic regulation and can therefore be considered as binding obligations of the licensees.

Further information can be found on the web sites: www.bmu.de, www.bfs.de, www.rskonline.de, www.kta-gs.de and http://regelwerk.grs.de/.

Hungary

The first laws and regulations on radioactive materials and radiation therapy in Hungary were issued in 1964, followed by the issuance of several further laws. The Act I of 1980 on Atomic Energy and the Executive Orders on its implementation can be considered as significant milestones.

Based on a Government Decree (No.104/1990), from 1 January 1991 the scope of rights and responsibilities of the Hungarian Atomic Energy Commission (HAEC), as former licensing authority of nuclear installations, were redefined and the Hungarian Atomic Energy Authority (HAEA), a new independent State administration organization, as nuclear safety regulatory body, was established.

The new Atomic Act passed by the Parliament at the end of 1996 (Act CXVI of 1996 on Atomic Energy) and its executive orders (Government Decree 87/1997(V.28) on the duties and scope of authority of the HAEC and of the HAEA and Government Decree 108/1997(VI.25) on the procedures of the HAEA in nuclear safety regulatory matters with attachments of the mandatory Regulatory Requirements in 4 Volumes) introduced further changes in the scope of authority and organizational structure of the national regulatory bodies related to nuclear safety. The Act CXVI of 1996 on Atomic Energy has reinforced the distributed regulatory system, which has delegated the responsibilities of nuclear safety, radiation protection and environmental protection in connection with nuclear facilities to different authorities concerned. Additionally to the legally binding Requirements, the Director General of the HAEA has issued continuously Guidelines containing recommendations on how the Requirements should be implemented in the regulatory processes. The number of these Guidelines is 62 at this time.

In 2003 the Parliament amended the Atomic Act CXVI of 1996, and according to this decision the existence of HAEC was abolished and a dedicated minister (currently the Minister of Justice) appointed by the Prime Minister became the supervisor of the HAEA.

In 2005, a new revised set of Nuclear Safety Requirements (Regulations) was issued as attachment of Government Decree 89/2005(V.5). The legal framework that was used for the benchmarking of the Reference Levels is as follows:

- Act CXVI of 1996 on Atomic Energy;
- Government Decree 108/1997(VI.25).Korm. on the Procedures of the Hungarian Atomic Energy Authority in Nuclear Safety Regulatory Matters, including 4 Volumes of Nuclear Safety Regulations issued as its attachments (nuclear safety requirements for NPPs):
    - Volume 1: Regulatory Procedures for NPPs,
    - Volume 2: Requirements of Quality Management of NPPs,
    - Volume 3: Requirements of Design of NPPs,
    - Volume 4: Requirements of Operation of NPPs.
- Government Decree 89/2005.(V. 5.) Korm. on the Nuclear Safety Requirements of Nuclear Facilities and the Related Regulatory Activities, including 4 Volumes of Nuclear Safety Regulations (as above) revised and issued as its attachments. (Used for self-checking);
- Safety Guides of HAEA.

More information can be found on the web site of the HAEA: www.haea.gov.hu.

Italy

The present Italian Regulatory System related to nuclear installations is the result of an evolution of rules and standards that begun in the early '60s and that took the experience of licensing and operation of nuclear power plants of different types and generation into account. The Italian regulatory system is made up of three types of rules of different legal force depending on their origin; the first two types are the most relevant for this study: legislation by the Parliament and Decrees by Government or Ministries and Technical guides.

a) Main legislation and ministerial decrees; in the Italian system the source, however indirect, of legally binding rules must be either an act of Parliament (statute) or a Legislative Decree; the Government can issue governmental or ministerial decrees binding in law. An important feature of legally binding rules concerning Safety and Radiation Protection is that contravention to obligations by operators and/or users constitutes a misdemeanour and entails a penal sanction; compliance can be enforced by means of criminal proceedings after due process of law.

The main corpus making up, inter alia, the Italian system are itemised below, as regards Statutes and Legislative acts:

- Act no. 1860 of 31 December 1962 published in the Italian Official Journal no. 27 of 30 January 1963, the basic Atomic Law on the peaceful uses of nuclear energy.
- The Presidential Decree no. 185 of 1964: "Safety of plants and protection of workers and general public against the risk of ionising radiation associated to the peaceful use of Nuclear Energy" replaced in 1996 by the Legislative Decree no. 230/1995.
- Legislative Decree no. 230 of 17 March 1995 published in the Supplement to Italian Republic's Official Journal no. 136 of 13 June 1995, implementing six EURATOM Directives on radiation protection (EURATOM 80/836, 84/467, 84/466, 89/618, 90/641 and 92/3).
- Presidential Decree no. 1450 containing requirements and procedures for the acquisition of the operational personnel licences (1971).
- Presidential Decree no. 519/1975 "Civil responsibilities in the field of nuclear safety".
- Legislative Decree no. 241 of 31 August 2000, implementing the 96/29/EURATOM directive regarding "Health protection of the population and workers against the risks deriving from ionising radiations".

Several Acts of legislative force were issued for the institution of the Regulatory Body and for its subsequent re-organisations. The first one was Act no. 933 (1960), establishing the National Committee for Nuclear Energy (CNEN), and the last one was Legislative Decree no. 300 (1999) instituting the Agency for the Environmental Protection and Technical Services (APAT). The mandate of APAT is more generally addressed to Environmental Protection issues; one APAT Department has the mission to discharge the Regulatory Body responsibilities coming from the above-mentioned Laws. In this frame, the Agency performs licensing and inspection activities for any civil Nuclear Installation, performs inspections related to Physical Protection and Safeguards, provides technical support for setting up regulations, for planning and implementing Radiological Emergencies measures.

b) Technical guides; the issue of technical guides, previously carried out by the Directorate for Nuclear Safety and Health Protection, is now assigned in Law to APAT by article 153 of the Legislative Decree no. 230/1995. They contain recommendations and are a tool to implement rules of good practice. 28 technical guides have been issued on Safety and Radiation Protection matters ranging from procedural to detailed technical guidance. They are publicly available and have been issued after consultation of all the stakeholders.

Further information can be found on APAT web site www.apat.gov.it.

Lithuania

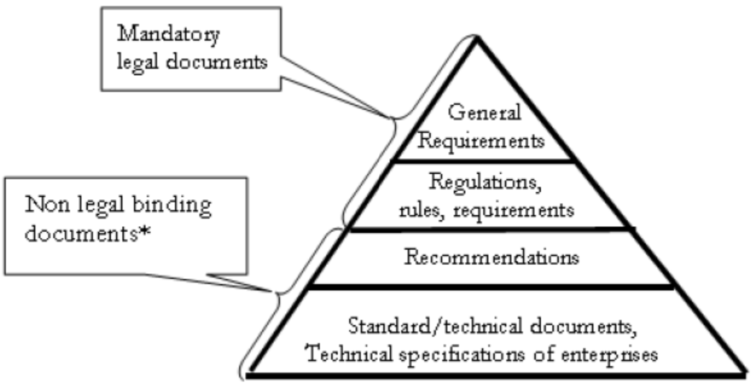
In Lithuania, the scope of the legislation covers the construction, operation and decommissioning of nuclear facilities and control of nuclear materials and wastes. The legal acts are updated, as necessary. The main legal document governing nuclear energy is the **Law on Nuclear Energy** passed by the Seimas in 1996. There are some other laws directly relating to safe operation of nuclear energy, such as the Law on Radioactive Waste Management, the Law on Radiation Protection, the Law on Control of Import, Export and Transit of Strategic Commodities, the Law on Civil Protection, the Law on Construction, etc.

The main regulations in Lithuania are: General Regulations for Nuclear Power Plant Safety, Nuclear Safety Regulations for Reactors of Nuclear Power Plants, Licensing of Nuclear Power Related Activities.

The following documents are under preparation and were used in benchmarking: Requirements for the Ignalina NPP Transient and Accident Analysis, Requirements for Risk Assessment and Management of Risk, Rules on Operational Experience Evaluation at Nuclear Facilities

State Nuclear Power Safety Inspectorate (VATESI), the regulatory body for nuclear safety, has the responsibilities to control the nuclear and radiation safety (partially) of nuclear facilities, radioactive wastes and nuclear materials, as well as of physical protection of nuclear facilities. In addition, VATESI implements emergency preparedness and response functions and organizes the research related to nuclear safety. VATESI is under the administrative control of the Government of the Republic of Lithuania. VATESI responsibilities and authorities are given in the Law of Nuclear Energy (1996) and VATESI Statute (2002). The very important function is establishment of nuclear safety requirements through rules, regulations, and other legal documents. As prescribed by the Law on Nuclear Energy (Article 4, part 2), the standards and regulations confirmed by VATESI are mandatory to all natural and legal persons.

The hierarchic diagram of VATESI documentation is given below:



* These documents become mandatory after the operating organization chooses them and advises VATESI about it.

More information can be found on the web site www.vatesi.lt

The Netherlands

The basic legislation governing nuclear activities is contained in the **Nuclear Energy Act.** This Act is designed as an integral act to cover both the use of nuclear energy and radioactive techniques, as well as to lay down rules for protection against the risks. In practice, however the act has developed virtually as a protection act. The Act sets out the basic rules on nuclear energy, makes provision for radiation protection, designates the various competent authorities, and outlines their responsibilities.

The Minister of Housing, Spatial Planning and the Environment, the Minister of Social Affairs and Employment, and the Minister of Economic Affairs grant licences jointly for nuclear power plants jointly. Together, these ministers form the competent authorities as defined by the Nuclear Energy Act and are jointly responsible for assessing applications. The Minister of Housing, Spatial Planning, and the Environment acts as the coordinator in this respect.

With regard to nuclear energy, the purpose of the Act is to serve the following interests (Article 15b): the protection of people, animals, plants and property; the security of the State; the storage and guarding of fissionable materials and ores; the supply of energy; the payment of compensation for any damage or injury caused to third parties; the observance of international obligations.

A number of decrees have also been issued containing regulations that are more specific. The most important of these in relation to the safety aspects of nuclear installations are:

- The Nuclear Installations, Fissionable Materials and Ores Decree (Bkse), the Radiation Protection Decree (BsK), and
- The Transport of Fissionable Materials, Ores and Radioactive Substances Decree (Bvser).

The Bkse regulates all activities (including licensing) that involve fissionable materials and nuclear installations. The Radiation Protection Decree regulates the protection of the public and workers against the hazards of all ionising radiation, in accordance with the relevant EURATOM Directive.

Pursuant to the Nuclear Energy Act (Article 21.1), a system of rules that are more detailed and regulations has been established in the areas of design, operation, and quality assurance of nuclear power plants. The system is referred to as the Nuclear Safety Rules (NVRs) and has been developed under the responsibility of the Minister of Social Affaires and Employment, and the Minister of Housing, Spatial Planning and the Environment.

The NVRs are based on the Codes and Safety Guides of the IAEA Nuclear Safety Series programme, now referred to collectively as the IAEA Safety Standards Series (SSS). Using an agreed working method, the relevant SSS safety principles, requirements, and guidelines were studied to see whether and how they would be applicable in The Netherlands. This procedure resulted in a series of amendments to the IAEA Codes and Safety Guides, which then became the draft NVRs. The amendments were formulated for various reasons: to introduce a choice from a range of different options, to give further guidance, to be more precise, to be more stringent, or to adapt the wording to specific Dutch circumstances (e.g. with respect to the risk of flooding, population density, seismic activity and local industrial practices).

These draft NVRs were reviewed by the regulatory body and, after a formal commenting procedure for the regulated licensees and advice of the Reactor Safety Commission, formally established under the responsibility of respective ministers (requirements) or directors-general (safety guides).

Further information can be found on ministry web site http://www.vrom.nl.

Romania

The Romanian legislative framework regulating the peaceful use of nuclear energy was subject to a continuous development since 1974.

Law No. 111/1996 on the Safe Deployment of Nuclear Activities entered into force on 26 December 1996 and was subsequently modified and completed by the Law no. 193/2003.

With regard to NPPs, the Law 111 applies to the activities of research, design, holding, siting, construction, installation, commissioning, operation, modification, preservation, decommissioning, import and export of nuclear installations, supply and procurement of products and services designated for nuclear installations.

The National Commission for Nuclear Activities Control (CNCAN) is a governmental organisation, which acts as the regulatory body for the safety of all nuclear activities in Romania and is therefore responsible for issuing licences.

The following regulations concerning NPPs have been used for the purpose of this study:

- Nuclear Safety Norms - Nuclear Reactors and Nuclear Power Plants (1975), which contain provisions concerning licensing basis documentation, site evaluation criteria and design criteria for NPPs.
- Norms for prevention and extinction of fires, applicable in the nuclear activities (1976);
- Nuclear Safety Norms on Planning, Preparedness and Intervention in Nuclear Accidents and Radiological Emergencies (1993);
- Norms on issuing of practice permits for operating, management and specific training personnel of nuclear power plants, research reactors and other nuclear installations (2004), which contain provisions regarding the training and licensing of NPP personnel.
- The set of Norms on Quality Management Systems for nuclear installations (2003) which contain provisions related to the quality assurance and safety of operation, maintenance, in-service inspection, testing, modifications, training of personnel, procurement activities, etc.
- Technical Prescriptions for Design, Execution, Assembling, Repair, Verifying, Operation of Pipes under Pressure and of Elements of Pipes from Nuclear Plants and Facilities (NC2-83) issued by the National Authority for Control & Approval of Boilers, Pressure Vessels and Hoisting Equipment.

A set of regulations which are envisaged to be published by the end of this year were also used in the benchmarking, as a "B" justification on the legal side:

- Norms on Fire Protection in Nuclear Power Plants;
- Norms for Containment Systems for CANDU Nuclear Power Plants;
- Norms for Shutdown Systems for CANDU Nuclear Power Plants;
- Norms for Emergency Core Cooling Systems for CANDU Nuclear Power Plants;
- Norms regarding Modifications to Nuclear Power Plants;
- Norms regarding Probabilistic Safety Assessment for nuclear power plants;
- Norms regarding Periodic Safety Review for nuclear power plants.

The licensing conditions and regulatory decisions are also legally binding. Compliance is also mandatory with Licensee's documents formally approved by CNCAN.

More information can be found on the web site www.cncan.ro

Slovakia

Pursuant to Atomic Act, the supervision of peaceful use of nuclear safety is performed by Nuclear Regulatory Authority (UJD) within its competencies. UJD is a central state administration body ensuring the performance of state regulatory activities in the field of nuclear safety of nuclear installations, including supervision of the management of radioactive waste, spent fuel and other fuel cycle phases, as well as of nuclear materials, including their control and records.

Concerning the nuclear safety, the basic legal framework is laid down by completely new Act No. 541/2004 Coll. on Peaceful Use of Nuclear Energy (Atomic Act). Since 1st December 2004, this new Atomic Act has abrogated former Atomic Act No. 130/1998 Coll. as well as all of 13 regulations issued on the former Atomic Act No. 130/1998 Coll. basis. A new set of regulations that work out the new Atomic Act provisions in detail was accepted and approved by the Slovak Government Legislative Council in August 2005.

There are regulation on special materials and equipments, on small quantities of nuclear materials, on details of the notification of events, on periodical safety assessment, on nuclear safety requirements, on the provision for physical protection, on professional qualification, on management of nuclear material, radioactive waste and spent fuel, on safeguards, on emergency planning, on shipment of radioactive materials, on requirements for quality system documentation, as well as details concerning quality requirements for nuclear installations, details concerning quality requirements for classified equipment and on documentation needed for certain decisions

The Atomic Act regulates rights and obligations of natural and legal persons in peaceful use of the nuclear energy, nuclear material, radioactive waste, physical protection, shipment of nuclear material, radioactive waste and spent fuel, licensing procedure of the nuclear installations, nuclear safety, emergency planning, quality assurance system, staff training, civil liability for nuclear damage, shut-down of a nuclear installation for other than safety concerns, inspections, sanctions. However, radiation protection is not within the scope of this Atomic Act but remains within the competencies of Public Health Authority subordinated to the Ministry of Health as stated in Act No. 272/1994 Coll. Besides acts and regulations as legally binding, the UJD also formally issues Safety Guides, which contains methods suggested by the UJD to address special topics related to nuclear safety. Safety Guides composes of non-binding provisions but they may be important as criteria within the licensing procedure.

The licensing procedure consists of three major stages: siting, construction commencement, and permanent operation. Before granting a licence for permanent operation, the regulatory authority carries out control under the approved programs for hot and cold testing and grants approval for fuel loading, physical start up, energy start up and trial operation. The basic condition essential to licensing in terms of nuclear safety is to prepare and submit a Safety Analysis Report and other prescribed safety documentation and to meet the conditions of the regulatory authority's preceding licensing procedures and decisions. Under the nuclear installation licensing procedure, International Atomic Energy Agency standards and recommendations are used and applied.

More information on the Slovak legislative and regulatory system can be found on the UJD web site: www.ujd.gov.sk.

Slovenia

The present Slovenian legislative and regulatory framework governing nuclear and radiation safety has a long-standing history which has its roots in former Yugoslav legislation. While at the beginning the legislation has focused mostly on the ionising radiation safety (act of 1965 and of 1976) in the 80's it incorporated also all basic provisions related to nuclear safety (act of 1984 and more than 10 regulations, class E and Z).

In July 2002, the Parliament adopted a new Act on Ionising Radiation Protection and Nuclear Safety (hereinafter referred to as a "2002 Act"). The 2002 Act provides that the regulations which have been issued on the basis of previous acts shall apply until new regulations and decrees, which are to be adopted pursuant to provisions of 2002 Act, are issued.

Based on the 2002 Act 4 governmental decrees and 14 ministerial regulations were adopted and issued until mid 2005. All these regulations are legally binding.

Besides the main principles the 2002 Act includes with respect to nuclear safety also provisions on:

- Classification of facilities (nuclear, radiation and less important radiation facilities);
- Licensing procedures (siting, construction, trial operation, operation, decommissioning);
- Radiation contamination and intervention measures;
- RADWASTE and SF management;
- Physical protection, non-proliferation and safeguards;
- Inspection and enforcement.

It includes provision on competent regulatory body. In nuclear and radiation safety the competencies are divided among two regulatory bodies, namely the Slovenian Nuclear Safety Administration (SNSA) which is accountable for nuclear safety and safety of industrial radiation sources and Slovenian Radiation Protection Administration (SRPA), accountable for radiation protection of patients, medical surveillance of exposed workers, surveillance of workplaces, dosimetry and dose registers and education in the area of radiation protection.

In the licensing process, the key document governing the technical and safety measures for the construction and operation of the nuclear facility is the Safety Analysis Report (SAR).

Further information on Regulatory body and legislative framework can be found on web site: http://www.gov.si/ursjv/en/index.php

Spain

The Spanish legal system relating nuclear energy was implemented through the development in 1964 of the Nuclear Energy Act (Law 25/1964) as amended, the Law establishing the Nuclear Safety Council (Law 15/1980) and Electricity Industry Law (Law 54/1997). This set of laws defines the safety principles or criteria, details the procedures to be applied for the necessary authorisations, and the mechanism for inspections and evaluations. Basic principles determine that the responsibilities derived from the usage of nuclear energy remain in the licensee of the installation. The Nuclear Safety Council (CSN) is the sole competent Authority for Nuclear Safety and Radiation Protection, independent from the Government and in charge of performing inspections and assessment of nuclear and radioactive installations. The Electricity Industry law introduces a new legal framework for faults and penalties, modifies the coverage required for civil liability, and assigns to the CSN a stronger role in the procedure of penalties.

The Government issued additional decrees to complete and clarify requirements established by law. The following decrees are the most significant regulations:

- Royal Decree 1836/1999 Regulation on Nuclear and Radioactive Installations (1999 revision) Defines the licensing system for sitting, construction, commissioning, operation and decommissioning.
- Royal Decree 783/2001 Regulation on protection of public and workers against the risks of ionising radiations (revision 2001). Includes the basic criteria and measures for radiation protection, as established in the Directive 96/29 issued by the EURATOM board.
- Decree governing the coverage of nuclear risks (1967). This one develops the Nuclear Energy Act in the field of the responsibility of the licensee, establishing the system for coverage for civil liability derived from such responsibility.
- Royal Decree 413/1997 governing the occupational protection of outside workers potentially exposed to ionising radiation due to their intervention in the controlled zone (1997). This regulation transposes the contents of EURATOM Directive 90/641.
- Royal Decree 1546/2004 approving Basic Nuclear Emergency Plan (2004 revision). This one defines the co-ordinated action of the different Public Organisations in case of a nuclear accident. It defines the emergency plans for each province in which is the site of a nuclear installation.

Based on the previous laws and regulations, and following the regulation of the country of the original plant design when applicable, an operation authorisation (license) is issued in the form of a Ministerial Order. This license covers a period of 10 years and includes the appropriate limits and conditions under which the operation of the plant must be conducted. This limits and conditions related to nuclear safety and radiological protection are legally binding. Other licensing documents (like SAR, Tec. Spec., Operations requirements, Doses calculation Manual, Emergency Plan, etc) also referred to in Royal Decree 1836/1999 and stated in each license are legally binding documents for each licensee. However since those documents are not public, they were not credited as national requirements according to the criteria of this study.

In addition, the CSN has the legal power to issue Instructions (with the same legal status than governmental regulations). The CSN issues Safety Guides, which contain methods, suggested by the CSN to address special topics related to nuclear safety and radiation protection. The Safety Guides are currently classified in sections covering the main areas of competence of CSN.

Further information can be found on web site: http://www.csn.es.

Sweden

Legally binding generic regulatory documents are Acts (laws), Ordinances, and Regulations. With respect to reactor safety, there are the Act on Nuclear Activities (1984:3 with later amendments), the Ordinance on Nuclear Activities (1984:14 also with later amendments), and regulations issued by SKI in the SKIFS series. SKI is mandated by the Ordinance on Nuclear Activities to issue such regulations that are allowed according to the Act. The Act (1984:3) contains basic provisions for safety in connection with nuclear activities and applies to the construction, operation and decommissioning of nuclear facilities as well as other handling of nuclear material and nuclear waste. It also contains the obligations to obtain a licence and the obligations connected with the holding of a licence. In addition, the Act contains provisions about public insight into the safety- and radiation protection work of the licensee and legal sanctions in cases of non -compliance with the regulations or the decisions of the regulatory body. Radiation protection as such is covered by another law, the Radiation Protection Act (1988:220). General obligations in cases of accidents which can threaten life and the environment are included in the Act (2003:778) on Protection against Accidents and The Ordinance (2003:789) on Protection against Accidents.

The following SKI Regulations and General Recommendations are referred to in the reactor harmonisation study. The General Recommendations on how to interpret the regulations have been issued in direct connection to the regulations and are included in the respective SKIFS publication. The licensees have to follow there recommendations or take other measures which are justified to be equal from the safety point of view.

- **Regulations and General Recommendations concerning Safety in Nuclear Facilities (SKIFS 2004:1)**: Basic requirements on design, safety management, physical protection, emergency preparedness, assessment and reporting of safety, operations and maintenance, management of nuclear materials and waste, and decommissioning.
- **Regulations and General Recommendations concerning the competence of Operations Personnel at Reactor Facilities (SKIFS 2000:1):** Requirements on competence analysis, training, and authorisation as well as requirements on simulators for operational training.
- **Regulations and General Recommendations concerning Mechanical Components in certain Nuclear Facilities (SKIFS 2000:2, revised as SKIFS 2005:2):** Requirements on measures, control- and inspection activities on mechanical components to be taken during plant modifications, maintenance, and in-service inspections.
- **Regulations and General Recommendations concerning Design and Construction of Nuclear Power Reactors (SKIFS 2004:2):** Requirements on design principles, withstanding of failures, conditions and events, and requirements on the design and operation of the reactor core.

Criteria and requirements concerning severe accident management were decided by the Government in 1986. Most of these requirements are now covered by SKIFS 2004:1 and 2.

The SSI regulations on Emergency Preparedness at certain Nuclear Facilities from the radiation protection point of view (SSI FS 2005:2) are also referred to in the study.

The Swedish Nuclear Power Inspectorate (SKI) is the regulatory body for reactor safety, nuclear materials safety, nuclear non-proliferation and nuclear waste safety.

The Swedish Radiation Protection Authority (SSI) is the regulatory body for radiation protection and emergency preparedness against radiation accidents.

More information can be found on the web sites: www.ski.se and www.ssi.se

Switzerland

With respect to nuclear installations, the legislative and regulatory framework in Switzerland consists of the federal constitution, federal laws or acts, federal ordinances (containing more detailed interpretations of the federal laws), and regulatory guidelines. The latter are issued by the Swiss Federal Nuclear Safety Inspectorate (HSK), an organisation that currently is part of the Federal Office of Energy (Bundesamt für Energie, BFE) and legally established as the competent authority for supervising nuclear installations in Switzerland at all stages of their lifetime. The Swiss regulatory body is composed of the HSK as supervisory authority for nuclear safety and the Section for Nuclear Energy (again part of BFE) as supervisory authority for nuclear security and safeguards. By law, prime responsibility for nuclear installations rests with the holder of the licence for a NPP.

Licences (for siting, construction, operation, and decommissioning) are issued at federal level. Each licence contains licence conditions that are mandatory for the licence holder. Within the licence conditions, HSK issues permits / approvals e.g. for safety limit settings, plant safety system modifications or cycle start-up after refuelling outage.

Since February 1 2005 legally binding provisions for authorisation / regulation / supervision / inspection of nuclear installations in Switzerland have been newly established with the Federal Nuclear Energy Act (Kernenergiegesetz, KEG) and its associated Nuclear Energy Ordinance (Kernenergieverordnung, KEV), as well as the Radiological Protection Act of 1991 (Strahlenschutzgesetz, StSG) and the Radiological Protection Ordinance of 1994 (Strahlenschutzverordnung, StSV). In particular, KEG / KEV contain a range of requirements, which relate to an important part of the reference level requirements. In addition, HSK regulatory guidelines contain many detailed requirements which may be considered legally based; most of these Guidelines are explicitly referenced in the new KEV. The status of legal requirements as identified in the reference levels of each safety issue was assessed on this basis.

The process of implementation of the new legislation (KEG/KEV), which entails the issuing of additional ordinances and regulatory guidelines as well as rewriting existing ones, is ongoing; addressing the harmonization issues identified in this report is intended to be part of this process.

More detailed information about HSK and its mandate may be found at www.hsk.ch. This web site includes a link to the 3rd Swiss report on the implementation of the obligations of the international "Convention on Nuclear Safety", which provides full details on the legislative and regulatory framework, and a link to all HSK Regulatory Guidelines that are currently in force.

United Kingdom

The operators of nuclear plants in the UK must, like their counterparts in other industries, conform to the Health and Safety at Work Act 1974 (HSW Act). The HSW Act is goal setting in nature and places a fundamental duty on employers to ensure, so far as is reasonably practicable, the health, safety, and welfare at work of all their employees. It also imposes a duty to ensure that members of the public are not exposed to risks to their health or safety because of the activities undertaken. The Health and Safety Executive (HSE), which is the parent body for the Nuclear Installations Inspectorate (NII), enforces the HSW Act.

The Nuclear Installations Act 1965 (as amended) (NI Act) augments the HSW Act, preventing nuclear plants being installed or operated on a site until the HSE has granted a nuclear site licence to a corporate body. A licence is not transferable but a new licence may be granted to another corporate body, subject to the same evaluation process as for an initial licence.

Each licence contains a standard set of 36 non-prescriptive licence conditions for all plants to provide consistent safety requirements. They are phrased in general terms that make the licensee responsible for developing and applying detailed safety standards and procedures for the plant. Thus, each licensee can adopt arrangements that best suit their business, so long as safety is being properly managed. When considering a licence application, HSE scrutinises the suitability of the proposed organisation and location together with the hazards and risks associated with the proposed activities.

The licensee is responsible for the safety of their plant and must provide NII with a written demonstration of safety. This is known as the ‘safety case’: this covers all stages in the life of the plant from construction through to decommissioning and must be updated to reflect changing conditions. Under the NI Act, all significant safety-related activities need some form of permission from NII. This 'permissioning regime’ prevents licensees from substantially modifying plant or altering operating arrangements without NII involvement. Assessment is the process by which the NII, on behalf of HSE, establishes whether the safety case is adequate and the Safety Assessment Principles are used for that purpose. These principles are published in a public document. NII also has other documents, such as Technical Assessment Guides (TAGs), Technical Inspections Guides (TIGs), and other specific guidance, that have been published or are being added progressively to its web site that inform licensees and the public about how NII assesses licensees’ proposals and the requirements that need to be met for permission to be granted.

NII exercises control through a number of legal instruments under powers derived from the licence conditions and NII inspectors may also use their enforcement powers under the HSW Act to issue Prohibition and Improvement Notices and to prosecute for breaches of that Act. Breaches of licence conditions are offences under the HSW Act.

More information can be found on NII’s web site:

http://www.hse.gov.uk/nuclear/index.htm

Annex 4 – Participating organizations

Country	Organization	Address	Participants
Belgium	Association Vinçotte Nuclear **(AVN)**	148 Walcourtstraat, B-1070 Bruxelles, BELGIUM	**Benoit De Boeck, Pieter De Gelder**
Bulgaria	Nuclear Regulatory Agency **(NRA)**	69 Shipcheski Prokhod Blvd., 1574 Sofia, BULGARIA	**Elisabeth Tsvetanova, Ventzislav Miliovsky**
Czech Republic	State Office for Nuclear Safety **(SONS)**	SUJB, Senovazne namesti 9, 182 00 Praha 1, CZECH REPUBLIC	**Jaromir Sipek, Zdenek Tipek**
Finland	Radiation and Nuclear Safety Authority **(STUK)**	Laippatie 4, P.O. Box 14, FIN-00881 Helsinki, FINLAND	**Pekka Salminen, Hannu Ollikkala, Kirsi Alm-Lytz , Ilari Aro, Pentti Koutaniemi**
France	Autorite de Surete Nucleaire **(ASN)**	10 route du Panorama Robert Schumann – BP 93 – 92266 Fontenay-aux-Roses Cedex, FRANCE	**Olivier Gupta, Etienne Kalalo, Thomas Maurin**
Germany	Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit **(BMU)**	Referat RS I 5, Multilaterale regulatorische Zusammenarbeit, Robert-Schuman-Platz 3, D 53175 Bonn, GERMANY	**Michael Herttrich, Matthias Gohl**
	Gesellschaft für Anlagen- und Reaktorsicherheit mbH **(GRS)**	Schwertnergasse 1, 50667 Köln, GERMANY	**Manfred Simon, Gerald Diepolder**
Hungary	Hungarian Atomic Energy Authority **(HAEA)**	OAH, 1036 Budapest, Fényes Adolf utca 4, HUNGARY H-1539 Budapest PO Box 676, HUNGARY	**Lajos Voross, Andras Toth**
Italy	Agency for Environmental Protection and for Technical Services **(APAT)**	Via Vitaliano Brancati 48, 00144 Roma, ITALY	**Giovanni Bava**
Lithuania	State Nuclear Power Safety Inspectorate **(VATESI)**	Gostauto 12, LT-01108, Vilnius, LITHUANIA	**Saulius Svirmickas, Sigitas Slepavicius**
The Netherlands	Kernfysische Dienst **(KFD)**	VROM /VI/KFD, IPC 560 Postbus 16191 2500 BD Den Haag NETHERLANDS	**Piet de Munk**
Romania	National Commission for Nuclear Activities Control **(CNCAN)**	14 Libertatii Blvd, Bucharest 5, ROMANIA	**Lucian Biro, Madalina Tronea, Florian Baciu**
Slovakia	Nuclear Regulatory Authority **(NRA)**	UJD, Okruzna 5, 918 64 Trnava, SLOVAKIA	**Peter Uhrik, Pavel Bobaly, Stefan Cepcek, Jan Husarcek**
Slovenia	Slovenian Nuclear Safety Administration **(SNSA)**	Zelezna cesta 16, P.O. Box 5759, SI 1001 Ljubljana, SLOVENIA	**Djordje Vojnovic, Artur Mühleisen**
Spain	Consejo de Seguridad Nuclear **(CSN)**	Pedro Justo Dorado Dellmans, 11. 28040 Madrid SPAIN	**Ivan Recarte, Maria Moracho**
Sweden	Swedish Nuclear Power Inspectorate **(SKI)**	Klarabergsviadukten 90, S-106 58 Stockholm SWEDEN	**Erik Jende, Lars Bennemo, Lars Gunsell**
Switzerland	Swiss Federal Nuclear Safety Inspectorate **(HSK)**	P. O. Box 5232 Villigen – HSK, SWITZERLAND	**Willem van Doesburg**
UK	Nuclear Safety Directorate **(NSD)**	Redgrave Court, Merton Road, Bootle, Merseyside, L20 7HS UNITED KINGDOM	**Paul Woodhouse, Mike Robbins, Gill Haydon**

13/01/06

Western European
WENRA
Nuclear Regulators Association

WENRA

Pilot Study on Harmonisation of Reactor Safety in WENRA Countries

March 2003

Western European Nuclear Regulators' Association

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Abstract

WENRA Working Group on Reactor Harmonisation

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1. Background

In November 1999, WENRA decided to set up a Working Group in charge of investigating how to proceed towards a harmonised view on reactor safety in EU countries with nuclear programmes.

The Working Group should address principal differences and similarities in the substance of safety requirements in the areas of deterministic and probabilistic requirements as well as in the area of safety management and safety culture. The Group should not go too deeply into legal and technical details.

In March 2000, an outline of a Pilot Study was presented and approved by WENRA. The objective of the Pilot Study was to develop and test a methodology for systematic comparison of national requirements on selected safety issues. The results should make conclusions possible about specific needs for harmonisation of national requirements connected with these issues, in order to reach a comparable safety level. The detailed means necessary for each country to reach the harmonised level was not to be suggested, but the conclusions should be sufficiently clear about what needs to be further addressed on the national level.

A first report, with a preliminary study of six safety issues, was issued in February 2001. WENRA. endorsed the proposed methodology and asked the Working Group to continue its work along the suggested lines in order to make the study of the six safety issues more complete[1]. The Group was also asked to compare the reference levels used in the study with the most recent IAEA safety standards, and on that basis amend the reference levels if justified. In addition the Group was asked to propose how the study could be continued with further safety issues to be decided later.

2. What is harmonisation?

In order to design the methodology, an operational definition of harmonisation was necessary. For this purpose, the Working Group defined harmonisation as: *no substantial differences between countries from the safety point of view in generic formally issued national safety requirements, and in the resulting implementation on the Nuclear Power Plants.*

This definition has several implications for harmonisation work aiming at equal safety levels in the different countries:

both the legal and the implementation (technical) aspects need to be considered,
requirements need to be formally issued on a legal basis,
requirements need to be public and transparent,
requirements need to be generic (apply equally on the licensees),
harmonised requirements need to be interpreted and implemented in an equal way,
harmonised requirements need to be enforced in an equal way.

[1] Switzerland did not participate in the Pilot Study. The Netherlands did not participate in the second phase of the study.

3. Methodology

The Working Group set some conditions for the design of the methodology. The national requirements to be included in the study needed to be selected and described in a consistent way. Differences in the principal design of the legal systems and in regulatory practices should be disregarded as much as possible, in order to focus on the substantial differences and similarities of comparable requirements. It should be possible to identify and describe whether national requirements are applied in a consistent way or if, for instance, older reactors are exempted from certain requirements. As mentioned, not only the legal aspects of harmonisation but also the implementation should be considered, i.e. it should be possible to identify those cases where existing requirements are not implemented and cases where an implementation exist but not formally required. It should be possible to get a convenient overview of differences and similarities between the countries without losing too much information. The conclusions about differences should be based on a safety justification and should be detailed enough to provide input to a further more detailed analysis on the national level. The methodology should be transparent, i.e. it should be possible to understand on what basis the conclusions were drawn.

Following the conditions, the methodology was designed in eight working steps as follows:

1. Select significant safety issues to be included in the project.

2. Identify, for each country and safety issue the relevant legal documents.

3. Describe in a standardised way (description matrix) the substance of each country's national requirements and status of implementation.

4. On the basis of the national requirements, establish reference levels on each issue, reflecting the best national practices (the study uses the terminology "highest quartile" of existing requirements).

5. Compare with the most recent IAEA safety standards, amend the reference levels if justified, make a comment on the relations between the finally agreed reference levels and the IAEA safety standards.

6. Assess (in a panel with all countries represented) and document in a systematic way (comparison matrix) to what extent the reference levels exist and are implemented in each country.

7. Conclude from the panel assessments whether there are any substantial differences or not between the reference levels and the respective national practices, conclude about which differences can be justified from the safety point of view and which differences should be further addressed for harmonisation, provide justifications and explanations.

8. Summarise the results of the analysis.

In the following these steps are commented briefly.

3.1. Selection of significant safety issues

Since the scope of requirements on reactor safety is very large and requirements exist on several levels of detail, every harmonisation project needs for practical reasons to make an inventory and a selection of issues to deal with. This selection can be done according to different principles. One principle is to make a safety map by defining a structure of issues, with the ambition to cover the whole field of reactor safety. Another consistent approach is to identify the safety issues following from the Convention on Nuclear Safety. A further selection principle is to define and select harmonisation issues taking into account known differences between the countries as outlined in several reports from IAEA, the EC advisory groups and the committees and working groups within OECD/NEA. Another principle is to focus on common regulatory challenges for the near future and to identify the safety issues and requirements associated with these challenges.
It is also possible to group and select safety issues on the basis of safety factors to be included in a Periodic Safety Review[2].

There are also a number of pragmatic approaches that can be used in the selection of issues. One is to look through the IAEA Safety Requirements[3] or published requirements from different countries[4] to check which issues are addressed in those documents. Obviously issues included in these documents are considered to be important to safety, although the basis for this is not always clear.

The Working Group followed a pragmatic approach in its selection of issues. It was clear that only a few issues could be analysed within the frame of the Pilot Study. The selection was done primarily to find issues, which would provide a good test of the methodology. For this purpose the following selection criteria were used:

- the issues should represent different safety areas as identified in the IAEA Safety Requirements,
- some issues should at face value be more difficult and some more easy to analyse,
- there should be differences between the national requirements related to the issues.

The following issues were finally selected:

Safety area	Issues
Safety Management	- Safety Policy - Operating Organisation
Design	- Verification and Improvement of the Design
Operation	- Beyond Design Basis Accident Management
Safety verification	- Probabilistic Safety Analysis - Periodic Safety Review

3.2 Identification of national requirements

The Working Group used the following definition of a national requirement: a legally binding generic safety requirement currently in force, or a formally issued general recommendation. These requirements are of two types, both legally based but with different legal powers.

2 See IAEA Draft Safety Guide DS 307
3 For instance NS-R-1 Design and NS-R-2 Operation
4 For instance the Swedish SKIFS or the Finnish YVL-Guides.

A legally binding requirement, such as a law, ordinance or regulation, is mandatory and enforced, if necessary with use of legal sanctions. These requirements are issued by the parliament, the government, or the regulatory body on behalf of the government. All WENRA countries have an established procedure for issuing such requirements, including, to a varied degree, public hearings or other mechanisms for soliciting the opinion of the stakeholders. In some cases a cost/benefit analysis must also be made. In most countries, the legally binding requirements are expressed as "shall" statements.

A formally issued general recommendation is a rule or guideline that the regulatory body is authorised to issue with reference to a legally binding document or other formal authorisation. These recommendations are not legally binding and enforced like regulations, however they can not be ignored by the licensee without risking some sanction by the regulatory body. General recommendations are most often, in a similar manner as legally binding requirements, issued according to a formal procedure including soliciting the opinion of the stakeholders. The licensee has a choice to implement the specific recommendation or an alternative, justified to be equal from the safety point of view. In most countries these recommendations are expressed as "should" statements.

Both the legally binding requirements and formally issued general recommendations are published in official documents, which depending on national practice can be requested by the public and media. In many cases they are published in official publications and posted on the internet web site of the government and/or the regulatory body. From the harmonisation point of view, it is important that the requirements are open to the public and transparently enforced in order to ensure equal conditions for the licensees in the different countries. Requirements defined and enforced by specific licensing documents only, such as SARs and Operational Manuals have limitations here, since these are non-public documents.

The study deals only with generic requirements. This means that specific regulatory decisions, which are legally binding and documented, but not addressing all licensees in the same manner are excluded.

The legally binding generic requirements and formally issued general recommendations were identified from the document hierarchy of each country. This was in itself not an easy exercise, since the picture is more or less complicated. It is however clear that regardless of legal system and regulatory practice, the two types of requirements exist in all WENRA countries.

3.3 Description of national requirements

The identified national requirements related to each issue were described in a condensed way in the matrix format below, i.e. only the substance of the requirements was described and not the full text of the original document. The level of detail regarding the substance was however the same as in the original document. References to the original documents were given.

It was indicated in the matrix if there were any legal exemptions from the mentioned requirements or possibilities for exemptions, for instance if older reactors are excepted or if a certain reactor design is excepted. It was also indicated if compensatory measures were allowed, i.e. if a certain requirement could be considered fulfilled by complying with another requirement. Finally the implementation status as known by the regulatory body was indicated. In cases where measures related to the issue were judged as implemented but not formally required, a comment was made about this.

Description matrix
Substance of national requirements		Safety issue:		Country:

Ref req theme	Legally binding requirement	General recommendation	Comments[1]
1			
2			
3			
N			

1/I = implemented in all NPPs to an acceptable extent	I- = not sufficiently implemented in all NPPs
I+ = implemented above requirements in all NPPs	I? = implementation status not known
E = permanent exemptions allowed (NE= not allowed)	C = compensatory measures allowed (NC= not allowed)

The matrix was further structured to make it possible to identify which national requirements are related to the themes of the reference levels (see section 3.4). This would make it easier to understand the basis for the comparison assessments. However, in some cases the national requirements on the safety issue under examination were different from the reference levels to such an extent that it was not possible to directly relate them. In those cases the matrix has no references to the themes of the reference levels. One matrix was developed per issue (6) and country (8).

3.4 Establishment of reference levels

In order to compare in a systematic manner, and conclude on the substantial differences and similarities between national requirements, it was necessary to define a reference against which to make the comparisons. Without references it seemed impossible to draw systematic conclusions based on technical criteria. The references were elaborated as reference levels related to each selected safety issue. These levels were chosen from the national requirements already existing in the WENRA countries. Somewhere in these countries the reference levels should be in force and probably also implemented on the nuclear power plants. This also implies that the reference levels most probably had been scrutinised and reviewed by the stakeholders according to a formal procedure, and found to be realistic and useful for safety. Hence, the reference levels have a real background and are not abstractions.

The Working Group reviewed the legally binding requirements and recommendations existing in the different countries, related to each issue, and selected from these the most safety significant (key) requirements in the opinion of the Group. It was important that the result of this exercise was not a compromise between already existing national requirements, but an informed decision on a reasonable level for the near future based on current regulatory experience. The expected outcome was, in the terminology of the study, in the "highest quartile" of existing national requirements. This means that the reference levels do not necessarily reflect the most advanced requirement currently existing in at least one of the countries, but is selected among the most advanced existing requirements, where such a grading is possible. In the end this is an expert judgement where the Working Group applied its rather long experience as regulators to select reference levels, which were considered to be significant, realistic, cost-efficient and which have proved to be useful for safety.

The substance of the reference levels should reflect as much as possible the national requirements used as a model. This means that the reference levels should correspond as much as possible in strictness, prescriptiveness and level of detail to the national requirements used as model. In many cases this means that the reference levels are non-prescriptive and do not regulate in detail. In other cases the reference requirements are more detailed.

The Working Group tried to find a practical and neutral model for the structure and wording of the reference levels. For each safety issue, the reference levels were grouped under themes in order to provide a logical structure and an easier overview. The reference levels themselves are formulated in "shall" sentences. The number of reference levels under each theme depends on the complexity of the theme and varies between one and nine.

The themes are shown in the table below. The complete set of reference levels is presented in the enclosed annex.

Safety area	Issue	Reference level themes
Safety Management	- Safety Policy	1. Issuing and communication of a safety policy 2. Strategy for implementing the safety policy and monitoring safety performance 3. Evaluation of the safety policy
	- Operating Organisation	1. Organisational structure 2. Management of safety and quality 3. Sufficiency and competency of staff
Design	- Verification and Improvement of the Design	1. Selection of design basis events and hazards 2. Demonstration of reasonable conservatism and safety margins of the design basis 3. Definition and application of technical acceptance criteria 4. Extension of the design 5. Instrumentation and hardware provisions for the management of severe accident conditions 6. Improvement of the design
Operation	- Beyond Design Basis Accident Management	1. Procedures and guidelines for dealing with beyond design basis accidents 2. Training and exercises for accidents beyond design
Safety verification	- Probabilistic Safety Analysis	1. Scope and content of PSA 2. Quality of PSA 3. Use of PSA
	- Periodic Safety Review	1. Objective of the periodic safety review 2. Scope of the periodic safety review 3. Methodology of the periodic safety review

Areas, issues and reference level themes selected for the Pilot Study

3.5 *Comparison with IAEA safety standards*

For this comparison, the following procedure was used:

2. The relevant IAEA documents were identified from the categories Safety Requirements and Safety Guides of the IAEA Safety Standards Series. These two categories correspond in the IAEA system to the definition of national requirements used in the Pilot Study.

   For the purpose of the Pilot Study, the Working Group used the most current version of the relevant IAEA standards, even if some documents still were under revision and not yet formally issued. Two of the documents were issued during the study.

   The following documents were found relevant for the purpose of the Pilot Study:

   Safety Requirements

   Safety of Nuclear Power Plants: Design. NS-R-1, IAEA Vienna, 2000.
   Safety of Nuclear Power Plants: Operation. NS-R-2, IAEA Vienna, 2000.

   Safety Guides

   The Operating Organisation for Nuclear Power Plants. NS-G-2.4. IAEA Vienna, 2001.
   Safety Assessment and Verification for Nuclear Power Plants. NS-G-1.2. IAEA Vienna, 2001.
   Periodic Safety Review of Nuclear Power Plants. DS 307, Draft 8. IAEA Vienna, 2001-12-07[5].

3. The IAEA documents above were screened in order to find the corresponding IAEA requirements related to each reference level. These IAEA requirements were documented on the reference levels paper under the headline "Related IAEA safety standards".

   As a first step the IAEA Safety Requirements were screened (shall statements). As a second step the corresponding Safety Guide was screened in order to check for any additional relevant requirements (should statements).

4. An analysis was made about the correspondence between each reference level and the related IAEA standard. An amendment was made of the reference level in the following cases:

   The IAEA standard addressed additional aspects which were considered useful for safety reasonable and practicable by the Working Group,
   The IAEA standard had a more clear and functional wording.

5. After amendments, an overall conclusion was made and a comment about the correspondence between the final set of reference levels on each issue and the related IAEA safety standards, regarding scope and strictness.

The IAEA standards provide an internationally agreed framework for the safety of nuclear installations. In many cases they are rather extensive descriptive documents, each dealing with or making cross-references to several safety issues. It is not easy to directly compare these

[5] Endorsed by CSS for publication in June 2002.

documents with the more explicit national requirements in force in WENRA countries, on which the reference levels were modelled. However the substance can be compared.

Only a few amendments were made to the original reference levels, as a result of the comparison with the IAEA standards, mostly regarding the issues Verification and improvement of the Design and Periodic Safety Review. In those cases the IAEA standards had a more functional wording on some points.

As a general conclusion, there is a rather good correspondence in substance between the reference levels and the most recent IAEA standards[6]. In no cases were the IAEA requirements found to be stricter than the original reference levels. On the issues Operating Organisation, Verification and improvement of the Design, Beyond Design Basis Accident Management and PSA, several reference levels are stricter than the IAEA standards. For instance regarding measures to cope with accidents beyond the design basis, the reference levels include specified measures, some of which only mentioned in the IAEA Safety Guides as examples of Member State practice.

3.6 Assessment of differences between national practices and reference levels

A consistent approach is needed for making these assessments with sufficient reliability. From the first test of the methodology it was concluded that several provisions were needed in order to reduce subjectivity and increase transparency of these assessments. Several modifications of the methodology were implemented during the study to provide for this:

- The descriptions of national requirements were restructured so it is easier to identify how they compare with the reference levels,
- A panel was created for the comparison assessments consisting of representatives from all participating countries,
- A documented procedure was developed for the assessments,
- Documented criteria were used in the assessments,
- The documentation format (comparison matrix) was simplified.

The comparison assessment was documented in a matrix consisting of three sheets. The matrix shows the situation in each country in terms of which reference levels exist in national requirements and are implemented. Sheet 1 was designed for documentation of the assessment results and the general conclusions. Sheet 2 was intended for explaining differences, which could be justified and sheet 3 was used to explain differences, which could not be justified and hence should be addressed for harmonisation. One comparison matrix (3 sheets) was developed for each safety issue under examination.

Each reference level was allocated to one of two categories in the comparison matrix addressing the legal aspect and one of two categories addressing the implementation:

Rather strict criteria were applied in these assessments. To qualify as a national requirement, the requirement had strictly to satisfy “formally issued, generic and legally based” as explained in section 3.2. Required means that the reference level was judged to be equal in substance to a national requirement. Minor differences in wording and contextual differences might exist, such

[6] The Working Group identified several cases where the wording of the IAEA documents could be directly related to input given from WENRA countries during the production of the documents.

that the reference level can be included in other national requirements, rather than being self-standing, but must in such cases be possible to identify. No difference in substance means that the reference level and the national requirement will lead to equal implementation measures at the NPP level, i.e. are judged in the end to result in equal safety margins of the barriers and defence in depth system.

Comparison matrix- Issue: Sheet 1(3)

Assess-ment	Belgium	Finland	France	Germany	Italy	Spain	Sweden	UK
Required								
Not Required								
Imple-mented								
Not Imple-mented								
Conclu-sion X, Y								

Implemented means that the reference level was judged to having been implemented in all NPPs of the country.

An important modification of the methodology during the Pilot Study was the establishment of a panel for making the assessments. The panel consisted of the entire Working Group with one representative for each participating country. In the panel session, each member explained and if necessary evidenced his prepared national position on the safety issues, which then was opened for questioning by the other members. The final classification on sheet 1 was decided collectively. In cases of disagreement, the country representative was given time to adjust his position including consulting his home office.

This procedure improved the reliability and consistency of the assessments to a great extent. In many cases complex judgement had to be exercised and the procedure supported that kind of problem solving in a good way. Voting was not necessary.

3.7 Conclusion about differences between national practices and reference levels

A summary conclusion of the overall position of each country on the safety issue under examination was confirmed by the panel and documented in the bottom row of the comparison matrix sheet 1. This conclusion was drawn in two dimensions:

(X) Difference between national requirements and reference levels,
(Y) Difference between national implementation and implementation of reference levels.

Under each dimension the following was indicated:

A. In principle already harmonised,
B. Differences exist but can be justified from the safety point of view,
C. Differences exist which should be addressed for harmonisation.

To qualify as an A, all reference levels had to be graded as required or implemented, i.e. there were no substantial differences between the reference levels and the national requirements and no differences in the implementation. For a B, there were substantial differences, i.e. at least one reference level was not required or not implemented, but this difference was possible to justify from the safety point of view. Conclusion C indicated that there were one or more differences, which could not be justified.

Approved justifications were:

- Regulations are under development or revision and will include the missing reference level(s),
- The reference level is covered by a different national requirement to such an extent that the added safety value of the reference level is minor,
- Implementation of a reference level is lacking in an older plant for which a shut down decision has been taken,
- Implementation of a reference level is in progress and it is only a matter of time before completion,
- Implementation of a reference level is not reasonably practicable on a specific reactor design and has been exempted on the basis of a technical justification that has been accepted by the regulatory body.

The last justification applies only in a few specific cases where technical back-fitting would be unreasonable from the safety point of view.

Since substantial differences can be graded as B or C, there were cases where both applied (BC). In those cases, substantial differences were identified, some of those possible to justify and some not.

On sheet 2 of the matrix, the differences between national requirements and reference levels possible to justify (B-differences) were described and the justifications provided.

Comparison matrix- Issue: …… Sheet 2(3)

Justification of differences

Country	Difference	Justification
Belgium		
Finland		
France		
Germany		
Italy		
Spain		
Sweden		
UK		

On sheet 3, the differences not possible to justify (C-differences) were described.

Comparison matrix- Issue: ……… Sheet 3(3)

Differences which should be addressed for harmonisation

Country	Difference
Belgium	
Finland	
France	
Germany	
Italy	
Spain	
Sweden	
UK	

3.8 Summary of results

The main result of the study is that none of the WENRA countries that were involved in the Pilot Study totally complies with the reference levels as stated in the Annex. In many cases the differences between national requirements and reference levels have been justified according to the pre-established criteria. However, several differences which should be further addressed by the respective country for harmonisation (C-differences) have been identified. Hence, for those cases measures should be taken to bring the national situation up to the reference level.

In the opinion of the Working Group, it is also important when reviewing the results to look at the B-differences, i.e. the differences between national requirements and reference levels, which were possible to justify. Although these differences need not to be addressed for harmonisation, they should be reviewed anyhow. For instance it could be useful to check if the national requirements could be improved or clarified on these points. In cases where differences with regard to a certain reactor design have been justified, since harmonisation measures are not judged to be reasonably practicable, an extended analysis of possible compensatory measures is recommended in the framework of the next Periodic Safety Review.

4. Conclusions about the methodology

Most of the objectives, set for the Pilot Study, were met. It can be concluded that the methodology was adequate for its purpose. National requirements on selected safety issues have been systematically compared and the major gaps and differences have been identified. Convenient overviews have been provided of differences and similarities between the countries. Furthermore, the conclusions are based on a safety justification and are detailed enough to provide input to a further more detailed analysis on the national level. It was not possible, however, to provide fully verified conclusions about the implementation of the reference levels in the different countries. This has to do with the following constraints on the study.

In line with the Terms of Reference, the comparison of formal requirements did not address the more detailed use of criteria and methods to verify compliance. The same requirement could be enforced differently in different regulatory systems, and hence lead to different implementation. The Pilot Study also assessed the implementation, but it was not possible to do this in sufficient detail to identify such differences. The implementation was assessed on the basis of current knowledge of the respective regulatory body, but it was not possible to provide the panels with evidence of the implementation. For these reasons, conclusions about implemented safety provisions in the different countries should be drawn with precaution.

The introduction of the panel assessments greatly improved the quality and consistency of the comparison assessments. Uncertainties in the assessments are mainly connected with lack of time to make a detailed analysis in some cases. The reliability of the assessments seems to be sufficient for the objectives of the Pilot Study.

The introduction of the IAEA safety standards in the study proved to be helpful and provided confidence in the scope and strictness of the reference levels.

This Pilot Study has contributed to understand how to approach harmonisation and has provided a basis for further progress on this important safety and policy issue. The study has further provided a systematic opportunity to learning from best national and international practices in order to promote safety and has already contributed to improvements.

Annex

Reference levels developed and used in the Pilot Study

Safety area: Safety Management
Issue: Safety policy

1. Issuing and communication of a safety policy

Reference levels:

1.1 A written safety policy shall be issued by the licensee.

1.2 The safety policy shall be clear about giving safety first priority in all plant activities.

1.3 The safety policy shall include a commitment to continuously develop safety.

1.4 The safety policy shall be communicated to all staff[7], with tasks important to safety, in such a way that the policy is understood and applied.

1.5 The safety policy shall be communicated to subcontractors, in such a way that the policy is understood and applied in their on-site activities.

2. Strategy for implementing the safety policy and monitoring safety performance

Reference levels:

2.1 The safety policy shall require a strategy for implementing the safety policy and monitoring safety performance.

2.2 The safety policy shall require safety objectives and targets, clearly formulated in such a way that they can be easily monitored and followed up by the plant management.

3. Evaluation of the safety policy

Reference level:

3.1 The adequacy and the implementation status of the safety policy shall be evaluated by the licensee on a regular basis, more frequent than the periodic safety reviews.

[7] This is understood as the licenseés own staff

Safety area: Safety Management
Issue: Operating Organisation

1. Organisational structure

Reference levels:

1.1 The organisational structure for safe and reliable operation of the plant, and for ensuring an appropriate response in emergencies, shall be justified and documented.

1.2 The adequacy of the organisational structure, for its purposes according to 1.1, shall be assessed on a regular basis, more frequent than the periodic safety reviews.

1.3 Responsibilities, authorities and lines of communication shall be clearly defined and documented for all staff with duties important to safety.

1.4 Changes to the organisational structure which might be significant for safety shall be justified in advance, carefully planned and evaluated after implementation.

2. Management of safety and quality

Reference levels:

2.1 The licensee shall ensure that the plant is operated in a safe manner and in accordance with all applicable legal and regulatory requirements.

2.2 The licensee shall ensure that decisions on safety matters are preceded by appropriate investigation and consultation.

2.3 The licensee shall ensure that the staff is provided with the necessary resources and conditions to carry out work in a safe manner.

2.4 The licensee shall ensure that safety performance is continuously monitored through an appropriate review system in order to ensure that safety is maintained and improved as needed.

2.5 The licensee shall ensure that relevant operating experience, international development of safety standards and new knowledge gained through R&D-projects are systematically analysed and continuously used to improve plant activities.

2.6 The licensee shall ensure that plant activities (processes) are controlled through a documented quality management system covering all activities, including relevant activities of vendors and contractors, which may affect the safe operation of the plant.

2.7 The quality management system shall be regularly audited by independent auditors, and kept up-to-date.

2.8 Significant safety issues shall be subjected to appropriate safety review, by a suitably qualified independent review function, before being submitted to the regulatory body.

3. Sufficiency and competency of staff

Reference levels:

3.1 The required number of staff for safe operation, and their competence, shall be analysed in a systematic and documented way.

3.2 The sufficiency of staff for safe operation, their competence and suitability for safety work shall be verified on a regular basis and documented.

3.3 A long term staffing plan shall exist for activities which are important to safety.

3.4 Changes to the level of staffing which might be significant for safety shall be justified in advance, carefully planned and evaluated after implementation.

3.5 The licensee shall always have, in house, sufficient and competent staff and resources to understand the licensing basis of the plant (e.g. Safety Analysis Report or Safety Case and other documents based thereon), as well as to understand the actual design and operation of the plant in all plant states.

3.6 The licensee shall maintain, in house, sufficient and competent staff and resources to specify, set standards manage and evaluate safety work carried out by contractors.

Safety area: Design
Issue: Verification and Improvement of the Design

1. Selection of design basis events and hazards

Reference levels:

1.1 The current design basis shall be clearly and systematically defined and documented.

1.2 The design basis shall include a set of postulated initiating events, with consideration of failures and hazards (internal and external, natural and man-induced), selected with deterministic or probabilistic methods or a combination of both, to demonstrate that the necessary safety functions are accomplished and the safety objectives met.

2. Demonstration of reasonable conservatism and safety margins of the design basis

Reference levels:

2.1 The initial and boundary conditions shall be specified in a conservative way.

2.2 The single failure criterion shall be applied in all design basis analyses of postulated initiating events.

2.3 Non-safety systems, including off-site power, shall be assumed to operate only if they aggravate the effect of the initiating event.

2.4 The safety systems shall be assumed to operate at their minimum performance level.

2.5 Any failure, occurring as a consequence of a postulated initiating event, shall be included in the design basis analysis.

2.6 The impact of uncertainties, which are of importance for the results, shall be addressed in the design basis analyses.

3. Definition and application of technical acceptance criteria

Reference levels:

3.1 Radiological and other technical acceptance criteria shall be assigned to each plant condition (typically normal operation, anticipated operational occurrences, design basis accidents, additional failure assumptions, and severe accidents), according to its probability of occurrence.

3.2 Criteria for protection of the fuel cladding shall be specified, including fuel temperature, DNB, cladding temperature, fuel rod integrity and maximum allowable fuel damage during any design basis accident.

3.3 Criteria for the protection of the (primary) coolant pressure boundary shall be specified, including maximum pressure, maximum temperature, thermal- and pressure transients and stresses.

3.4 For PWR only: Criteria in 3.2 shall be specified as well for protection of the secondary coolant system.

3.5 Criteria shall be specified for protection of the containment, including temperatures, pressure and leak rates.

4. Extension of the design

Reference levels:

4.1 Consideration shall be given to the performance of the plant in specified accidents beyond the design basis, including a selection of severe accidents, to determine those sequences for which reasonable practicable preventive or mitigation measures can be identified (accident vulnerability study). For this study a combination of engineering judgement and probabilistic methods can be used and evaluations be made on a best estimate basis.

4.2 Consideration shall be given, in the same manner as in 4.1, to combination of postulated initiating events with internal and external hazards.

4.3 The specified accidents beyond the design basis shall include station blackout, ATWS, multiple SG tube rupture, loss of main heat sink, and loss of required safety systems in the long term after a postulated initiating event.

5. Instrumentation and hardware provisions for the management of severe accident conditions

Reference levels:

5.1 Adequate instrumentation shall exist which can survive severe accident environmental conditions in order to manage such accidents according to guidelines/procedures for severe accidents.

5.2 Necessary information from instruments shall be relayed to the control room and presented in such a way to enable a timely assessment of the plant status and critical safety functions in accident conditions.

5.3 Means shall exist for containment isolation in a severe accident, including bypass prevention[8].

5.4 The containment leaktightness shall be ensured for a reasonable time after a severe accident.

5.5 Means shall be provided to manage pressure and temperature in the containment during a severe accident

5.6 Means shall be provided to control combustible gases in a severe accident.

5.7 Means shall be provided for containment overpressure protection in a severe accident.

5.8 Means shall be provided for prevention of high pressure core melt scenarios.

5.9 Means shall be provided to prevent containment melt through.

6. Improvement of the design

Reference level:

6.1 The current design shall on a regular basis, and when needed as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach, against current requirements and practices to identify deviations. The safety significance of identified deviations shall be determined with respect to possible design improvements or backfitting or other measures justified from a safety point of view.

[8] It is understood that the means mentioned in 5.3-5.9 shall be able to perform its functions in relevant severe accident conditions, although not formally qualified.

Safety area: Operation
Issue: Beyond Design Basis Accident Management

1. Procedures and guidelines for dealing with beyond design basis accidents

Reference levels:

1.1 Symptom based Emergency Operating Procedures (EOPs) shall exist to re-establish or compensate for lost safety functions. They shall include measures or actions to prevent core damage, such as feed and bleed, alternative water and power supplies.

1.2 Transition from the use of EOPs to Severe Accident Management and corresponding organisational arrangements shall be stated in procedures[9].

1.3 Guidelines / procedures shall address containment protection including hydrogen management, temperature, and pressure control inside the containment..

1.4 Guidelines / procedures shall address containment isolation and protection of personnel and the public including management of exposures and releases.

1.5 Guidelines / procedures shall address core debris cooling and prevention of re-criticality.

1.6 Guidelines / procedures shall address prevention of high pressure core melt scenarios.

2. Training and exercises for accidents beyond design

Reference levels:

2.1 The control room staff and on-site technical support shall be regularly trained and exercised, using simulators and diagnostic tools for at least the EOPs, and as far as practicable for the guidelines for management of severe accidents.

2.2 The transition from EOPs to guidelines for management of severe accidents shall be exercised.

2.3 Planning and regular exercises shall exist for emergency repairs and other interventions needed to restore necessary safety functions.

[9] It is understood that EOPs are normally used by the control room staff and mostly address prevention within and beyond the design basis, while guidelines for management of severe accidents are normally used by the emergency management team and mostly address mitigation of severe accidents.

Safety area: Safety verification
Issue: Probabilistic Safety Analysis (PSA)

1. Scope and content of PSA

Reference levels:

1.1 PSA shall be developed for levels 1 and 2.

1.2 PSA shall include all modes of operation, all relevant initiating events and hazards, including internal fire, internal flooding, severe weather conditions and seismic events.

1.3 PSA shall include all relevant dependencies (functional dependencies, area dependencies and other common cause failures).

1.4 PSA shall contain uncertainty and / or sensitivity analyses.

1.5 PSA shall be based on a realistic modelling of plant response, taking into account human performance to the extent assumed in operating and accident procedures.

1.6 Human errors shall be analysed, taking into account the factors which can influence the performance of the operators in all plant states.

2. Quality of PSA

Reference levels:

2.1 PSA shall be performed, documented and maintained according to the quality management system of the licensee.

2.2 PSA shall be performed according to the state-of-the-art methodology.

3. Use of PSA

Reference levels:

3.1 PSA shall be used for safety management purposes. Its role in the decision making process shall be defined.

3.2 PSA shall be used to identify the need for modifications to the plant and its procedures, in order to reduce the risk from the plant.

3.3 PSA shall be used to assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff edge effects".

3.4 PSA shall be used to assess the adequacy of plant modifications, changes to technical specifications and procedures and to assess the significance of operational occurrences.

3.5 Insights from PSA shall be used as input to development and validation of the safety significant training programmes of the licensee, including simulator training of control room operators.

Safety area: Safety verification
Issue: Periodic Safety Review

1. Objective of the periodic safety review

Reference levels:

1.1 The licensee shall have the prime responsibility for performing the review.

1.2 The review shall confirm the compliance of the plant with its licensing requirements and any deviations shall be resolved.

1.3 The review shall identify and evaluate the safety significance of deviations from applicable current safety standards and best practices.

1.4 All reasonably practicable improvement measures shall be taken by the licensee as a result of the review.

1.5 An overall assessment of the safety of the plant shall be provided, and adequate confidence in plant safety for continued operation demonstrated, as a result of the full scope review, taking into account all identified strengths and decided corrective actions, as well as any shortcomings that cannot be reasonably and practicably resolved.

2. Scope of the periodic safety review

Reference levels:

2.1 The review shall be made periodically, at least every ten years.

2.2 The scope of the review shall be clearly defined and justified.

2.3 The scope shall be as comprehensive as reasonably practical with regard to significant safety aspects of an operating plant.

2.4 As a minimum the following areas shall be covered by the review:

- plant design as built and actual condition of systems, structures and components,
- current safety analyses and their use,
- operating experience during the review period and the effectiveness of the system used for experience feed-back,
- organisational arrangements,
- safety performance and the effectiveness of safety and quality management,
- staffing and qualification of staff,
- emergency preparedness, and
- radiological impact on the environment.

3. Methodology of the periodic safety review

Reference levels:

3.1 The review shall use an up to date systematic and documented methodology, taking into account deterministic as well as probabilistic assessments.

3.2 Each area shall be reviewed and the findings compared to the licensing requirements as well as to current safety standards and practices. Conclusions shall be drawn with regard to reasonable and practical improvement measures taking into account interactions and overlaps between the different safety issues.

3.3 If studies made for other purposes are utilised in the periodic safety review, their contribution to the review shall be explained. These studies shall be available and appropriate references given.

Western European
WENRA
Nuclear Regulators Association

Position Paper

Western European
WENRA RHWG
Nuclear Regulators Association

Position paper

on Periodic Safety Reviews (PSRs) taking into account the lessons learnt from the TEPCO Fukushima Dai-ichi NPP accident

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Study by

WENRA Reactor Harmonization Working Group

March 2013

01
Introduction

A severe accident involving several units took place in Japan at Tepco's Fukushima Dai-ichi nuclear power plant in March 2011. The immediate cause of the accident was an earthquake followed by a tsunami coupled with inadequate provisions for tsunamis in the original design. Opportunities to improve protection against a tsunami were not taken in a timely and effective manner, which could have been possible for example as part of an effective periodic safety review (PSR) process.

IAEA organised in the end of August 2012 the 2nd Extraordinary Meeting of the Contracting Parties to the Convention of Nuclear Safety. The main topic of the meeting was the lessons learnt from the Fukushima Dai-ichi NPP accident. In the summary report of the meeting, the Contracting Parties were encouraged to reinforce efforts for continuous improvement by performing periodic reassessments of safety, through periodic safety reviews or alternative methods.

The European stress tests organised by the ENSREG also emphasised the importance of the PSR process1. In the action plan for the follow-up of the peer review of the stress tests performed on European nuclear power plants, ENSREG encourages WENRA to undertake a review of the associated Reference Levels, particularly with respect to external hazards2.

Since operation of the first generation of commercial nuclear power plants started in the 1950's, there have been substantial developments in safety standards, operating practices and in technology, resulting from new scientific and technological knowledge. Lessons have been learnt from operating experience and better analytical methods have been developed. These developments should be considered by the licensees and the regulatory bodies in the interest of continuous safety improvement.

One of the key aspects of nuclear safety and continuous improvement is the periodic safety review. WENRA reference revels (RLs) for existing nuclear power plants[3] cover the topic of PSR in Issue P. According to the RLs, the PSR shall be made periodically, at least every ten years. The review shall confirm the compliance of the plant with its licensing basis and any deviations shall be resolved. In addition, the review must consider any issues that might limit the future life of the facility or its components and explain how they will be managed. The review shall also identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available. All reasonably practicable[4] improvement measures shall be taken by the licensee as a result of the review.

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[1] ENSREG summary report, stress tests performed on European nuclear power plants, April 2012

[2] ENSREG action plan, Follow-up of the peer review of the stress tests performed on European NPP, July 2012

[3] WENRA Reactor Safety Reference Levels, January 2008

[4] The words Reasonably Practicable are used in terms of reducing risk as low as reasonably practicable or improving safety as far as reasonably practicable. The concept of reasonable practicability is directly

The licensee has the prime responsibility for performing the PSR. The regulator will review the results and can require the licensee to implement additional safety improvements in order to permit continued operation. In the end of the PSR process, the licensee shall collect all reasonably practicable improvement measures in an integrated implementation plan which is agreed with the regulator. The licensee and the regulator inform the government and the public about the scope and the results of the PSR and resulting safety improvements according to the national procedures within the regulatory oversight process. PSR scope, methodology and the roles and responsibilities are described in more detail in IAEA Safety Guide NS-G-2.10 “Periodic Safety Review of Nuclear Power Plants”, August 2003.

RHWG considers it important that the same method of PSR, which takes into account the potential consequences of safety challenges, is used in all phases of operation including the decommissioning phase of NPPs and for other nuclear facilities, such as research reactors and radioactive waste management facilities. Decommissioning phase of nuclear installations is covered by WENRA Decommissioning Safety Reference Levels Report[5] where RL D-55 says “The licensee shall carry out at regular intervals a review of the safety of the facility under decommissioning at a frequency established by the regulatory body”.

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analogous to the ALARA principle applied in radiological protection, but it is broader in that it applies to all aspects of nuclear safety. It should be taken to mean that, in addition to meeting the normal requirements of good practice in engineering, further safety or risk reduction measures for the design or operation of the facility should be sought and that these measures should be implemented unless the utility is able to demonstrate that the efforts to implement the proposed measures are grossly disproportionate to the safety benefit they would confer.

[5] WENRA Working Group on Waste and Decommissioning (WGWD), Decommissioning Safety Reference Levels Report, version 2.0, November 2011

02
The role of PSRs and the concept of continuous improvement

A strong PSR process is a very important contributor to continuous improvement of safety of nuclear power plants. In case that PSR results indicate the need for improvement measures, these measures are to be defined and implemented in a timely and effective manner.

WENRA published a pilot study on “Long term operation (LTO) of nuclear power plants” in March 2011. A main conclusion of this study was that regulators generally review the acceptability of continued operation through the process of PSRs. Enhancement to the safety level is generally achieved following the PSR process. The scope and the frequency can vary slightly depending on a country's specific practice; however they are on the whole in line with IAEA guidance. In all WENRA countries the general requirements for PSR have been specified in the national legislation and/or regulatory guidelines.

In all WENRA countries, licensees are expected to perform at least every ten years a PSR of their plant, which is an opportunity to review not only the conformity of the plant, but also identify the possible safety improvements. Safety improvements can be related to the plant design but also to organisational issues (management system, procedures,…). On the basis of the results of the PSR, regulators generally review the continued acceptability of the continuation of operation of the plant until the next PSR.

PSR significantly contributes to the continuous improvement of safety. The concept of continuous improvement is illustrated in Figure 1, which is a simplified representation of safety through plant life, and does not for example show the timescales for implementing plant improvements or the effects of ageing of plant systems, structures and components.

When the existing reactors were commissioned, their original safety level met the required safety level based on the safety requirements which were in force then. Safety requirements for NPPs can be updated based on the operating experience and safety research and advances in science and technology. New reactors are designed to meet higher levels of safety than the existing ones. Despite the fact that existing reactors undergo PSRs as a result of which safety enhancements are implemented, it is likely that there will remain a difference between the safety level of oldest and newest reactors. One example is a difference between the severe accident mitigation provisions integrated as a design basis in new reactors compared to the back-fitting measures in the older reactors. In some cases, it will be reasonably practicable to enhance safety to reach a higher safety level but sometimes further enhancement toward the benchmark is not reasonably practicable.

The need for improvements can also occur anytime between PSRs and significant issues that may put at risk the safety of the plant shall be addressed without delay. The safety assessments performed in WENRA countries after the TEPCO Fukushima Dai-ichi NPP accident or the Forsmark NPP event are also examples of actions performed outside the frame of PSRs.

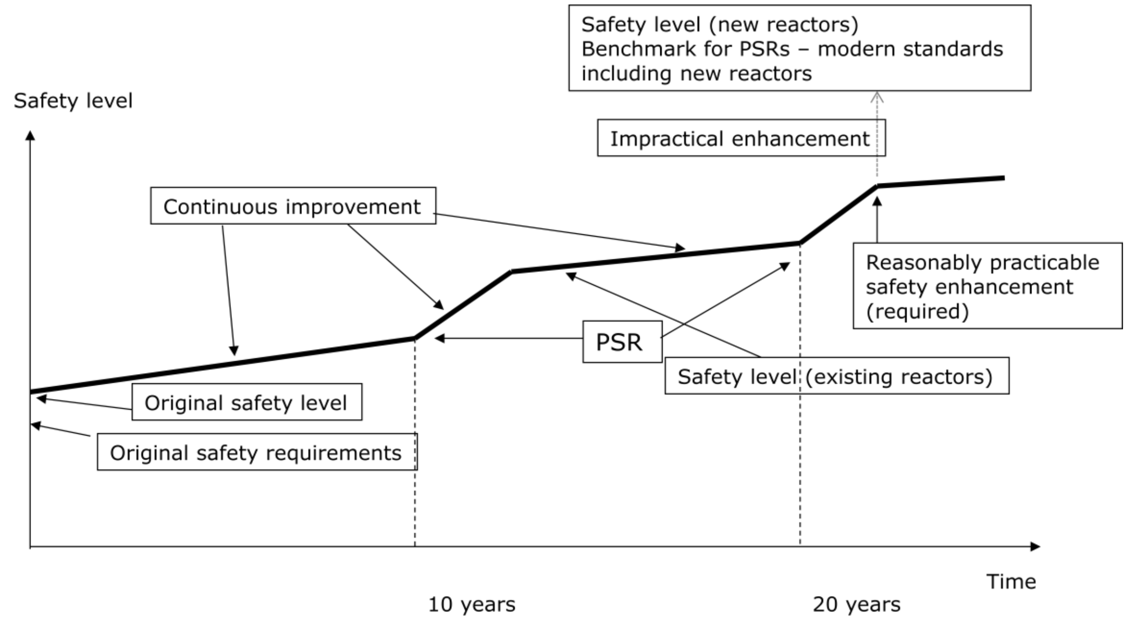

Figure 1. The concept of continuous improvement.

03
Safety standards and internationally recognized good practices used as reference in PSRs

It is stated in WENRA RL P1.3 that the PSR shall “identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available”. While developing the pilot study on LTO, the RHWG performed an internal study on the national practices in implementation of RL P1.3. The study was based on a questionnaire concerning the ‘reference level for the PSR’. This term is used in the IAEA Safety Guide NS-G-2.10 to represent the plant level constituted by the applicable safety goals, standards, methods, practices and the plant design basis, which is actually known as ‘benchmark level for PSR’ within the RHWG.

In a questionnaire, a specific item aimed at describing what are the “current safety standards” used in PSRs. The majority of the WENRA countries described them as consisting of the following:

- National nuclear law and regulations;
- National regulatory guidelines and standards; and
- IAEA safety standards;

Many WENRA countries include also

- WENRA Reactor Safety Reference Levels (RLs);
- standards and regulation of the country of origin of the reactor design or other countries; and
- safety requirements for new nuclear power plants;

And some countries include

- the WENRA safety objectives for new nuclear power plants; and
- the current level of science and technology.

In general, the differences in regulations, standards and approaches amongst the WENRA countries are not so large. The attempt to use requirements for new nuclear power plants in the PSR seems to impose a variety in the understanding of the “current safety standards” (from the original design basis, through the design extension and reaching the WENRA safety objectives for new nuclear power plants). However, implementation of all the WENRA RLs for existing reactors has a positive influence for the practical application of the standards in the PSR process. RLs include for example the concepts of design extension and severe accident management, and will be assessed at the latest in the next PSRs as the regulatory requirements continue to be harmonised within the WENRA countries.

In the WENRA statement on safety objectives for new nuclear power plants published in November 2010 it was stated that those objectives should also be used as a reference for identifying reasonably practicable safety improvements for existing plants during periodic safety reviews. Based on these safety objectives, WENRA decided to develop common positions on selected key safety issues for the design of new nuclear power plants. The report compiling these common positions can also be used as a more detailed reference in PSRs.

04
Reassessment of possible plant faults and hazards

The TEPCO Fukushima Dai-ichi accident demonstrates the importance of properly implementing the Defence-in-Depth principle, to ensure safety, getting the design basis for external hazards right, providing adequate protection against external hazards, and the need to ensure a strong PSR process together with independent regulatory body to drive it. The accident has also confirmed the need to undertake a comprehensive analysis of all potential plant faults and hazards as part of the PSRs using both deterministic and probabilistic methods in a complementary manner to provide as full coverage of all safety aspects as possible.

PSR should raise issues for further development of safety and those measures should be timely implemented that can be considered justified considering operating experience and safety research and advances in science and technology. In the safety assessment, specific considerations are needed for multi-unit sites and to address long term measures, as well as to cover all areas with significant amounts of radioactive material at the site.

The current WENRA RL P2.2 states that the scope of the PSR shall be as comprehensive as reasonably practical and defines areas that shall be covered as a minimum. The mentioned areas are derived from the 14 safety factors defined in the IAEA Safety Guide NS-G-2.10. This RL P2.2 mentions safety analyses in general but does not explicitly mention hazard analysis which is one of the IAEA safety factors. In RL P2.2, safety analyses include the analysis of plant faults and of hazards and also credible combinations and induced effects. They constitute both deterministic and probabilistic aspects.

IAEA Safety Guide NS-G-2.10 defines the safety factor of hazard analysis as to ensure that SSCs important to safety, including the control room and the emergency control centre, are adequately protected against relevant internal and external hazards. For this safety factor, the PSR review should take into account the current methodology, analytical methods, safety standards, knowledge of credible magnitude and associated frequency of occurrence of the hazard (and uncertainty related to this knowledge), understanding of environmental effects, the capability of the plant to withstand the hazard based on its current condition, and appropriateness of operating organisation procedures to prevent or mitigate the hazard.

For external hazards, the list of relevant hazards that may affect plant safety shall be reviewed for completeness in PSRs. For each relevant hazard, the PSR shall verify, by means of current methodology, analytical techniques and data, that the frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or that the preventive and mitigatory measures in place are adequate. For example, there should be an assessment of the impact of any changes in hazard levels, due to changes in hazard magnitude derivation methodologies. If the hazard level has changed the SSCs which are expected to resist the hazards should be reassessed to confirm their hazard withstand capability.

Due to the Fukushima Dai-ichi accident the European stress tests (ENSREG) was performed which included the assessment of external hazards including in particular earthquake, flooding, and extreme weather conditions and combination of hazards. As part of the PSR process, the safety justification against external hazards shall be re-evaluated at least every ten years if not specifically addressed otherwise.

On multi-unit sites, the plant should be considered as a whole in safety assessments and interactions between different units need to be analysed. Hazards that may affect several units need to be identified and included in the analysis. It would be preferable to carry out the site specific studies for all units at the plant site at the same time, taking into account the possible interactions among different units. Even if some PSR studies were applicable to several similar NPPs, site specific aspects should be reviewed separately in PSRs.

05
Possible changes in PSR procedures based on the lessons learnt from the TEPCO Fukushima Dai-ichi accident

Concerning the scope of PSR, it is recognised that natural hazards should be more systematically reviewed during a PSR. No major modifications are expected concerning the PSR process itself. However, it is expected that WENRA countries take measures to make the process as transparent as possible to the stakeholders and the public. For example, the outcome of the PSR including resulting safety improvements should be published. This should improve societal confidence on the nature and scope of the PSR and the licensees commitment to address any PSR findings.

06
The need to update WENRA RLs related to PSR

The IAEA Safety Guide NS-G-2.10 is currently under revision. No major modifications are planned in the latest draft of this revision which would influence significantly the current WENRA RLs Issue P.

As lessons learnt from the Fukushima Dai-ichi NPP accident, the following areas are recognised for improvements in the WENRA RLs Issue P:

- the timely and effective implementation of improvements derived from the PSR
- review of site characteristics regarding external hazards
- more explicit guidance on the need for comprehensive analysis of all hazards and plant faults
- taking into account multiple-unit issues.

WENRA
WESTERN EUROPEAN NUCLEAR
REGULATORS ASSOCIATION

RHWG
REACTOR HARMONISATION
WORKING GROUP

WGWD
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Western European
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Nuclear Regulators Association

WENRA

Pilot study on Long term operation (LTO) of nuclear power plants

March 2011

Western European

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Pilot study on Long term operation (LTO) of nuclear power plants

Study by

WENRA Reactor Harmonization Working Group

March 2011

RHWG

Reactor Harmonization Working Group

TABLE OF CONTENT

1. INTRODUCTION

Many operators in Europe have recently expressed their intention to operate their nuclear power plants longer than foreseen by their original design (this is referred to in this document as "long term operation" or LTO). This happens in a context where new plants are under construction in Europe and where these new plants are designed to meet higher safety levels than the existing ones.

Regulators will have to take position on the safety aspects of continued operation of nuclear power plants. To achieve better consistency between these positions, WENRA asked the Reactor harmonization working group (RHWG) to consider the issue of continued operation of existing nuclear power plants.

2. OVERVIEW OF THE SITUATION IN WENRA COUNTRIES

A questionnaire on "long term operation" was circulated inside the RHWG, from which the main conclusions are the following:

1) All WENRA member countries, except Italy and Lithuania, operate one or more reactors. About one quarter of these reactors are older than 30 years, a few of them having already exceeded 40 years of operation;
2) In most WENRA member countries, there is no reference to the lifetime of the plant in the license. However, in the safety analysis report, there are generally some design assumptions related to the lifetime of some key components, of which the reactor pressure vessel is the most important one. When such values are mentioned, they are generally between 30 and 40 years;
3) When a lifetime is specified in the license, the licensee has in general the possibility to ask for an extension, which needs to be supported by appropriate ageing management programmes and other relevant justifications;
4) In both cases (2) and (3), regulators generally give a position on continuation of operation through the process of periodic safety reviews (PSR), which periodicity is 10 years in every country;
5) In a majority of WENRA member countries, operators have already expressed their intention to operate some of their plants beyond the "design lifetime" and generally for an additional 10 to 20 years. Only a few LTO justification files have already been submitted and reviewed by the regulators;
6) As for safety, two most common limiting factors for long term operation are identified in WENRA countries:
    - ageing of key systems, structures or components (in particular, those that are not replaceable),
    - fulfilment of "modern" safety requirements.

   Other limiting factors were mentioned such as personnel competence and skills;
7) Enhancement of the safety level is generally achieved following the PSR process and not only for LTO applications. However, PSR is not considered as the only tool/occasion to enhance the safety level;
8) Replacement of components such as steam generators, vessel heads, are performed in many countries throughout the lifetime of the plant, but rarely coupled to LTO. No country reported about key components that have to be specifically replaced in link with LTO;
9) Research programmes related to ageing are common practice for all countries. However some countries initiated R&D projects specifically dedicated to LTO.

3. SUMMARY OF THE RHWG DISCUSSIONS

3.1. <u>About the wordings “design lifetime” and “long term operation”</u>

RHWG found useful to clarify the concepts of “design lifetime” and “long term operation”.

3.1.1 <u>Design lifetime</u>

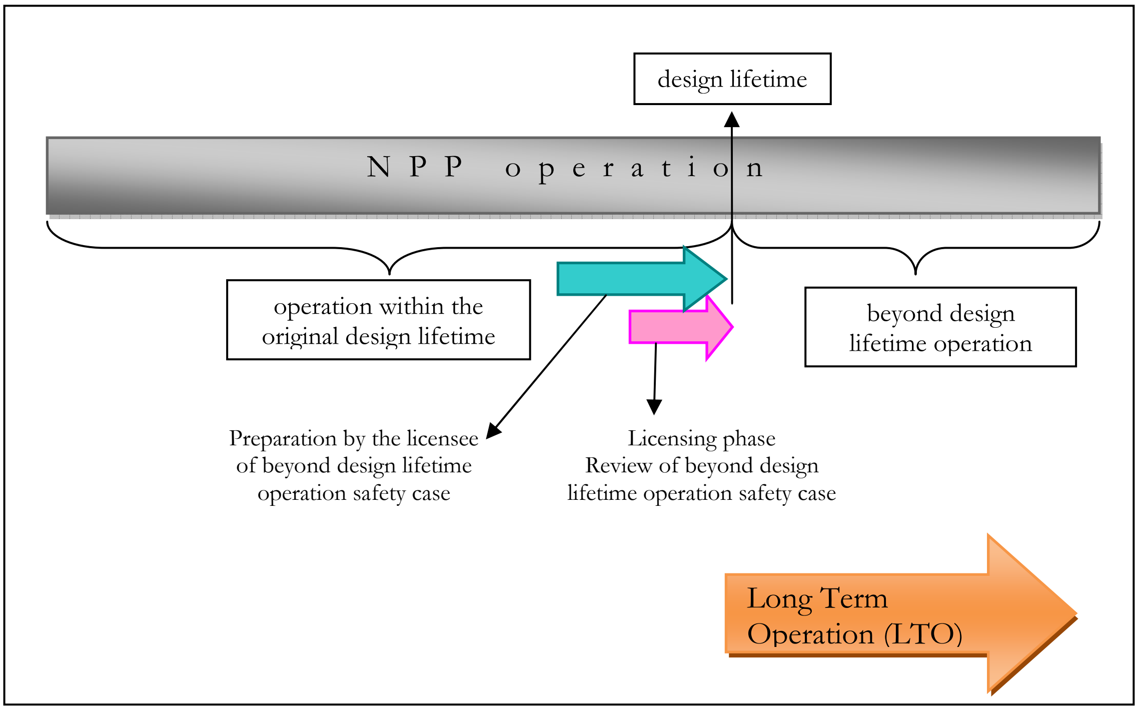
The definition of the design lifetime (or design life) can be found in the IAEA safety glossary:
“***Design life*** *- The period of time during which a facility or component is expected to perform according to the technical specifications to which it was produced.*”

This definition is referring to certain values used in technical specifications. The concept seems clear regarding a specific component, but is more difficult to perceive when related to a whole facility. In most of WENRA countries, there is no reference in the license to the lifetime of the plant. However, in the safety analysis report, there are generally several design assumptions related to the lifetime of some key components that cannot be replaced, such as the reactor pressure vessel. When such values are mentioned, it is generally between 30 and 40 years. Assuming this, the RHWG has agreed on a definition that “Design lifetime of a nuclear power plant is the minimal value of lifetimes of all its non-replaceable structures, systems and components”. It is to be underlined that in terms of safety, there may be no real cliff edge effect due to ageing when a nuclear power plant is being operated longer than the initial design lifetime of some of its components. For instance, the initial design lifetime of the reactor vessel may not be relevant anymore as having been re-evaluated considering actual plant operation and condition as well as current knowledge about ageing phenomena.

3.1.2 <u>Long term operation</u>

A definition of term Long Term Operation (LTO) can be found in IAEA Safety Report Series No 57 – Safe long term operation of nuclear power plants (2008): “***Long term operation*** *of a nuclear power plant may be defined as operation beyond an established time frame set forth by, for example, licence term, design, standards, licence and/or regulations, which has been justified by safety assessment, with consideration given to life limiting processes and features of systems, structures and components.*”

In this study, LTO has been understood as defined by the IAEA, taking as the “established timeframe” the design lifetime as understood by the RHWG (see above: design lifetime of a nuclear power plant is the minimal value of lifetimes of all its non-replaceable structures, systems and components), reminding that LTO should be in line with national regulation and nuclear power plant license.



3.2. <u>The two aspects of "long term operation"</u>

Continuation of operation of a nuclear power plant refers to two kinds of expectations:

- demonstrating and maintaining plant conformity to its currently applicable regulatory requirements;
- enhancing plant safety as far as reasonably practicable.

As a consequence, two reasons for limiting the lifetime of a plant to a certain value could be the following:

- it appears that at a given time, the plant will no more comply with its currently applicable regulatory requirements; or
- implementation of the safety enhancements that the regulator considers necessary for the plant to be further operated are not carried out.

RHWG considers that the first aspect (demonstrating conformity, even in the long term) is well addressed in the IAEA publications (as for example: Safety report series No. 57 – Safe long term operation of nuclear power plants or Technical report series No. 448 – Plant life management for long term operation of light water reactors). Exchange of experience feedback on the findings of conformity checks and on the acceptable methodologies to assess ageing of some key components (for instance the reactor pressure vessel) would be beneficial. This could be done under other frameworks than WENRA (for instance, bilateral relations, IAEA or NEA workshops…).

As a consequence, the discussions inside RHWG have been focused on the second aspect. New reactors will be commissioned which are designed to meet higher level of safety than the existing ones. Despite the fact that existing reactors undergo periodic safety reviews as a result of which safety enhancements are implemented, it is likely that there will remain a difference between the safety level of oldest and newest reactors (an example of a difference between existing and new reactors being the severe accident mitigation provisions – issue F in WENRA RLs). Whether this difference is acceptable or not in the long term implies not only technical judgement but also political, economical and financial considerations which are clearly out of the scope of the RHWG work. However, the RHWG can provide indications about what is technically feasible and foster harmonisation of the regulator's positions on this issue across WENRA countries.

3.3. <u>The role of periodic safety reviews (PSR)</u>

In all WENRA countries, licensees are expected to perform regularly ("*at least every ten years*", WENRA RL P 1.1.) a periodic safety review of their plant, which is an opportunity to review not only the conformity of the plant, but also to identify the possible safety improvements which could be implemented[1][2][3].

Not all periodic safety reviews are related to "long term operation" as defined above: for instance the first periodic safety review of a nuclear power plant takes place well before the components have reached their envisaged design lifetime. However, all periodic safety reviews have a link with continuation of operation of the plant: on the basis of the results of the periodic safety review, regulators generally take position on the continuation of operation of the plant until the next periodic safety review.

Hence, there is a link between a regulatory position on "long term operation" of a nuclear power plant, and the orientations and results of the last periodic safety review of this plant, in particular in terms of safety expectations. As a matter of fact, most WENRA countries have made a more or less explicit link between considering LTO and performing the corresponding PSR.

There were discussions whether a PSR related to LTO is or is not different from a "usual" PSR. The overall conclusion was that the methodology and scope are identical but some topics (e.g. ageing) would be paid a greater attention and that additional time for the review might be necessary. The forecast duration of further operation of the plant is a key parameter in the decision making process in such cases, in particular when identifying reasonably practicable enhancements. There was a general concern regarding potential consecutive applications for short periods of further operation in which some safety enhancements would not be reasonably practicable in one period but may be if the consecutive periods of time were amalgamated.

PSR scope and methodology are already described in IAEA safety standards and are not a priority topic for harmonisation inside WENRA. On the contrary, the "*applicable current safety standards and internationally recognised good practices currently available*" to be used as safety targets is a topic for harmonisation.

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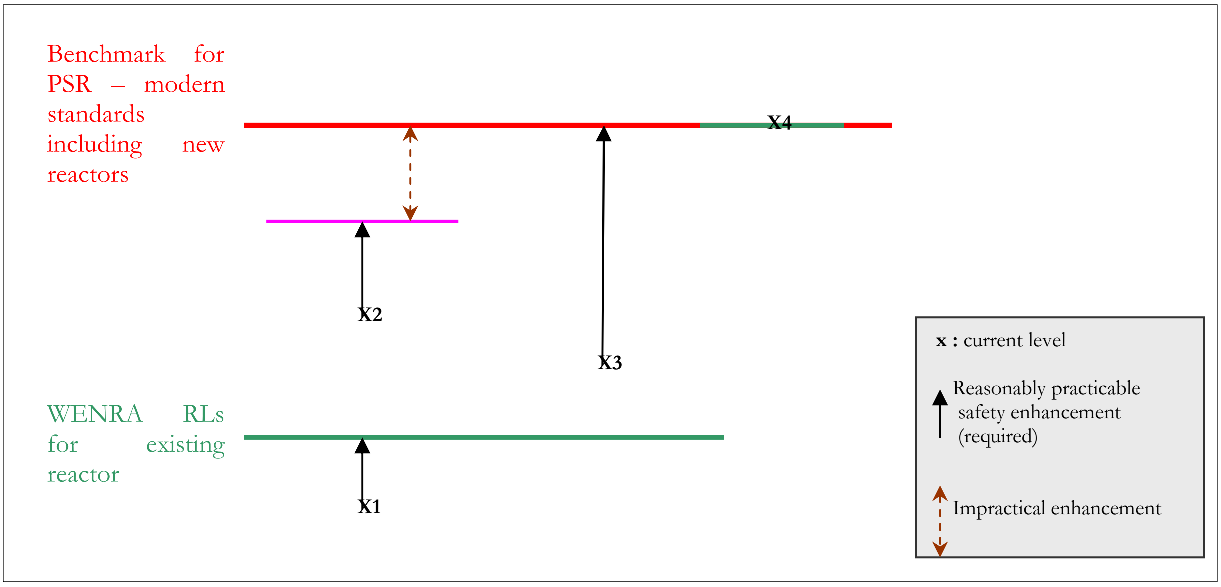
[1] Correcting anomalies actually improves the safety of the plant but should not strictly be considered as a "safety improvement" as it brings back the plant to its expected safety level.

[2] Safety improvements are related to plant design (plant modification) but also to operation practices (management system, operating procedures…)

[3] Improvements can also occur anytime between PSR Sometimes, it may not be acceptable to delay some safety improvements until the next PSR



The above diagram is conceptual and is intended to represent the process of comparing, for a particular feature, existing reactors with modern standards in a PSR and, where appropriate, moving towards the higher standard.

As for the horizontal lines:

- The green line represents WENRA RLs, and the "X" represent illustrative levels for a variety of safety issue;
- The red line represents modern standards, including but not restricted to WENRA's new reactor objectives, and is the bench mark for comparison in a PSR;
- The green and red lines may in some cases be at the same level (e.g. management for safety);
- The space between the green and red line represents the room for safety enhancements to be looked at.

As for the "x":

- The "X1" below the green line reflects the transition period to implement WENRA RLs allowed for in national plans for implementation;
- Those "X" below red line are safety issues that have to be compared to modern standards.
  - In some of these cases it will be reasonably practicable to enhance safety to reach the targets (redline) as in "X3";
  - In some cases, e.g. "X2", it will be reasonable to enhance safety to a level represented by the purple line, but further enhancement toward the benchmark is not reasonably practicable;
  - In other cases there may be no identifiable reasonably practicable options for enhancement;
- The "X4" represents these cases where the existing situation is already meeting the modern standard.

3.4. Applicable current safety standards and internationally recognised good practices currently available

It is stated in issue P of the WENRA Reference Levels" for existing plants, January 2008 version (P 1.1., 1.3., 1.4., 2.1.), that the periodic safety review shall "*identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available*". It is also stated that "*All reasonably practicable improvement measures shall be taken by the licensee as a result of the review*".

In their position statement on safety objectives for new nuclear power plants, WENRA members have stated that "*these objectives [safety objectives for new nuclear power plants as defined in the November 2010 document] should be used as a reference for identifying reasonably practicable safety improvements for […] existing plants during periodic safety reviews*".

This notably clarifies the reference that shall be considered in the periodic safety reviews. Regarding safety improvements that will be required for long term operation, one important element in the evaluation of what is "reasonable" will be the remaining time for which the considered plant will be operated before final shutdown.

4. MAIN CONCLUSIONS AT THIS STAGE

As a result of the discussions within the RHWG, the following facts about LTO can be formulated:

- There is no real cliff edge effect neither in the level of safety or technical degradation due to ageing when reaching the original design lifetime. The licensee may be able to justify operation beyond the original design lifetime;
- Periodic safety review is an appropriate time to assess long term operation;
- Technical ageing of components is one aspect of the LTO and is covered by existing documents and international standards. This means that it is not the main focus of the harmonisation work proposed by the RHWG;
- In periodic safety reviews for existing reactors, WENRA safety objectives for new nuclear power plants and other relevant modern standards should be used as a reference with the aim of identifying reasonably practicable safety enhancements. Regarding safety enhancements that will be required for long term operation, one important element in the evaluation of what is "reasonable" will be the remaining time for which the considered plant will be operated before final shutdown.

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PSA EXPLANATORY NOTE

March 2007

PSA EXPLANATORY NOTE

(Issued: March 2007)

Objective of the Note

The RHWG has agreed upon a "PSA- Explanatory Note" explaining the group´s understanding behind the current reference levels for PSA and the related benchmarking.

Roles of deterministic and probabilistic approaches in the safety analysis

We consider that the safety of nuclear power plants shall rely essentially on a deterministic design based on the concept of defence in depth. The design provisions adopted by the licensee are justified based on, among other elements, the study of a limited number of representative event sequences (bounding cases) resulting from the full range of postulated initiating events, and the application of deterministic rules and criteria which include margins and conservative assumptions The results of such studies must satisfy criteria intended to limit the consequences of the specified events. More severe consequences can be accepted for less frequent events or conditions.

In this respect, a probabilistic safety analysis (PSA) shall be used to complement the conventional deterministic analyses. Indeed, PSA is based on proven methods such that risk can be assessed realistically with the help of logical models representing the plant responses to a broad range of initiators and failures under different operating modes. The probabilistic evaluation of these models offers insights in the relative safety importance of initiators, response of SSC's and of operating procedures. PSAs provide an overall view of safety characteristics, including both equipment and operator's behaviour. PSA helps to assess whether the design objectives regarding reliability, protection against vulnerabilities and effectiveness of different lines of defence have been achieved satisfactorily. It can be used to prioritise the safety issues related to the design or operation of reactors, and it is also a tool to support the dialogue between the licensee and the regulatory body. For operating reactors, PSA contributes to assessment of their overall safety performance and highlights points for which design or operating changes can be examined or even judged necessary. For future reactors, PSA is developed while the design is being defined, so as to highlight situations involving multiple failures for which arrangements must be made to reduce their frequency or limit their consequences.

Scope and content of PSA

For each plant, a specific PSA shall be developed for level 1[1] and level 2[2] including all modes of operation, all relevant initiating events, including internal fire and internal flooding. The licensee has to develop a PSA which represents the plant specificities. When the licensee owns a standardized fleet, this can be obtained by developing a "basic PSA" which represents the reactor

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[1] *Level 1 PSA identifies the sequence of events that can lead to core damage, estimates the core damage frequency and provides insights into the strengths and weaknesses of the safety systems and procedures provided to prevent core damage.*

[2] *Level 2 PSA identifies ways in which radioactive releases from the plant can occur and estimates their magnitude and frequency. This analysis provides additional insights into the relative importance of accident prevention and mitigation measures.*

type, and that is adapted to each plant of the same type, taking into account its specificities.

Additionally, external hazards such as severe weather conditions and seismic events shall be addressed[3]. in the PSA so that the overall risk of a plant is assessed realistically.

Quality of PSA

The licensee shall document all the technical content of the study to ensure its traceability and facilitate applications. In particular, the results of the basic PSA, the uncertainty assessments and the sensitivity studies shall be presented in a clear and legible manner to enable detailed external review of the PSA.

It is important to note that PSA shall be performed according to up-to-date proven methodology, and taking into account international experience currently available.

Moreover the licensee shall regularly update the PSA to correspond to the operating experience and to reflect changes in the design of the plant, new technical information, and more sophisticated methods and tools that become available. The status of the PSA should be reviewed regularly to ensure that it is maintained as a representative model of the plant.

The quality requirements to be applied shall be commensurate with the role of the PSA in the licensee's decision making process. The more important the role, the better the quality requirements for:

- the scope of PSA application,
- the level of detail,
- up to date methodologies and modelling.

N.B: When we refer to international experience currently available, it is in terms of methodology and quality used to develop a PSA and not in terms of scope of the PSA. Therefore it is not contradictory that some countries do not meet RL 1.1 on the scope of PSA but meet RL 2.2 on the quality of PSA. It may simply mean that they have developed only a level 1 PSA with a sufficient level of quality (in compliance with up-to-date proven methodology, and taking into account international experience currently available).

Use of PSA

The uses of PSAs are very broad and can be encompassed by the term "PSA application". This term qualifies any approach to reactor safety management that makes use of probabilistic methods to support decision-making, particularly in terms of changes in design, operation and preparation for accident management.

As nuclear safety regulators, we decided to orient the reference levels on PSAs towards practical applications that clearly enhance the effectiveness of safety management. Other applications of PSAs are possible but it is up to the licensee to develop and apply them if he wishes so, provided that they do not degrade the safety level.

[3] *This means that these two hazards shall be included in the PSA, except if a justification is provided for not including them, based on site-specific arguments on these hazards or on sufficient conservative coverage through deterministic analyses in the design, so that their omission from the PSA does not weaken the overall risk assessment of the plant.*

The methods and data used for PSAs and its characteristics—including their scope— depend on the application. The relevance of the PSA results must be assessed against the findings of other safety analyses on a case by case basis, according to the application considered. For certain applications, probabilistic objectives (absolute or relative values, total or partial) can be set, taking into account the uncertainties. Nevertheless, this is not required by our reference levels because the added value of a PSA for safety in general does not require quantitative objectives; some countries practice that such objectives must be considered as guideline values and not as strict limits.

Uncertainties and limitations of PSAs

There are basically two types of uncertainties: uncertainties related to quantitative input data and uncertainties related to modelling and simplifications. Concerning the uncertainties related to the most important quantitative input data, Monte Carlo simulation can be used to obtain the uncertainty of the overall result.

The uncertainties related to modelling and simplifications and to the assumptions made for quantification include the initiating event grouping choices, the choices of scenarios and models for the supporting thermohydraulic and neutronics calculations, the uncertainties related to knowledge of the phenomena, the uncertainties related to the modelling of human actions, to the simplified modelling and the estimation of software reliability, to the estimation of the reliability of equipment operating beyond its qualification conditions, and to the choice of probabilistic methods. The variation of the results according to the principal simplifications and assumptions can be assessed by means of sensitivity studies.

The limitations of PSAs concern their completeness. The level of completeness is assessed according to the relevance of the models, the difficulties associated with quantification and with regards to the use of the results. Incompleteness concerns, for example:

- the scope (lack of processing of internal fire or flooding events or external events),
- the choice of human interventions processed in the PSAs,
- the definition of the component families affected by the common cause failures (common cause failures affecting components belonging to different systems not being processed in all cases),
- unidentified scenarios.

The impact of incompleteness cannot usually be assessed quantitatively. Nevertheless, its assessment contributes to defining the limits of the scope of PSAs

The uncertainties and the limits associated with PSAs imply that the interpretation of their results and their use in the decision-making process should be done in a cautious way. On the other hand the PSA makes visible the uncertainties and limitations that otherwise would be hidden behind deterministic assumptions. Therefore it is essential that all possible contributions from different kinds of safety analysis can be integrated into a consistent overall picture.

PSA and risk informed approach
Reference levels on PSAs do not explicitly refer to risk-informed approach as this term has different meanings according to countries.

A risk-informed approach is justified where the two complementary processes of deterministic and probabilistic assessment lead to a more complete basis for decision-making in order to maintain or improve safety. Therefore a risk-informed approach based on deterministic design assumptions complemented by a probabilistic assessment can be useful in order to address design and operational issues in an efficient and effective manner.

However, a risk-based approach solely based on numerical results might be detrimental to safety.

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Guidance Document

Issue F: Design Extension of Existing Reactors

September 2014

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Guidance Document

Issue F: Design Extension of Existing Reactors

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Guidance for the WENRA Safety Reference Levels for existing Reactors in their update in relation to lessons learned from the TEPCO Fukushima Dai-Ichi accident.

29 September, 2014

Western European
WENRA RHWG
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00
Introduction

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The purpose of this Guidance is to provide explanations of the intent of the Safety Reference Levels (RLs) of Issue F, to contribute to a consistent interpretation and to permit insights into the considerations which have led to their formulation. In addition, some background information is provided for easy reference. This Guidance does not define any additional requirements. Furthermore, it is important to recognize differences in national regulations and in reactor designs when using this document. However, the overall content and meaning is in all cases relevant.

Section 2 of this Guidance includes a listing for design extension conditions which are needed to be taken into account in the safety analyses. Furthermore, a listing of initiating and consequential events for design basis accidents has been included in this Guidance as an Annex, although it is relevant for Issue E (Design Basis Envelope for Existing Reactors). This is considered useful as it contributes to an overall picture of the foundation for both design basis accidents and design extension conditions (see also Figure 1).

01
Objective

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F1.1 As part of the defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety of the nuclear power plant by:

- **enhancing the plant's capability to withstand more challenging events or conditions than those considered in the design basis**
- **minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events and conditions.**

Conditions more complex and/or more severe than those postulated as design basis accidents (DBAs) can occur. These conditions shall be investigated as Design Extension Conditions (DEC) so that any reasonably practicable[1] measures to improve the level of safety of a plant, compared to the level reached with the design basis (Issue E), are identified and implemented.

In Issue F, Design Extension Conditions are consistent with the definition in IAEA SSR-2/1:

> *Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.*

This includes the cases in which, for existing reactors, such considerations occurred after the initial design of the plant has been completed.

The treatment of DECs in IAEA SSR-2/1 is also acknowledged, in particular requirement 20 and the following text:

> *A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences if they do occur. […]*

> *The main technical objective of considering the design extension conditions is to provide assurance that the design of the plant is such as to prevent accident conditions not considered design basis conditions, or to mitigate their consequences, as far as is reasonably practicable.*

[1] Determining "reasonably practicable" implies weighing the efforts to reduce the risk against the benefits of risk reduction.

It should be noted that "further improving the safety" as stated in the reference level is not referring to the concept of "continuous improvement". This concept has been introduced in RL A2.3, which is referred to in F5.1. The improvement addressed in RL F1.1, on the other hand, is a process which is performed once by assessing whether the requirements laid down in the RLs of Sections 1 to 4 of Issue F are fulfilled, and implementing the necessary measures in those cases (if any) where they are not. (This process may be performed at different times for different fields.)

The main criterion for the implementation of improvements is reasonable practicability. What is reasonably practicable may change over time, for example because of developments in technology. Hence, there is a need for regular review of the DEC (see RL F5.1), which is a part of continuous improvement as addressed in RL A2.3.

All possible conditions exceeding the design basis events for which reasonably practicable measures can be identified to prevent accident sequences leading to severe fuel damage and/or to mitigate their consequences are included in DEC. Thus, DECs include sequences where severe fuel damage can be avoided (including multiple failure sequences), as well as severe accident sequences – corresponding to the two categories of DEC defined below (RL F1.2). This is presented schematically in Figure 1.

However, there may be conditions exceeding design basis events for which no additional measures are required to prevent severe accidents, due to the existence of margins in the design basis, or due to provisions which had been installed earlier.

The required capability of the plant to withstand the design basis events is determined based on conservative analyses. In addition, the licensee may decide to set design specifications exceeding the required capability, providing what can be called a design reserve. Furthermore, the actual capability of the SSCs may exceed this required capability, due to the chosen design and construction options (robustness). The use of conservative methodologies for analyses, the design reserve and robust design and construction of SSCs lead to a certain margin for the capability of the plant to withstand the design basis events.

Due to this margin, the plant will in reality be able to cope with some more challenging events than those covered by the design basis events and severe fuel damage could therefore be avoided in these cases. These more challenging events are belonging to the DEC A (DEC category for which severe fuel damage is to be prevented, see RL F1.2). It is one of the objectives of the DEC analysis to evaluate whether the extent to which the plant is capable to withstand more challenging events is sufficient. If this is not the case, reasonably practicable improvements should be implemented to enhance the plant's capability to withstand the DEC A.

In case of the more challenging events with postulated severe fuel damage, belonging to DEC B (DEC category with severe fuel damage, see RL F1.2), there will be conditions in the containment which can differ very markedly from those in case of design basis events. Therefore, mitigative provisions are likely to be needed for DEC B to minimize the radioactive releases harmful to the public and the environment as far as reasonably practicable.

The topic of margins is discussed further in the guidance to F3.1 (f).

There are a number of clear and basic differences regarding the treatment of DBA and DEC, e.g.:

- Methodology of analysis: Conservative or best estimate plus uncertainties for DBA, best estimate (with or without uncertainties) acceptable and, in some cases, preferred (see guidance to RL F3.1) for DEC; additional postulates like single failures for DBA, no systematic additional postulates for DEC.
- Technical acceptance criteria: Generally less restrictive and based on more realistic assumptions for DEC.
- Radioactive releases tolerated: Higher consequences are usually tolerated (if it is demonstrated that releases are limited as far as reasonably practicable) for DEC.

F1.2 There are two categories of DEC:

- **DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;**
- **DEC B with postulated severe fuel damage.**

The analysis shall identify reasonably practicable provisions that can be implemented for the prevention of severe accidents. Additional efforts to this end shall be implemented for spent fuel storage with the goal that a severe accident in such storage becomes extremely unlikely to occur with a high degree of confidence.
In addition to these provisions, severe accidents shall be postulated for fuel in the core and, if not extremely unlikely to occur with a high degree of confidence, for spent fuel in storage, and the analysis shall identify reasonably practicable provisions to mitigate their consequences.

Objectives

To reach the objective of enhancing the plant's capability to withstand events or conditions which are more challenging than those considered for the definition of the design basis, and to minimise radioactive releases as far reasonably practicable, both prevention and mitigation of severe accidents are highly important. Category DEC A deals with prevention, whereas category DEC B concerns mitigation. Based on the principle of defence-in-depth, preventive measures have clear precedence over mitigative measures. There are differences regarding selection for analysis and objectives between DEC A where the aim is to avoid fuel damage, and DEC B where severe fuel damage is postulated.

The requirements in the RL differ for fuel in the reactor core and for spent fuel in storage:

- Despite all reasonable preventive measures, DEC with severe core damage have to be considered with the purpose of identifying reasonably practicable mitigative measures.
- Measures for sufficiently mitigating the consequences of severe accidents in spent fuel storages could be difficult to implement. Therefore, it is the goal that such accidents are extremely unlikely with a high degree of confidence.

Events extremely unlikely to occur

The demonstration that an accident is extremely unlikely with a high degree of confidence should take account of the assessed frequency of the condition and of the degree of confidence in the assessed frequency. The uncertainties associated with the data and methods should be evaluated, including the use of sensitivity studies, in order to underwrite the degree of confidence claimed. The demonstration should not be claimed solely based on compliance with a general cut-off probabilistic value. Probabilistic and deterministic elements both are required for this demonstration.

It should be ensured that the provisions relied upon to demonstrate the extreme unlikeliness remain in place and valid throughout the plant lifetime. For example, in-service inspection and other periodic checks may be necessary.

All analytical methods applied should be validated against the specific phenomena in question, and verified.

The concept of “extremely unlikely with a high degree of confidence” constitutes an essential element of the concept of “practical elimination”, as defined by IAEA.
According to IAEA SSR-2/1, *“[t]he possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high level of confidence to be extremely unlikely to arise”*. This is further discussed and elaborated in Position 5 of the RHWG Report “Safety of new NPP designs” of March 2013.

The term “practical elimination” has not been used in the RLs. It is usually applied almost exclusively in the context of severe accidents leading to large or early releases. In the safety reference levels, and also in this Guidance, “extremely unlikely with a high degree of confidence” refers in some cases also to large or early releases; in other cases it refers to severe accidents in the spent fuel pool, and also to certain events (F2.2).

Apart from Issue F, “extremely unlikely with a high degree of confidence” is also used in Issue T.

02
Selection of design extension conditions

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F2.1 A set of DECs shall be derived and justified as representative, based on a combination of deterministic and probabilistic assessments as well as engineering judgement.

The DECs have to be selected and analysed for the purpose of further improving the safety of the nuclear power plant (see guidance to F1.1 regarding the meaning of "further improving") by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, events and accidents that compared to design basis events and accidents are either more severe or involve additional failures. Coverage of DECs can be provided by representative cases – analogous to the choice of a set of design basis events according to RL E4.2, which can serve as representative cases for design basis event analyses to cover all relevant events.

However, the approach of the analysis differs between design basis events and DEC. For the design basis events, the design and analysis are covered by considering conservative bounding cases. In the selection of representative cases for DEC analysis, where the aim is to identify reasonably practicable improvements, a more realistic approach should in general be used: Selecting a very demanding enveloping scenario for the DEC analysis, or setting a very low radiological target for mitigative measures, might lead to the conclusion that no reasonably practicable measures can be identified. Such an approach might not help to demonstrate that there are no reasonably practicable measures to achieve the plant's ability to withstand less demanding scenarios (still exceeding the design basis events). Therefore, the events which are considered in the selection of the representative DECs should cover a wide range of scenarios, from less demanding to more demanding (see also guidance to F2.2).

F2.2 The selection process for DEC A shall start by considering those events, and combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to severe fuel damage in the core or in the spent fuel storage. It shall cover:

- **Events occurring during the defined operational states of the plant;**
- **Events resulting from internal or external hazards;**
- **Common cause failures.**

Where applicable, all reactors and spent fuel storages on the site have to be taken into account. Events potentially affecting all units on the site, potential interactions between units as well as interactions with other sites in the vicinity shall be covered.

This RL refers to the selection process for DEC A. It stipulates that a wide scope of events and combination of events exceeding the design basis events which may lead to severe fuel damage in the core or in the spent fuel storage are to be considered at the beginning of this process. This is followed by a process of narrowing down the range in the further course of the selection procedure.

The selection process of representative scenarios should notably make use of the PSA results, the overall understanding of the physical phenomena involved, the margins in the design and the systems' redundancy and diversity. In cases in which this does not provide a sufficient basis for the selection process, preliminary analyses of accident sequences triggered by events and combination of events should also be performed.

Only a sub-set of the events and combinations of events considered at the start will be selected for DEC A. From this sub-set the representatives DECs according to RL F2.1 are derived, which subsequently are subjected to the DEC analysis (see RL F3.1).

The initiating events considered as the basis for the selection of DECs of category A should be justified and take into account the following list[2]. In addition, a plant and site specific adjustment and justification will be necessary to demonstrate that a comprehensive set of DECs of category A has been compiled.

Thus, the final sets of conditions selected for DEC A analysis will be plant and site specific, developed on the basis of the following non-exhaustive list.

<u>Initiating events for design extension conditions (DEC A):</u>

- initiating events induced by earthquake, flood or other natural hazards exceeding the design basis events (see Issue T)[3]
- initiating events induced by relevant human-made external hazards exceeding the design basis events[3]
- prolonged station black out (SBO; for up to several days[4])
  - SBO (loss of off-site power and of stationary primary emergency AC power sources)
  - total SBO (SBO plus loss of all other stationary AC power sources), unless there are sufficiently diversified power sources which are adequately protected
- loss of primary ultimate heat sink, including prolonged loss (for up to several days)
- anticipated transient without scram (ATWS)
- uncontrolled boron dilution
- total loss of feed water

[2] The list mainly applies to PWR and BWR. For other designs used in WENRA countries (AGR and PHWR), the list will need to be adapted to the reactor type and justified to the regulatory authority of the relevant country.

[3] This could include subsequent loss of ultimate heat sink combined with station black out or combined with a total station black out.

[4] The prolonged loss of function should consider the time period until external help and/or recuperation of safety systems can be established.

- LOCA together with the complete loss of one emergency core cooling function (e.g. HPI or LPI)
- total loss of the component cooling water system
- loss of core cooling in the residual heat removal mode
- long-term loss of active spent fuel pool cooling
- multiple steam generator tube ruptures (PWR, PHWR)
- loss of required safety systems in the long term after a design basis accident

A listing of initiating and consequential events for the design basis is provided in the Annex to this Guidance. This listing is relevant for Issue E (Design Basis Envelope for Existing Reactors). It is included in the Guidance since it is considered useful to provide an overall picture of the foundation for both design basis accidents and design extension conditions in this Guidance.

Events and combinations of events that can be regarded as extremely unlikely with a high degree of confidence (see guidance to F1.2 for interpretation), based on information available prior to the DEC selection process or on deliberations performed during this process, do not need to be considered further for the DEC selection. For example, this can apply to a particular natural hazard that is extremely unlikely by appropriate site selection; or failure of the RPV, if it is considered extremely unlikely due to design, manufacturing, quality control etc. It may also concern some common cause failures (CCFs) which can be considered extremely unlikely with a high degree of confidence and thus are screened out, or large reactivity insertion.

For events or combinations of events, which cannot be considered with a high degree of confidence to be extremely unlikely to occur and which may lead to accident conditions more challenging than those included in the design basis accidents, the DEC A analysis should be carried out in order to ensure that they are already sufficiently covered (provisions or measures already realised by the design of the plant), or to identify reasonably practicable measures (additional provision or measure to be implemented) to prevent severe fuel damage.

It is conceivable that for an existing plant the analysis of a potential DEC A leads to the result that existing provisions are insufficient to prevent severe fuel damage and no further measures for improving the resistance of the plant on the prevention level are reasonably practicable. Although they are part of the DEC A analysis, the corresponding events or combinations of events will not be covered by the set of representative DECs of category A for the existing plant. In these cases, it has to be investigated if there are reasonably practicable means to mitigate their consequences within DEC B.

F2.3 The set of category DEC B events shall be postulated and justified to cover situations, where the capability of the plant to prevent severe fuel damage is exceeded or where measures provided are assumed not to function as intended, leading to severe fuel damage.

For DEC B (severe accidents) an approach different from that for the selection of DEC A has to be taken, since there will usually be a very large number of possible scenarios, based on a wide range of plant specific severe accident conditions and phenomena, which cannot all be captured at the start of a selection process. Accordingly, no list of initiating events is provided for DEC B.

A set of severe fuel damage scenarios has to be identified for analysis according to RL F3.1, covering the different situations and conditions which can occur at the outset and during the course of a severe accident. The selection process of representative scenarios should notably make use of the PSA results, the overall understanding of the physical phenomena involved, the margins in the design and the systems' redundancy and diversity. As far as necessary, preliminary analyses of scenarios should be performed as part of the selection process.

Ensuring adequate confinement of radioactive substances, especially by protecting the containment integrity, is the main goal in DEC B. Special consideration should be given to the sequences that could lead to large or early releases to the environment (e.g. high pressure core melt), in order to attenuate the threats or to show that these sequences become very unlikely to occur with a high degree of confidence (to the extent this is required in RLs F4.8 to F4.14).

For existing plants, it cannot be excluded that there are states with severe fuel damage which have to be postulated according to RL F2.3 and which

- were not considered in the past, and
- cannot be considered extremely unlikely with a high degree of confidence, and
- do not lead to the identification of practicable additional measures of prevention (DEC A) and/or mitigation (DEC B) of severe accidents, and
- lead to radiological consequences which exceed the acceptable limits (in particular, to large or early releases).

These cases should be identified and judged by the licensee on a case-by-case basis to determine whether the associated risk is acceptable. For cases where additional measures have been identified as practicable, but are not sufficient to render large or early releases extremely unlikely with a high degree of confidence, a similar judgment has to be made, taking into account the practicable measures.

03
Safety analysis of design extension conditions

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F3.1 The DEC analysis shall:

(a) rely on methods, assumptions or arguments which are justified[37], and should not be unduly conservative;

(b) be auditable, paying particular attention where expert opinion is utilized, and take into account uncertainties and their impact;

(c) identify reasonably practicable provisions to prevent severe fuel damage (DEC A) and mitigate severe accidents (DEC B);

(d) evaluate potential on-site and off-site radiological consequences resulting from the DEC (given successful accident management measures);

(e) consider plant layout and location, equipment capabilities, conditions associated with the selected scenarios and feasibility of foreseen accident management actions;

(f) demonstrate, where applicable, sufficient margins to avoid "cliff-edge effects"[38] that would result in unacceptable consequences; i.e. for DEC A severe fuel damage and for DEC B a large or early radioactive release;

(g) reflect insights from PSA level 1 and 2;

(h) take into account severe accident phenomena, where relevant;

(i) define an end state, which should where possible be a safe state, and, when applicable, associated mission times for SSCs.

[37] These methods can be more realistic than for DBA, including best estimate. Modified acceptance criteria may be used in the analysis.

[38] A cliff edge effect occurs when a small change in a condition (a parameter, a state of a system…) leads to a disproportionate increase in consequences.

The DECs which have been selected according to RLs 2.1 to 2.3 are to be subjected to the DEC analysis.

Point (a):

Justified methods depend on the type of analysis which is performed. The purpose of analyses performed for a DEC can be:

(1) to review whether the fundamental safety functions can be guaranteed by existing equipment (installed for design basis accidents) for the selected set of DEC A events; or otherwise

(2) to identify and to evaluate reasonably practicable preventive (DEC A) or mitigative (DEC B) measures for enhancing safety or enlarging margins to avoid possible cliff edge effects (see also (f)).

For (1), conservative approaches or best estimate methodology may be used. In case of (2), best estimate methodology should be preferred to avoid missing reasonably practicable improvements due to an unduly conservative approach (see also Guidance to F2.1 above).

Point (b):

In principle, it could be admissible to perform an analysis without considering uncertainties (see guidance to F1.1). However, the consideration of uncertainties is useful to ensure that the results of a best estimate analysis constitute a meaningful basis for the planning of reasonably practicable improvement measures.

Point (c):

The outcomes of the DEC analyses should be used for:

- Identification of SSCs that are important to prevent severe fuel damage (DEC A) or to prevent large or early releases (DEC B).
- Identification of administrative and procedural measures (operator actions, EOPs, SAMGs etc.) that are important to prevent severe fuel damage (DEC A) or to prevent large or early releases (DEC B).
- Identification of reasonably practicable additional provisions (regarding SSCs as well as administrative and procedural features) to prevent severe fuel damage (DEC A) or to prevent large releases and/or to allow sufficient time for protective actions for the public to be implemented (DEC B).

In addition, the general principle that radioactive releases harmful to the public and the environment have to be minimized as far as reasonably practicable has to be followed.

Point (f):

Within the analysis of DEC, cliff-edge effects should be identified and a sufficient margin to avoid cliff-edge effects should be demonstrated wherever applicable.

The onset of severe fuel damage would be the cliff-edge effect for a DEC A. What is considered as a sufficient margin to avoid a cliff-edge effect is to be decided on a case-by-case basis.

Different kinds of margins may have to be considered, depending on the nature of the DEC. The following examples illustrate this point for DEC A:

- For multiple failure events, the margin to avoid cliff-edge effects could be seen in various ways:
    - The capacity of required SSCs to achieve functional capability beyond their design basis needed to avoid severe fuel damage.

   - The number (or probability of occurrence) of additional failures, beyond a design basis accident, for which it remains possible to avoid severe fuel damage.

- For certain multiple failure events like total SBO, loss of primary ultimate heat sink and many other cases, the margin could be expressed in terms of the period of time available for measures to avoid severe fuel damage. The probability of these sequences may be taken into account.

- For events related to reactivity or loss of coolant, the margin could be expressed in terms of fuel temperature or enthalpy release.

- For external hazards within DEC, margins could in addition be expressed in terms of frequency or severity (see Guidance on Issue T for more information on natural hazards).

For postulated DEC B, the cliff edge effect should be understood in terms of a large increase of radiological consequences due to containment failure. A margin could be expressed in terms of likelihood or time delay of containment failure to occur.

Point (i):

When analysing a sequence in the framework of DEC analysis, an end state should be defined and justified for this analysis. For DEC A, the “defined end state” could be a “safe state” as defined in IAEA SSR-2/1:

> *Plant state, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and maintained stable for a long time.*

However, in case of DEC B, it is unlikely to reach such a safe state. Therefore, the DEC B analysis should cover a reasonable period of time, until some other defined end state is reached. This could be a “controlled state after severe accident”. This is a state after a severe accident where decay heat removal is ensured, the damaged or molten fuel is stabilized, re-criticality is prevented and long term confinement is ensured to the extent that there is limited release of radioactive nuclides.

04
Ensuring safety functions in design extension conditions

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General

F4.1 In DEC A, it is the objective that the plant shall be able to fulfil, the fundamental safety functions:

- **control of reactivity[39],**
- **removal of heat from the reactor core and from the spent fuel, and**
- **confinement of radioactive material.**

In DEC B, it is the objective that the plant shall be able to fulfil confinement of radioactive material. To this end removal of heat from the damaged fuel shall be established[40].

[39] Preferably, this safety function shall be fulfilled at all times; if it is lost, it shall be re-established after a transient period.
[40] For the fulfilment (or re-establishment) of the fundamental safety functions in DEC A and DEC B, the use of mobile equipment on-site can be taken into account, as well as support from off-site, with due consideration for the time required for it to be available.

For DEC A, the fundamental safety function of heat removal can be regarded as fulfilled if operation of the corresponding systems is interrupted for some time, but their function is restored without any relevant fuel damage occurring. In particular, when assessing the residual heat removal from the spent fuel pool, the thermal inertia which is provided by the water inventory of the pool has to be taken into account. However, all relevant cases of fuel inventory and decay heat power which are possible in the pool have to be duly considered, including the case of the reactor core being completely unloaded into the pool.

For DEC B, maintaining the fundamental safety function of confinement has the highest priority. The other fundamental safety functions are of importance insofar as they are required to support the confinement function. The irreversible loss of the confinement function, and the associated uncontrolled consequences, should be avoided. Severe accident management actions to prevent this irreversible loss of the confinement function which are leading to limited and controlled releases to the environment, are not considered as a loss of the confinement function if they are temporary, associated with specific predefined requirements (such as filtering of the releases) and do not lead to unacceptable off-site consequences[5], and thus are part of DEC B measures.

[5] However, consequences may justify the implementation of protective measures in the immediate vicinity of the plant, like evacuation of the public.

F4.2 It shall be demonstrated that SSCs[41] (including mobile equipment and their connecting points, if applicable) for the prevention of severe fuel damage or mitigation of consequences in DEC have the capacity and capability and are adequately qualified to perform their relevant functions for the appropriate period of time.

[41] SSCs including their support functions and related instrumentation.

Regarding the demonstration of the ability of SSCs to perform their functions under DEC:

- The verification of assured flow paths (in particular regarding the state of valves) and accessibility to critical SSCs in station black out conditions should be considered as an integral part of the demonstration of the capability of SSCs to perform their function relevant for safety.

- The "appropriate period of time" refers to the time after the event which is required to reach and sustain and end state according to RL F3.1 (i).

F4.3 If accident management relies on the use of mobile equipment, permanent connecting points, accessible (from a physical and radiological point of view) under DEC, shall be installed to enable the use of this equipment. The mobile equipment, and the connecting points and lines shall be maintained, inspected and tested.

Plant management under DEC may rely on the use of mobile equipment. This equipment and its storage place has to remain unaffected by the DEC (including the external hazards) in which the equipment is relied upon to meet the safety functions. This equipment should be able to operate under the conditions to be expected in this DEC. Consideration should be given to the location and number of connection points to guarantee their availability and timely accessibility under the conditions to be expected in this DEC, so that mobile equipment can be connected to the plant and provide the expected service.

A program for inspections, periodic testing and maintenance on mobile equipment should be established, in accordance with the requirements in Issue K.

F4.4 A systematic process shall be used to review all units relying on common services and supplies (if any), for ensuring that common resources of personnel, equipment and materials expected to be used in accident conditions are still effective and sufficient for each unit at all times. In particular, if support between units at one site is considered in DEC, it shall be demonstrated that it is not detrimental to the safety of any unit.

No further guidance is needed.

F4.5 The NPP site shall be autonomous regarding supplies supporting safety functions for a period of time until it can be demonstrated with confidence that adequate supplies can be established from off site.

The autonomy of the NPP site regarding supplies should be guaranteed for a period of time permitting transport of additional supplies on the site, taking into account the circumstances in case of design extension conditions, including external hazards exceeding the design basis and related potential damage to infrastructure. The period of time available should be justified by analysis, and then shown to be adequate by demonstrating that the supplies or materials can be delivered and utilised within this timescale[6].

Long-term sub-criticality

F4.6 In design extension conditions, sub-criticality of the reactor core shall be ensured in the long term[42] and in the fuel storage at any time.

[42] It is acknowledged that in case of DEC B, sub-criticality might not be guaranteed during core degradation and later on during some time in a fraction of the corium.

Regarding footnote 42, in case of core melt accidents (DEC B) re-criticality during the on-going core degradation in parts of a (previously) molten core may be difficult to model with any accuracy. Temporary re-criticality in a fraction of the corium is considered to be admissible as long as it is demonstrated that the confinement function is not threatened at any time.

Heat removal functions

F4.7 There shall be sufficient independent and diverse means including necessary power supplies available to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.

To secure the cooling of the core and the spent fuel, either an alternative ultimate heat sink (including a complete chain of systems providing a link to it) or a chain of independent and diverse systems of using the primary ultimate heat sink (if the primary ultimate heat sink is available for all events within the DEC involving external hazards[7]) should be in place. If there is an alternative ultimate heat sink, it should be independent as far as practicable from the primary ultimate heat sink (for example, water from river/water from pond, or seawater/air).

The alternative ultimate heat sink or the chain of diverse systems should be able to secure the cooling of the core and the spent fuel for an extended period of time in case of a design extension condition (beyond the point at which a defined end state (see guidance to RL F3.1) has been reached).

In case where the primary means to remove the decay heat from the core and the spent fuel in DEC are not effective anymore, the diverse means of decay heat removal shall be put into service, consistent with the timeframe defined in the safety analysis and actions described in EOPs and SAMGs.

Means which are used for design basis events and which are sufficiently robust to be available in DEC can be credited here, providing there is sufficient independence and diversity.

[6] Several WENRA countries stipulate a duration of 72 hours for this period of time.

[7] An example of a heat sink which is likely to be formally available in all cases is the atmosphere. However, some influences (temperature, moisture, volcanic or fire ashes, duststorm etc.) may impact its cooling efficiency, and hence its availability.

Confinement functions

In case severe spent fuel damage is considered in DEC B (RL F1.2), the RLs on confinement function should be applied, where relevant, to the spent fuel storages.

F4.8 Isolation of the containment shall be possible in DEC. For those shutdown states where this cannot be achieved in due time, severe core damage shall be prevented with a high degree of confidence.

If an event leads to bypass of the containment, severe core damage shall be prevented with a high degree of confidence.

Isolation of the containment penetrations should not impede vital functions which are needed for severe accident management (e.g. containment heat removal).

Special attention needs to be given to situations with an open containment during certain shutdown states. In this case, a core damage accident could more easily lead to large or early releases. Therefore, timely containment isolation should be guaranteed, or measures to prevent core damage with a high degree of confidence shall be available. Specific consideration has to be given to the time needed for the restoration of containment isolation and effective leak-tightness or for implementing the measures to prevent core damage, taking into account factors such as the progression of the accident sequences.

The reference to bypass of the containment in RL F4.8 is not to be interpreted as concerning failing isolation of a containment penetration or deliberate venting of the containment after the occurrence of an event. Rather, F4.8 refers to cases in which the event itself creates a pathway for leakages from the containment (for example, interfacing system loss of coolant accidents). In these cases, core damage could lead more easily to large or early releases and shall therefore be prevented with a high degree of confidence.

F4.9 Pressure and temperature in the containment shall be managed.

This RL covers all types of over-pressurization as well as risks related to under-pressure where relevant.

The following RLs F4.10 and F4.11 could be seen as special, important cases concerning different mechanisms of over-pressurization.

F4.10 The threats due to combustible gases shall be managed.

The threats due to combustible gases (including but not limited to hydrogen) should be understood to cover combustible gases which may originate from the reactor core, spent fuel storage (if applicable) or from the interaction of corium (from reactor core or spent fuel) with concrete. They also include combustible gases which migrate from the building where they were produced, for example into the containment venting system.

Furthermore, the threats due to combustible gases include high temperature resulting from combustion as well as pressure waves and formation of high-energy fragments (missiles) created by explosions.

F4.11 The containment shall be protected from overpressure. If venting is to be used for managing the containment pressure, adequate filtration shall be provided.

Over-pressurization by non-condensable gases and/or steam has to be taken into account. Venting of the containment may be one option to avoid the irreversible loss of the confinement function due to overpressure.

Should venting be used to protect against over pressurization of the containment, adequate filtering should be implemented so that:

- For off-site consequences, RL F4.14 is met;
- For on-site consequences, anticipated conditions referred to in LM3.5 and LM4.1 are not exceeded.

As a consequence, for some DEC A situations, filtration during venting may not be needed provided that the radiological consequences of the venting are acceptable (see F3.1 (d)).

For multi-unit sites, conditions at other units should be taken into account. Venting systems should be resistant to the relevant external events and DEC B environmental conditions for the time frame for which they are required to operate.

F4.12 High pressure core melt scenarios shall be prevented.

High-pressure core melt scenarios could lead to the irreversible loss of the confinement function. Therefore, it should be demonstrated that such scenarios are extremely unlikely with a high degree of confidence (according to the interpretation in the guidance to RL F1.2).

F4.13 Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

The RLs of Issue F do not require that fuel melt is generally rendered extremely unlikely with a high degree of confidence. Therefore, measures against containment degradation in case of fuel melt are required.

RL F4.13 applies to all situations with molten fuel spreading outside the reactor vessel and can concern for example the risks of steam explosions, direct containment heating or the basemat penetration by the corium. Instability of the reactor building caused by the mass of the water injected into this building as part of efforts to control the molten fuel should also be taken into account.

The advantages and disadvantages of different strategies have to be carefully weighed (for example, “dry cavity”, early cavity flooding).

F4.14 In DEC A, radioactive releases shall be minimised as far as reasonably practicable.

In DEC B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

(a) allow sufficient time for protective actions (if any) in the vicinity of the plant; and
(b) avoid contamination of large areas in the long term.

The delay of releases in time in DEC B is not only relevant for protective actions in the vicinity of the plant; it can also be important for the implementation of any additional measures in the plant (or neighboring units) to delay releases further, or to prevent them altogether.

This RL also implies that the leak tightness of the containment and its penetrations should be maintained in the long term in case of DEC A. Furthermore, it sets limits for the degradation of the containment leak tightness due to exposure to DEC temperatures, pressures and radiation (e.g. degradation of rubber seals), differentiating between DEC A and DEC B.

Instrumentation and control for the management of DEC

F4.15 Adequately qualified instrumentation shall be available for DEC for determining the status of plant (including spent fuel storage) and safety functions as far as required for making decisions[43].

[43] This refers to decisions concerning measures on-site as well as, in case of DEC B, off-site.

The status of the plant and the safety functions, as far as required, should be monitored or at least ascertainable in case of DEC. In particular, the instrumentation should reliably provide adequate information both on reactor core and spent fuel as well as containment status. The instrumentation should have been demonstrated to be able to perform its safety-related functions in DEC environmental conditions, in order to manage such accidents according to EOPs and SAMGs. Instrumentation for key parameters should also be able to perform its functions for a sufficient period of time in case of a total SBO (see guidance to F4.18).

The lighting at key locations for operators should also remain operational under DEC environmental conditions, for a sufficient period of time in case of a total SBO.

In case of DEC B, it has to be noted that information on the plant status (in particular, concerning the possibility of future releases) is also relevant for deciding on emergency measures off-site.

F4.16 There shall be an operational and habitable control room (or another suitably equipped location) available during DEC in order to manage such situations.

Habitability of the control room (or another suitably equipped location) should by preference be achieved by control room design features. In addition, temporary use of personal protection equipment may be taken into account while acknowledging the associated limitations of such equipment.

The “other suitably equipped location” could be a supplementary control room or local control panel, if they are adequately equipped and protected for management of the DEC (A and/or B).

Necessary information from instrumentation should be relayed to the operational control room (or another suitably equipped location) and be presented in such a way to enable a timely assessment of the plant status (including spent fuel storage) and safety functions as far as required in DECs.

Emergency power

F4.17 Adequate power supplies during DEC shall be ensured considering the necessary actions and the timeframes defined in the DEC analysis, taking into account external hazards.

There should be adequate means (stationary and/or mobile) to ensure the required power supply to support fundamental safety functions in case of DEC, including the external events within the DEC, as defined – for natural hazards – in Issue T.

This RL could be fulfilled by providing a stationary diverse AC power supply to account for common cause failures (for example: due to component failure or loss of primary emergency diesel generators’ cooling system) as part of DEC A provisions.

F4.18 Batteries shall have the adequate capacity to provide the necessary DC power until recharging can be established or other means are in place.

DC power supply should be provided during DEC for all functions that are required. For example, where appropriate:

- to guarantee uninterrupted power supply for needed I&C (accident instrumentation – see also RL F4.15),
- for valve drives required for containment isolation,
- to start emergency diesel generators.

DC power supply could be enhanced, for example, by improving battery discharge times, implementing load shedding strategies and preparing dedicated on-time recharging options.

05
Review of the design extension conditions

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F5.1 The design extension conditions shall regularly[44], and when relevant as a result of operating experience and significant new safety information, be reviewed, using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the selection of design extension conditions is still appropriate. Based on the results of these reviews needs and opportunities for improvements shall be identified and relevant measures shall be implemented.

[44] See RL A2.3.

This RL emphasizes that the regular assessment of the overall safety of a nuclear power plant, as required in RL A2.3, has to include the design extension conditions. Reasonably practicable measures for improvement which have been identified shall be implemented in a timely manner, in accordance with A2.3.

List of Acronyms

AC	alternating current
AGR	advanced gas-cooled reactor
AM	accident management
AOO	anticipated operational occurrence
ATWS	anticipated transient without scram
BWR	boiling water reactor
DB	design basis
DBA	design basis accident
DBE	design basis event
DC	direct current
DEC	design extension conditions
DiD	defence in depth
EOPs	emergency operating procedures
HPI	high pressure injection
IAEA	International Atomic Energy Agency
LOCA	loss of coolant accident
LPI	low pressure injection
NPP	nuclear power plant
PHWR	pressurized heavy water reactor
PIE	postulated initiating event
PMF	postulated multiple failure
PWR	pressurized water reactor
PSA	probabilistic safety assessment
RHWG	Reactor Harmonization Working Group
RL	(safety) reference level
RPV	reactor pressure vessel
SAMGs	severe accident management guidelines
SBO	station blackout
SSCs	systems, structures and components
WENRA	Western European Nuclear Regulators' Association

Figure 1:
Scheme of means, events and plant conditions

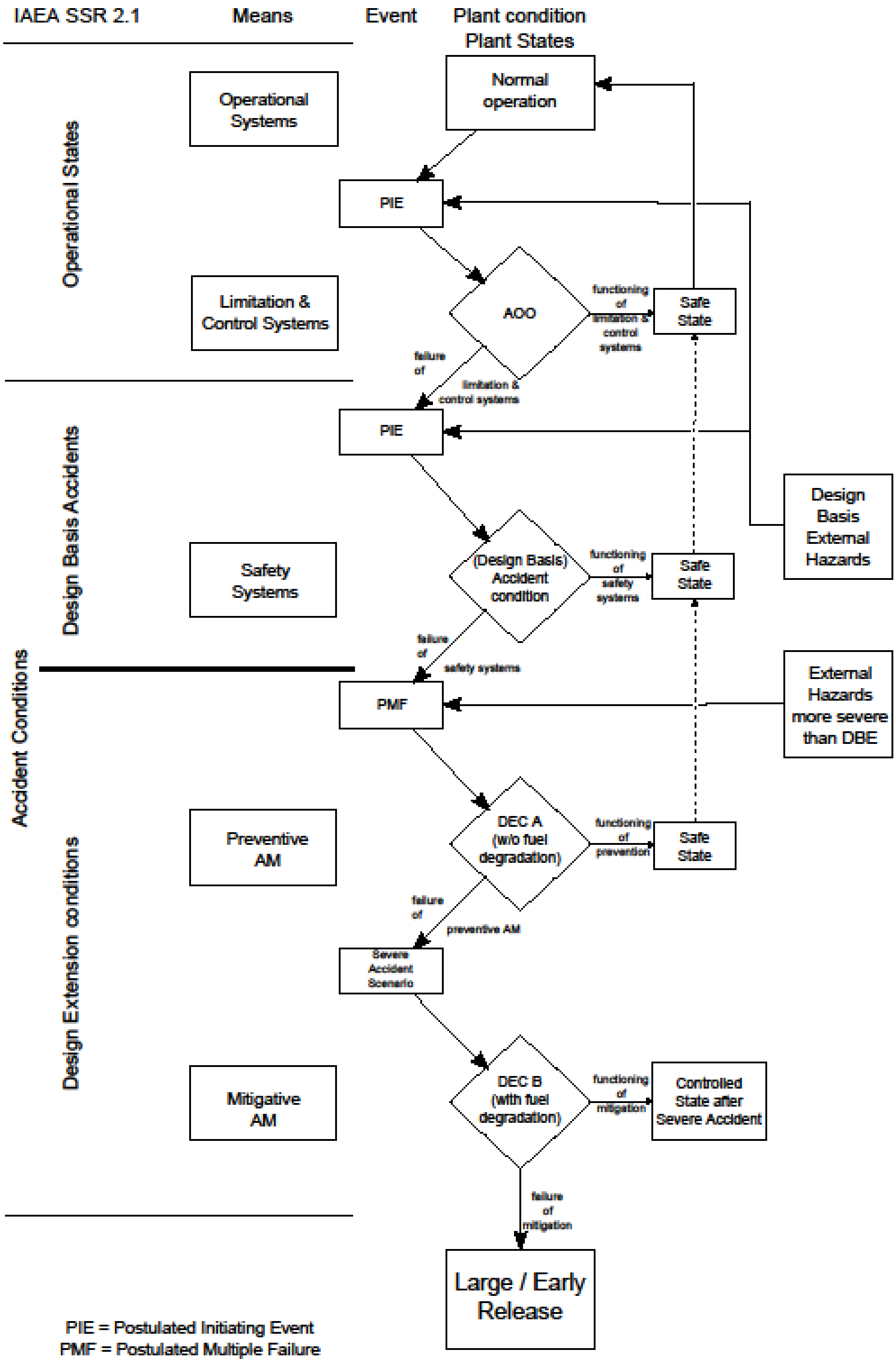
The Figure 1 gives a schematic, simplified overview of the means, events and plant conditions for the operational states and accidental conditions of an NPP. The goal of the figure is not to capture all details but to support the text in the Guidance. Human error, for example, is not mentioned explicitly in the figure. However, 'failure of safety systems', 'failure of preventive AM'… also include failures due to human error, where appropriate.

For clarity, the figure links in a simplified way postulated initiating events (PIEs) to design basis accidents (DBAs) and postulated multiple failures (PMFs) to design extension conditions (DECs), although for some plants some PMFs may be taken into account in the DBA list.

At the right hand side, the external hazards (including natural hazards and human made hazards) are shown as events leading to PIEs or PMFs. Other events (e.g. internal events) could have been added as well. However, the revision of the WENRA Safety Reference Levels contains a new Issue on natural hazards (Issue T). Therefore, only external hazards are shown in the figure 1 in order to illustrate how the natural hazards (and more generally the external hazards) do fit in the more general requirements on design basis (Issue E) and design extension conditions (Issue F).

The figure contains arrows and lines connecting the different plant conditions and events in a top down manner. This is a simplification for clarity as accident scenarios often will not follow such kind of gradual degradation. For example, there can be scenarios going directly from a PIE during normal operation to a design basis accident condition, and a common cause failure added to an AOO generally leads to DEC A. However, adding more possible arrows and lines to the figure would not have been beneficial for the purpose of the figure as illustration supporting the text in this Guidance.

Figure 1: Scheme of means, events and plant conditions



PIE = Postulated Initiating Event
PMF = Postulated Multiple Failure

Annex: Non-Exhaustive List of Initial and Consequential Events for the Design Basis

This listing is relevant for Issue E (Design Basis Envelope for Existing Reactors). It is included in the Guidance for Issue F since it is considered useful to provide an overall picture of the foundation for anticipated operational occurrences, design basis accidents and design extension conditions in this Guidance. In particular, DBAs and DECs of category A are connected and should be seen as complementary.

Like the list for DEC A, this list mainly applies to PWR and BWR. For other designs used in WENRA countries (AGR and PHWR), the list will need to be adapted to the reactor type and justified to the regulatory authority of the relevant country.

As in the case of the listing for DEC A, an adequate justification should be provided if items from this list were not included in the corresponding analyses, and a plant and site specific adjustment and justification will be necessary to demonstrate that a comprehensive list of anticipated operational occurrences and design basis accidents has been compiled.

Events for design basis (anticipated operational occurrences and design basis accidents)

Initiating events

- initiating events induced by earthquake, flood or other natural hazard (see Issue T)
- initiating events induced by aircraft crash, other nearby transportation, industrial activities and site area conditions which reasonably can cause fires, explosions or other threats to the safety of the nuclear power plant, or other human made hazards
- small, medium and large LOCA (up to break of the largest diameter piping of the Reactor Coolant Pressure Boundary)
- breaks in the main steam and main feed water systems
- forced decrease of reactor coolant flow
- forced increase of reactor coolant flow (BWR)
- forced increase or decrease of main feed water flow
- forced increase or decrease of main steam flow
- inadvertent opening of valves at the pressurizer (PWR)
- inadvertent operation of the emergency core cooling system
- inadvertent opening of valves at the steam generators (PWR)
- inadvertent opening of main steam relief/safety valves (BWR)
- inadvertent closure of main steam isolation valves
- steam generator tube rupture (PWR, PHWR)

- inadvertent turbine trip (due to loss of main heat sink, loss of external load etc.)
- uncontrolled movement of control rods
- uncontrolled withdrawal/ejection of control rod
- boron dilution in the reactor coolant system or spent fuel pool (PWR)
- core instability (BWR)
- chemical and volume control system malfunction (PWR)
- pipe breaks or heat exchanger tube leaks in systems connected to the reactor coolant system and located partially outside containment (Interfacing System LOCA)
- fuel handling accidents
- loss of off-site power
- load drop by failure of lifting devices

<u>Initiating events as well as consequential events (could be both types) resulting from internal hazards</u>

- fire
- explosion
- flooding

<u>Consequential events</u>

- missile generation, including turbine missiles
- release of fluid (oil etc.) from failed systems
- vibration
- pipe whip
- jet impact

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Report

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Nuclear Regulators Association

Report
Safety of new NPP designs

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Study by Reactor Harmonization Working Group RHWG
March 2013

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01
Introduction
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One of the objectives of the Western European Nuclear Regulators' Association (WENRA), as stated in its terms of reference, is to develop a harmonized approach to nuclear safety and radiation protection and their regulation.

A significant contribution to this objective was the publication, in 2006[1], of a report on harmonization of reactor safety in WENRA countries. This report addresses the nuclear power plants (NPPs) that were in operation at that time in those countries; it includes about 300 "Reference Levels"[2].

Since then, the construction of new nuclear power plants has begun or is being envisaged in several European countries. Hence, it was considered timely for WENRA to develop the safety objectives for new nuclear power plants. A report "Safety objectives for new power reactors – study by RHWG – December 2009" and "WENRA statement on safety objectives for new nuclear power plants – November 2010" have been published by WENRA (www.wenra.org). The statement includes seven safety objectives, which are the basis for further harmonization work of WENRA. Based on these safety objectives, WENRA decided to develop common positions on selected key safety issues for the design of new nuclear power plants.

This report sets out the common positions established by the Reactor Harmonization Working Group (RHWG) of WENRA on the selected key safety issues. The work was initiated and also a major part of the work was carried out before the TEPCO Fukushima Dai-ichi accident. Therefore, the report discusses also some considerations based on the major lessons from the Fukushima Dai-ichi accident, especially concerning the design of new nuclear power plants, and how they are covered in the new reactor safety objectives and the common positions.

Within the WENRA Safety Objectives for New Nuclear Power Plants the words "reasonably practicable" or "reasonably achievable" are used. In this report the words Reasonably Practicable are used in terms of reducing risk as low as *reasonably practicable* or improving safety as far as *reasonably practicable*. The concept of *reasonable practicability* is directly analogous to the ALARA principle applied in radiological protection, but it is broader in that it applies to all aspects of nuclear safety. In many cases adopting practices recognized as good practices in the nuclear field will be sufficient to show achievement of what is "reasonably practicable".

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1 Harmonization of Reactor Safety in WENRA countries, report by RHWG, January 2006

2 These "Reference Levels" were updated in January 2008

For some design expectations in this report, ”reasonable practicability” should be taken to mean that, in addition to meeting the normal requirements of good practice in engineering, further safety or risk reduction measures for the design or operation of the facility should be sought and that these measures should be implemented unless the utility is able to demonstrate that the efforts to implement the proposed measures are grossly disproportionate to the safety benefit they would confer.

This study presents WENRA safety expectations for the design of new NPPs. These expectations are defined in addition to the recent design requirements presented in international texts such as the ones presented in IAEA SSR-2/1 which also covers other fields to ensure safety at the design stage[3].

[3] As stated in IAEA SSR-2/1, the safety of a nuclear power plant is ensured by means of proper site selection, design, construction and commissioning, and the evaluation of these, followed by proper management, operation and maintenance of the plant. In a later phase, appropriate transition to decommissioning is required.

02

WENRA safety objectives for new nuclear power plants

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The WENRA safety objectives for new nuclear power plants were developed on the basis of a systematic review of the Fundamental Safety Principles (SF-1 document issued 2006 by the IAEA). Grounding the safety objectives on the fundamental safety principles has been explained in the December 2009 study by the RHWG. The WENRA Objectives O1-O7 cover the following areas:

O1. Normal operation, abnormal events and prevention of accidents

O2. Accidents without core melt

O3. Accidents with core melt

O4. Independence between all levels of Defence-in-Depth

O5. Safety and security interfaces

O6. Radiation protection and waste management

O7. Leadership and management for safety

The safety objectives address new civil nuclear power plant projects. However, these objectives should also be used as a reference to help identify reasonably practicable safety improvements for “deferred plants” and existing plants during Periodic Safety Reviews.

The safety objectives are formulated in a qualitative manner to drive design enhancements for new plants with the aim of obtaining a higher safety level than that expected from existing plants. For instance, to be able to comply with the qualitative criteria proposed in Objective O3 “Accidents with core melt”, confinement features should be designed to cope with core melt accidents, even in the long term.

The WENRA safety objectives call for an extension of the safety demonstration for new plants, consistent with reinforcement of Defence-in-Depth. Some situations that are considered as “beyond design” for existing plants, such as multiple failures conditions and core melt accidents, are taken into account in the design of new plants.

WENRA considers that these safety objectives reflect the current state of the art in nuclear safety and can be implemented at the design stage using the latest available industrial technology of nuclear power plants. However, since nuclear safety and what is considered adequate protection can never be static, these safety objectives may be subject to further evolution reflecting the need to strive for continuous improvement.

WENRA expects new nuclear power plants to be designed, sited, constructed, commissioned and operated in line with these objectives.

The WENRA statement on safety objectives for new nuclear power plants is included in Annex 1.

03
Selected key safety issues

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The WENRA safety objectives are by nature high level and even when the WENRA statement was published in November 2010 it was recognized that supplementing them with some more detailed common positions on selected issues would help to clarify the meaning. The safety issues where common positions have been developed were chosen on the basis that they were particularly relevant to the expectations for new reactors in comparison with existing reactors. The topics were selected so that they would be relevant for the design of new reactors, constitute an entity and also to make it possible to complete the work by the end of 2012, taking into account the resources of the RHWG.

Objective O4 "Independence between all levels of Defence-in-Depth" seeks enhancement of the effectiveness of the independence between all levels, to provide as far as reasonably practicable an overall reinforcement of Defence-in-Depth. **Position 1** presents WENRA's Defence-in-Depth approach, describing WENRA's expectation that multiple failure events and core melt accidents should be considered in the design of new nuclear power plants. **Position 2** presents the expectations on the independence between different levels of Defence-in-Depth. **Position 3** describes methodology for identification of multiple failure events that should be considered in the design, the related design expectations and the associated safety demonstration.

Objective O4 also mentions strengthening of each Defence-in-Depth level separately. This is achieved by the application of redundancy, diversity and separation principles within one level of Defence-in-Depth. According to safety objective O2 "Accidents without core melt", the core damage frequency should be reduced as far as reasonably achievable, taking into account all types of credible hazards and failures and credible combinations of failures.

Objective O3 "Accidents with core melt" requires that for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public and that sufficient time is available to implement these measures. **Position 4** presents the design provisions to deal with core melt accidents and an interpretation of what limited protective measures could mean in practice.

Objective O3 states also that accidents with core melt which would lead to early or large releases have to be practically eliminated. **Position 5** presents a discussion on means for practical elimination, gives examples of typical LWR accident sequences that could be considered for practical elimination and expectations for the safety demonstration.

Objective O2 “Accidents without core melt” requires providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts. **Position 6** describes the expectations for how external hazards should be considered in the design of new NPPs and **Position 7** deals with design expectations concerning an intentional crash of a commercial aircraft on a NPP. Airplane crash is an example of the safety and security interface, which is discussed in Objective O5 “Safety and security interfaces”.

03.1 Position 1: Defence-in-Depth approach for new nuclear power plants

Introduction

The primary means of preventing accidents in a nuclear power plant and mitigating the consequences of accidents is the application of the concept of Defence-in-Depth (DiD)[4]. This concept should be applied to all safety related activities, whether organizational, behavioural or design related, and whether in full power, low power or various shutdown states. This is to ensure that all safety related activities are subject to independent layers of provisions, so that if a failure were to occur, it would compensated for or corrected by appropriate measures. Application of the concept of Defence-in-Depth throughout design and operation provides protection against anticipated operational occurrences and accidents, including those resulting from equipment failure or human induced events within the plant, and against consequences of events that originate outside the plant.

Therefore, Defence-in-Depth is a key concept of the safety objectives established by WENRA for new nuclear power plants. In particular, these safety objectives call for an extension of the safety demonstration for new plants, in consistence with the reinforcement of the Defence-in-Depth approach. Thus the DiD concept should be strengthened in all its relevant principles. In addition to the reinforcement of each level of the DiD concept and the improvement of the independence between the levels of DiD (as stated in the WENRA safety objectives), this also means that the principle of multiple and independent barriers should be applied for each significant source of radioactive material. It shall also be ensured that the DiD capabilities intended in the design are reflected in the as-built and as-operated plant and are maintained throughout the plant life time.

Some situations that are considered as “beyond design” for existing plants, such as multiple failure events and core melt accidents, are considered in the design of new plants. As a consequence, it has been considered useful to refine this approach which remains consistent with the IAEA SF-1 document.

This section focuses primarily on the proposal to refine the structure of the DiD levels. Other DiD related topics, i.e. the “Independence of Defence-in-Depth levels”, “Multiple failure events” and “Provisions to mitigate core melt and radiological consequences” are addressed in separate sections.

Historical development of the Defence-in-Depth as regards currently operating reactors

The concept of “Defence-in-Depth” has been introduced in the field of nuclear safety in the early 1970s. This concept was gradually refined to constitute an increasingly effective approach combining both prevention of a wide range of postulated incidents and accidents and mitigation of their consequences. Incidents and accidents were postulated on the basis of single initiating events selected according to the order of magnitude of their frequency, estimated from general industrial experience.

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[4] According to the IAEA safety glossary, this concept is depicted as a hierarchical deployment of different levels of divers equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive materials and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.

The definitions of the different levels of DiD were set as to mirror escalation from normal operation to accident so that if one level fails, a higher level comes into force. This does not mean that the situations considered in one level are systematically resulting from a failure of systems/features associated to the previous level of defence. The different levels of DiD were set as to cover the different situations that need to be considered in the design and operation of the plant. The approach was intended to provide robust means to ensure the fulfilment of each of the fundamental safety functions[5] of:

(1) Control of reactivity;
(2) Removal of heat from the reactor and from the fuel store;
(3) Confinement of radioactive material, shielding against radiation, as well as limitation of accidental radioactive releases.

In the early stage, the concept of Defence-in-Depth included three levels:

Levels of defence in depth	Objective	Essential means	Associated plant condition categories (for explanation - not part of original table)
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation	Normal operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences
Level 3	Control of accident within the design basis	Engineered safety features and accident procedures	Design basis accidents (postulated single initiating events)

[5] IAEA SSR-2/1

Then, the concept of Defence-in-Depth for the current operating reactors was further developed to take into account severe plant conditions that were not explicitly addressed in the original design (hence called "beyond design conditions"), in particular lessons learned from the development of probabilistic safety assessment and from the Three Mile Island accident (USA 1979) which led to a severe core melt accident and from the Chernobyl accident (Ukrainian Republic of USSR 1986). These developments led to two additional levels in DiD (see INSAG 10 – 1996):

Levels of defence in depth	Objective	Essential means	Associated plant condition categories (for explanation - not part of original table)
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation	Normal operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences
Level 3	Control of accident within the design basis	Engineered safety features and accident procedures	Design basis accidents (postulated single initiating events)
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management	Multiple failures Severe accidents
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response	

New reactor designs and associated evolution of the Defence-in-Depth levels

Rationale for an evolution of DiD levels

For new reactor designs, there is a clear expectation to address in the original design what was often “beyond design” for the previous generation of reactors, such as multiple failure events and core melt accidents, called Design Extension Conditions in IAEA SSR-2/1. This is a major evolution in the range of situations considered in the initial design to prevent accidents, control them and mitigate their consequences, and in the corresponding design features of the plant. It implies that the meaning of “beyond design basis accident” is not the same for existing reactors and for new reactors. Several scenarios that are considered beyond design basis for most existing reactors are now included from the beginning in the design for new reactors (postulated multiple failure events and core melt accidents).

In the DiD approach, the objectives of the different levels of defence are mainly defined as successive steps in the protection against the escalation of accident situations.

The phenomena involved in accidents with core/fuel melt (severe accidents) differ radically from those which do not involve a core melt. Therefore core melt accidents should be treated on a specific level of Defence-in-Depth.

In addition, for new reactors, design features that aim at preventing a core melt condition and that are credited in the safety demonstration should not belong to the same level of defence as the design features that aim at controlling a core melt accident that was not prevented. However, should a core melt accident occur, all plant equipment still available may be used.

The question has been discussed by RHWG whether for multiple failure events, a new level of defence should be defined, because safety systems which are needed to control postulated single initiating events are postulated to fail and thus another level of defence should take over. However, the single initiating events and multiple failure events are two complementary approaches that share the same objective: controlling accidents to prevent their escalation to core melt conditions.

Hence, at this stage of the discussion, it has been proposed to treat the multiple failure events as part of the 3rd level of DiD, but with a clear distinction between means and conditions (sub-levels 3.a and 3.b).

The scope of the related safety demonstration has to cover all risks induced by the nuclear fuel, including all fuel storage locations, as well as the risks induced by other relevant radioactive materials.

Refined structure of the levels of DiD

The refined structure of the levels of DiD proposed by RHWG is as follows:

<table>
<tr><th colspan="2">Levels of defence in depth</th><th>Objective</th><th>Essential means</th><th>Radiological consequences</th><th>Associated plant condition categories</th></tr>
<tr><td colspan="2">Level 1</td><td>Prevention of abnormal operation and failures</td><td>Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits</td><td rowspan="2">No off-site radiological impact (bounded by regulatory operating limits for discharge)</td><td>Normal operation</td></tr>
<tr><td colspan="2">Level 2</td><td>Control of abnormal operation and failures</td><td>Control and limiting systems and other surveillance features</td><td>Anticipated operational occurrences</td></tr>
<tr><td rowspan="2">Level 3 [1]</td><td>3.a</td><td rowspan="2">Control of accident to limit radiological releases and prevent escalation to core melt conditions [2]</td><td>Reactor protection system, safety systems, accident procedures</td><td rowspan="2">No off-site radiological impact or only minor radiological impact [4]</td><td>Postulated single initiating events</td></tr>
<tr><td>3.b</td><td>Additional safety features[3], accident procedures</td><td>Postulated multiple failure events</td></tr>
<tr><td colspan="2">Level 4</td><td>Control of accidents with core melt to limit off-site releases</td><td>Complementary safety features[3] to mitigate core melt, Management of accidents with core melt (severe accidents)</td><td>Off-site radiological impact may imply limited protective measures in area and time</td><td>Postulated core melt accidents (short and long term)</td></tr>
<tr><td colspan="2">Level 5</td><td>Mitigation of radiological consequences of significant releases of radioactive material</td><td>Off-site emergency response

Intervention levels</td><td>Off site radiological impact necessitating protective measures[5]</td><td>-</td></tr>
</table>

[1] Even though no new safety level of defence is suggested, a clear distinction between means and conditions for sub-levels 3.a and 3.b is lined out. The postulated multiple failure events are considered as a part of the Design Extension Conditions in IAEA SSR-2/1.

[2] Associated plant conditions being now considered at DiD level 3 are broader than those for existing reactors as they now include some of the accidents that were previously considered as "beyond design" (level 3.b). For level 3.b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic insights. Best estimate methodology and less stringent rules than for level 3.a may be applied if ap-

propriately justified. However the maximum tolerable radiological consequences for multiple failure events (level 3.b) and for postulated single failure events (level 3.a) are bounded by Objective O2.

(3) The task and scope of the additional safety features of level 3.b are to control postulated common cause failure events as outlined in Section 3.3 on “Multiple failure events”. An example for an additional safety feature is the additional emergency AC power supply equipment needed for the postulated common cause failure of the primary (non-diverse) emergency AC power sources.

The task and scope of the complementary safety features of level 4 are outlined in Section 3.4 on “Provisions to mitigate core melt and radiological consequences”. An example for a complementary safety feature is the equipment needed to prevent the damage of the containment due to combustion of hydrogen released during the core melt accident.

(4) It should be noted that the tolerated consequences of Level 3.b differ from the requirements concerning Design Extension Conditions in IAEA SSR-2/1 that gives a common requirement for DEC: “for design extension conditions that cannot be practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary”.

(5) Level 5 of DiD is used for emergency preparedness planning purposes.

In each level of DiD, some situations need to be practically eliminated as it cannot be demonstrated that, should they occur, their radiological consequences would be tolerable. Situations that could lead to early or large releases of radioactive materials have to be practically eliminated (see Section 3.5 on “Practical elimination”).

03.2 Position 2: Independence of the levels of Defence-in-Depth

Introduction

According to the 2010 WENRA "Statement on safety objectives for new nuclear power plants" WENRA expects new nuclear power plants to be designed, sited, constructed, commissioned and operated with the objective, among others, of "*enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately…), to provide as far as reasonably achievable, an overall reinforcement of defence-in-depth*." (Objective O4: "Independence between all levels of defence-in-depth").

This section focuses on the independence between systems, structures and components (SSCs) important to safety, allocated to different levels of Defence-in-Depth (DiD). It does not aim to address independence between SSCs important to safety within a level of defence-in-depth nor administrative/procedural aspects.

Furthermore, this section solely addresses those SSCs which are necessary to meet the acceptance criteria, related to the three fundamental safety functions, and the radiological goals defined at the different DiD levels according to WENRA safety objectives.

Definitions of key terms used in this section are given in the end. The levels of DiD which are referred to in this section are defined in Section 3.1 on Defence-in-Depth.

This section aims to give some guidance on how to enhance the effectiveness of the independence between the levels of DiD.

Independence between systems, structures and components (SSCs)

WENRA considers that independent SSCs for safety functions on different DiD levels shall possess both of the following characteristics:

- the ability to perform the required safety functions is unaffected by the operation or failure of other SSCs needed on other DiD levels;
- the ability to perform the required safety functions is unaffected by the occurrence of the effects resulting from the postulated initiating event, including internal and external hazards, for which they are required to function.[6]

As a consequence, the means to achieve independence between SSCs are adequate application of:

- diversity;
- physical separation, structural or by distance;
- functional isolation.

The following expectations on independence are related to the independence between SSCs as credited in the deterministic safety demonstration. If an accident was to occur, all available and effective equipment could obviously be used, including those not credited in the safety demonstration.

[6] Based on the IAEA safety glossary.

Basic safety expectations on the independence between different levels of DiD

(1) There shall be independence to the extent reasonably practicable between different levels of DiD so that failure of one level of DiD does not impair the defence in depth ensured by the other levels[7] involved in the protection against or mitigation of the event.

(2) The adequacy of the achieved independence shall be justified by an appropriate combination of deterministic and probabilistic safety analysis and engineering judgement.
For each postulated initiating event (starting with DiD level 2), the necessary SSCs should be identified and it shall be shown in the safety analysis that the SSCs credited in one level of DiD are adequately independent of SSCs credited in the other levels of DiD.[8]

(3) Appropriate attention shall be paid to the design of I&C, the reactor auxiliary and support systems (e. g. electrical power supply, cooling systems) and other potential cross cutting systems. The design of these systems shall be such as not to unduly compromise the independence of the SSCs they actuate, support or interact with.

Implementation of the basic safety expectations

In applying the above basic expectations, the following considerations shall be taken into account (some specific considerations are presented in the next section):

(1) SSCs fulfilling safety functions in case of postulated single initiating events (DiD level 3.a) or in postulated multiple failure events (DiD level 3.b) should be independent to the extent reasonably practicable from SSCs used in normal operation (level 1) and/or in anticipated operational occurrences (level 2). This independence is so that the failure of SSCs used in normal operation and/or in anticipated operational occurrences does not impair a safety function required in the situation of a postulated single initiating event or of a multiple failure event resulting from the escalation of such failures during normal operation or a level 2 event.

(2) SSCs fulfilling safety functions used in case of postulated single initiating events (DiD level 3.a) should be independent to the extent reasonably practicable from additional safety features used in case of postulated multiple failure events (DiD level 3.b). For the safety analyses of postulated multiple failure events, credit may be taken from SSCs used in case of postulated single initiating events as far as these SSCs are not postulated as unavailable and are not affected by the multiple failure event in question; SSCs specifically designed for fulfilling safety functions used in postulated multiple failure events should not be credited for level 3.a event analyses for the same scenario.

(3) Complementary safety features specifically designed for fulfilling safety functions required in postulated core melt accidents (DiD level 4) should be independent to the extent reasonably practicable from the SSCs of the other levels of DiD.

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[7] This should cover all plant states of the nuclear power plant.

[8] For future development designs a more systematic allocation of each SSC to one particular level of DiD, irrespective of the postulated initiating event, may provide a more robust demonstration of the independence between levels of DiD.

Specific considerations (examples on specific topics)

Emergency AC power supply

The emergency AC power supply belonging to DiD level 3.a may be used also in DiD level 2. An additional diverse emergency AC power supply shall be designed for DiD level 3.b because the common cause failure of the primary (non-diverse) emergency AC power sources is postulated. The emergency power supply on DiD Level 3.b may be also used for DiD level 4. The rationale for this is that additional independent on-site provisions are not likely to significantly increase the reliability of the emergency AC power supply. Lessons learnt from the Fukushima Dai-ichi accidents with regard to the supply of additional AC power supply provisions are addressed separately.

Separation of cables

Since principles of separation of cables already exist between the divisions of redundant systems and between safety and non-safety systems, it may not be reasonably practicable to introduce additional separation on the basis of levels of defence.

Reactor protection system (RPS) and other I&C aspects

The reactor protection system (RPS) shall be adequately independent from other I&C systems and must be functionally isolated from them. The RPS may have I&C functions on other DiD levels than 3, e.g. the scram system may be actuated by the RPS for specific DiD level 2 events. Diverse I&C means shall be designed for DiD level 3.b in case the common cause failure of the RPS has to be postulated.

Limitation and control systems (not the RPS) for the actuation of systems needed to handle level 2 events may be combined with I&C for normal operation.

Containment

On each level of defence there is a need for confinement as a safety function. This safety function may be accomplished for example by the use of the containment in combination with other SSCs. The containment is thus an example of a structure which is used on different levels of defence and for which it would not be reasonably practicable to require independence for different levels of Defence-in-Depth.

Reactor pressure vessel

The reactor pressure vessel in combination with other SSCs may be used to fulfil/accomplish several safety functions on several levels of DiD. For example, on DiD level 1 and 2 this may include removal and transfer of thermal energy from nuclear fuel during normal and abnormal operation. On DiD level 1, 2, 3 and 4 this may include the removal of residual heat from nuclear fuel to the ultimate heat sink and on level 1, 2, 3 and 4 this may also include the prevention of the dispersal of radioactive material. It would not be reasonably practicable to require independence for these different levels.

The list of specific considerations/examples shall give guidance on the implementation of the basic safety expectations and thus is not exhaustive.

Definitions

Functional isolation:

Prevention of influences from the mode of operation or failure of one circuit or system on another.[9] Functional isolation shall refer to the isolation of inter-connected systems and sub-systems from one another so as to prevent propagation of failure or spurious signals from one system to another and it also includes electrical isolation and information flow isolation.

Fundamental safety function:
A safety function is a specific purpose that must be accomplished for safety. In a nuclear power plant there exist the following three fundamental safety functions (from IAEA SSR-2/1):

(1) Control of reactivity;
(2) Removal of heat from the reactor and from the fuel store;
(3) Confinement of radioactive material, shielding against radiation, as well as limitation of accidental radioactive releases.

Independence between systems, structures and components:
Independent systems, structures and components (SSCs) for safety functions on different DiD levels shall possess both of the following characteristics:

- the ability to perform the required safety functions is unaffected by the operation or failure of other SSCs needed on other DiD levels;
- the ability to perform the required safety functions is unaffected by the occurrence of the effects resulting from the postulated initiating event, including internal and external hazards, for which they are required to function. [9]

Means to achieve independence between SSCs are adequate application of:
- physical separation, structural or by distance;
- functional isolation;
- diversity.

Reactor protection system:
System that monitors and processes the variables relevant for safety and which, on reaching pre-set actuation limits, automatically initiates the necessary actions of safety systems for the control of DiD level 3 events, in order to prevent an unsafe or potentially unsafe condition. The reactor protection system encompasses all electrical and mechanical devices and circuitry, from sensors to actuation device input terminals. [9]

Systems, structures and components important to safety (SSCs):
A general term encompassing all the plant elements (items) of a facility or activity which contribute to protection and safety, except human factors.

- Structures are the passive elements: buildings, vessels, shielding, etc..
- A system comprises several components and/or structures, assembled in such a way as to perform a specific (active) function.
- A component is a discrete element of a system.
  Examples of components are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves.

[9] Based on the IAEA safety glossary.

03.3 Position 3: Multiple failure Events

Introduction

Defence in depth (DiD) is a key element of the safety objectives established by WENRA for new nuclear power plants. In particular, these safety objectives call for an extension of the safety demonstration for new plants, in consistence with the reinforcement of the defence in depth. Some situations that are considered as “beyond design” for existing plants, such as e.g. multiple failure events, are to be considered in the design of new plants. As a consequence, it has been considered useful to refine this approach whilst remaining consistent with the IAEA SF-1 document (cf. with Section 3.1 on “Defence in depth approach for new nuclear power plants”).

In this refined DiD concept for new reactors level of defence 3 consists of level 3.a and level 3.b. Both levels aim to “control of accidents to limit radiological releases and prevent escalation to core melt conditions”. Level 3.a includes “Postulated single initiating events” and level 3.b includes “Selected multiple failure events including possible failure or inefficiency of safety systems involved in level 3.a”.[10]

Level 3.b is related to Objective O2, “Accidents without core melt”. According to Objective O2 it shall be ensured that accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering or evacuation). Design provisions considered in level 3.b for postulated multiple failures shall further decrease the frequency and/or mitigate consequences of sequences beyond those considered in the design basis for existing reactors so far, such as anticipated transients without scram (ATWS) or station black out (SBO) scenarios.

Scope

In a general sense, failure of safety or safety related system at a NPP may arise for different reasons. These failures could result due to

i) a single Postulated Initiating Event (PIE) with consequential failures;

ii) an external or internal hazard (e.g. earthquake, flooding, fire) affecting one or several safety (or safety related) systems;

iii) common cause failure for other reasons than a postulated hazard, affecting similar equipment in
   a. the same safety (or safety related) system, or
   b. several safety (or safety related) systems

iv) random failures that affect simultaneously several safety (or safety related) systems.

Failures resulting from a PIE (i) or a postulated hazard (ii) are part of the considered event and studied with the corresponding rules. This section deals with multiple failures resulting from common cause failures, affecting the same safety or safety related system (iii.a). Other common cause failures affecting different safety (or safety related) systems are not postulated.

[10] Level 3.b events are considered as a part of the Design Extension Conditions in IAEA SSR 2.1.

There should be other design provisions to prevent such failure modes. Combination of random failures that affect simultaneously several safety (or safety related) systems (iv) are not postulated deterministically from this approach, and should be considered in PSA.

Multiple failure events to be considered at the design stage are characterized as:

- a postulated common cause failure or inefficiency of all redundant trains of a safety system11 needed to fulfil a safety function necessary to cope with an anticipated operational occurrences (AOO) or a single PIE (see examples in Table 1), or
- a postulated common cause failure of a safety system or a safety related system needed to fulfil the fundamental safety functions in normal operation (see examples in Table 2).

Methodology of identification of multiple failure events

The identification of multiple failure events should start with a systematic deterministic approach based on a list of anticipated operational occurrences and postulated single initiating events.[12]

Safety (or safety related) systems to fulfil the related safety functions for these AOO and PIE have to be identified. Based on this a list of multiple failure events should be developed. Furthermore, a list of common cause failures of safety systems or safety related systems needed to fulfil the fundamental safety functions in normal operation should be compiled. This process is supported by PSA.

As a result an intermediate list should include:

- AOOs and a postulated common cause failure of redundant trains of a safety system;
- Single PIEs and a postulated common cause failure of redundant trains of a safety system;
- Complex or specific scenarios including common cause failures of safety systems or safety related systems needed to fulfil the fundamental safety functions in normal operation

The identification procedure shall be performed for any operational state and should include failures of spent fuel pool cooling.

Based on this, a selection of a reasonable number of limiting (bounding) cases, which present the greatest challenge to the acceptance criteria and which define the performance parame-

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[11] IAEA safety glossary: ***safety system.*** A *system* important to *safety*, provided to ensure the safe shutdown of the reactor or the *residual heat* removal from the core, or to limit the consequences of *anticipated operational occurrences* and *design basis accidents*.

Safety systems consist of the *protection system*, the *safety actuation systems* and the *safety system support features*. *Components* of *safety systems* may be provided solely to perform *safety functions*, or may perform *safety functions* in some plant *operational states* and non-safety functions in other *operational states*.

[12] The approach may start at the beginning of the design with a reduced list based on engineering judgment and should be completed stepwise in parallel to the developing design approach.

ters for safety related equipment, should be made using experience feedback, engineering judgment and probabilistic assessment.

In choosing the multiple failure events to be addressed in the design, the following factors should be considered together:

- the frequency of the event;
- the grace time for necessary human actions;
- the margins to cliff edge effects; and
- the radiological or environmental consequences of the event (care should be taken to scenarios with containment bypass).

Any general cut-off frequency should be justified, considering in particular the overall core damage frequency (CDF) aimed at.

The identification process should lead to a list of postulated multiple failure events which have to be considered in the design.

Design expectations

While the postulated single initiating events analyses in combination with the single failure criteria usually gives credit on redundancy in design provisions of safety systems and of their support functions, addressing multiple failure events emphasizes diversity in the design provisions of the third level of DiD.

Safety assessments of the plant conditions resulting from the multiple failures selected by deriving them from the defined methodology shall be performed deterministically in order to design additional safety features that aim at preventing core damage conditions. “Accident procedures” shall be in place to define the management of the safety features and to give guidance on necessary human actions. The appropriateness of the foreseen additional design features has to be assessed by PSA modelling and results.

The expectations for the additional safety features and the associated systems which are foreseen to cope with such conditions on the level 3.b of the DiD concept do not have to be as stringent as for 3.a if appropriately justified. This justification may be based on probabilistic arguments, complemented by additional factors similar to those in the previous section. Systems designed to comply with these conditions should have sufficient redundancy of active components to reach adequate reliability.

According to Section 3.2 on the “Independence of Defence-in-Depth Levels”, systems, structures and components (SSCs) fulfilling safety functions used in case of postulated single initiating events (DiD level 3.a) should be independent to the extent reasonably practicable from additional safety features used in case of postulated multiple failure events (DiD level 3.b). For the safety analyses of postulated multiple failure events, credit may be taken from SSCs used in case of postulated single initiating events as far as these SSCs are not postulated as unavailable and are not affected by the multiple failure event in question. SSCs specifically designed for fulfilling safety functions used in postulated multiple failure events should not be credited for level 3.a event analyses for the same scenario.

Safety demonstration

For the additional safety features on level 3.b of the DiD concept it shall be shown that under the assumption of the postulated multiple failures first a controlled state[13] and later on a safe state14 is reached and the radiological criteria of O2 “No off-site radiological impact or only minor radiological impact” will be fulfilled analogue to the requirement on level 3.a.

Once a controlled state is reached emphasis shall be paid to achieve a safe state in which the fundamental safety functions can be ensured and stably maintained for long time.

For the technical safety demonstration, acceptance criteria should be:

- reaching core sub-criticality quickly and maintaining it after;
- no or only limited fuel damage and ensuring of a coolable core geometry;
- prevention of energetic dispersal of fuel;
- limiting the pressure in the reactor coolant pressure boundary below a justified value;
- maintaining the fuel in the spent fuel pool covered with coolant with sufficient margin and ensuring that potential boiling conditions will not preclude potential necessarily access by personnel to perform accident procedures.

For level 3.b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic insights. Best estimate methodology and less stringent rules than for level 3.a may be applied if appropriately justified. However the maximum tolerable radiological consequences for multiple failure events (level 3.b) and for postulated single failure events (level 3.a) are bounded by Objective O2.

Examples of multiple failure scenarios

Some examples of multiple failure scenarios are given below. Plant specific lists of multiple failure scenarios may include these examples but probably will not be limited to it. The examples are:

[13] IAEA SSR-2.1: Plant state, following an *anticipated operational occurrence* or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to implement provisions to reach a *safe state*.

[14] IAEA SSR-2.1: Plant state, following an *anticipated operational occurrence* or accident conditions, in which the reactor is subcritical and the fundamental safety functions can be ensured and stably maintained for long time.

Table 1. Examples of postulated common cause failures of safety systems needed to fulfil a safety function necessary to cope with an AOO or a single PIE.

Denotation	Postulated Initiating Event	Loss of a safety system
LOCA	Small LOCA	Medium head safety injection
	Small LOCA	Low head safety injection
Station blackout	Loss of off-site power	Emergency power supply
Total loss of feed water	Loss of main feed water	Emergency feed water supply
ATWS	Anticipated Transient	Fast shutdown

Table 2. Examples of postulated common cause failures of safety systems needed to fulfil the fundamental safety functions in normal operation

Denotation	Initiating condition	Loss of a system
Loss of RHR	*normal operation*	Residual heat removal
Loss of UHS	*normal operation*	Ultimate heat sink
Loss of CCW/ECW	*normal operation*	Component cooling water / essential cooling water
Loss of spent fuel pool cooling	*normal operation*	Spent fuel pool cooling

03.4 Position 4: Provisions to mitigate core melt and radiological consequences

Introduction

WENRA has issued safety objectives for new reactors including Objective O3 “Accidents with core melt”:

> *reducing potential radioactive releases to the environment from accidents with core melt*[15]*, also in the long term*[16]*, by following the qualitative criteria below:*
>
> - *accidents with core melt which would lead to early*[17] *or large*[18] *releases have to be practically eliminated*[19;]
> - *for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.*

Design provisions to deal with accidents with core melt

The goal behind Objective O3 is that the nuclear power plants have to be designed in such a way that even in case of an accident with core melt *only limited protective measures in area and time are needed for the public and that sufficient time is available to implement these measures*. Any reasonably achievable solution which would further reduce the radiation doses of workers or the population or environmental consequences should be implemented.

In such an accident, the reactor containment structure is the main barrier for protecting the environment from the radioactive materials. Thus, it is essential to maintain the integrity of this barrier throughout the course of such an accident. In addition to the containment structure there have to be complementary safety features included in the design of the plant and procedures implemented to mitigate the consequences of core melt accidents. Consequently, the containment and the core melt management systems have to be designed to comply with Objective O3 and to keep radioactive releases during the severe accident conditions starting from all operational states as low as reasonably practicable. Any event resulting in a situation where Objective O3 is not fulfilled is considered a failure of the containment function.

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[15] *Core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool. It has to be shown that such accident scenarios are either practically eliminated or prevented and mitigated.*

[16] *Long term: considering the time over which the safety functions need to be maintained. It could be months or years, depending on the accident scenario.* This definition is different from the long term restrictions in food consumption, which is interpreted in the last section of this appendix.

[17] *Early releases: situations that would require off-site emergency measures but with insufficient time to implement them.*

[18] *Large releases: situations that would require protective measures for the public that could not be limited in area or time.*

[19] *In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA SSR 2.1).* Section 3.5 deals with the issue “Practical elimination” in more detail.

Provisions have to be taken to prevent accidents which would require protective actions for the public that could not be considered as limited in area and time (large release) and also to prevent accidents which would require protective actions for the public for which there would not be sufficient time to implement these measures (early release). These provisions have to make such accidents physically impossible to occur or to make it possible to consider with high degree of confidence that they are extremely unlikely to arise. Section 3.5 on “Practical Elimination” discusses this topic including examples of containment bypass and fuel melt sequences in the spent fuel pools.

In order to reliably maintain the containment barrier:

- Complementary safety features (DiD level 4) specifically designed for fulfilling safety functions required in postulated core melt accidents shall be independent to the extent reasonably practicable from the SSCs of the other levels of DiD. Independence of DiD levels is discussed in Section 3.2;
- Complementary safety features specifically designed for fulfilling safety functions required in postulated core melt accidents shall be safety classified and adequately qualified for the core melt accident environmental conditions for the time frame for which they are required to operate;
- The systems and components necessary for ensuring the containment function in a core melt accident shall have reliability commensurate with the function that they are required to fulfil. This may require redundancy of the active parts;
- It shall be possible to reduce containment pressure in a controlled manner in a long term taking into account the impact of non-condensable gases;
- If a containment venting system is included in the design, the safety margins in containment dimensioning shall be such that it should not be needed in the early phases[20] of the core melt accident, to deal with the containment pressure due to the non-condensable gases accumulating in the containment;
- Containment heat removal during core melt accidents shall be ensured. If included in the design, the containment venting system shall not be designed as the principal means of removing the decay heat from the containment;
- The strength of the containment including the access openings, penetrations and isolation valves shall be high enough to withstand, with sufficient margins to consider uncertainties, static and dynamic loads during core melt accidents that have not been practically eliminated (pressure, temperature, radiation, missile impacts, reaction forces). There shall be appropriate provisions to prevent the damage of the containment due to combustion of hydrogen;

In order to reduce the release of radioactive substances:

- there shall be provisions to reduce the amount of fission products in the containment atmosphere in case of the core melt accident;
- there shall be provisions to reduce the pressure inside the containment;

[20] Early phase is considered to last until the amount of radioactive material in the containment atmosphere has decreased significantly.

- if a containment venting system is included in the design to reduce the containment pressure in a core melt accident, it shall have a filtering capability;
- the containment penetrations should be surrounded by secondary structures to collect the potential leakages from the containment.

Any instrumentation required to decide on countermeasures shall be included in the design. This instrumentation shall be safety classified, adequately qualified for environmental conditions and it shall have reliability commensurate with the function that it is required to fulfil.

Analysis methodology

To show that the safety objective is reached, two complementary approaches are needed: deterministic and probabilistic. The following deals with scenarios that are not practically eliminated from the design point of view. The topic of practical elimination is discussed in Section 3.5 in more detail.

Deterministic analyses shall cover core melt scenarios starting from all operational states. Postulated core melt accidents are typically considered with realistic assumptions and best estimate methodologies. Adequate methods have to be utilised in order to show the robustness and reliability of the approach. On-site and off-site radiological consequences shall be analysed using stated and justified assumptions. Any possible influence from and on other nuclear facilities in the vicinity of the plant shall be analysed.

The probabilistic safety assessment (PSA) is complementary to the deterministic analyses. Comprehensive level 2 PSA of sufficient scope shall be carried out to demonstrate that the containment function can be shown to be reliable to meet Objective O3. PSA shall also be used to demonstrate that the selection of accident sequences for deterministic calculations is adequate for the design of severe accident provisions.

Intervention levels

These protective measures of sheltering, iodine prophylaxis, evacuation, and permanent relocation are associated with Generic Intervention Levels, which are used for emergency preparedness planning in the 5th level of the defence in depth. Such intervention levels have already been enforced by EU members in their national regulation to comply with Directive 96/29/Euratom - 13 may 1996 – article 50.2., and are consistent with the ICRP recommendations and IAEA GS-R-2. However, the intervention levels are not fully harmonised between European countries and there are some quantitative differences. Maximum admissible levels are set for food marketing in Europe.

In emergency preparedness planning, certain areas are defined around nuclear power plants for which arrangements are made for taking urgent protective actions of sheltering, evacuation, and iodine prophylaxis in case of an accident. IAEA GS-R-2 (2002) and GS-G-2.1 (2007) documents define the following zones:

(1) A precautionary action zone (PAZ, with the suggested radius of 3–5 km for reactors rated more than 1000 MW_{th}) is an area for which precautionary urgent protective action shall be taken before a release of radioactive material occurs or shortly after a release of radioactive material begins in order to reduce substantially the risk of severe deterministic health effects;
(2) An urgent protective action planning zone (UPZ, with the suggested radius of 5–30 km for reactors rated more than 1000 MW_{th}) is an area for which urgent protective action

shall be taken promptly in order to prevent stochastic effects and avert doses in accordance with international standards.

WENRA interpretation of limited protective measures

To achieve Objective O3, it is expected that the off-site radiological impact of accidents with core melt which are not practically eliminated only leads to limited protective measures in area and time (no permanent relocation, no long term restrictions in food consumption, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering). Iodine prophylaxis is not mentioned in Objective O3 list of protective measures, but it shall also be limited in area and time. Sufficient time shall be available to implement these measures.

For the design stage of a nuclear power plant, to achieve Objective O3 on the 4th level of the defence in depth, the following interpretations of limited protective measures are provided (specified zones are not meant to be used for emergency preparedness planning):

(1) Immediate vicinity of the plant: For new reactors, the design should be such that the possible release of radioactive substances in a postulated core melt accident, based on the analysed consequences of the accident, will not initiate a need for emergency evacuation beyond the immediate vicinity of the plant. The design goal should aim at having a radius of this immediate vicinity zone towards the lower end of the suggested PAZ range i.e. 3 km (evacuation zone).

(2) Limited sheltering and iodine prophylaxis: For new reactors, the design goal should be such that the possible release of radioactive substances in a postulated core melt accident, based on the analysed consequences of the accident, will not initiate a need for sheltering and iodine prophylaxis beyond the zone towards the lower end of the suggested UPZ range i.e. 5 km (sheltering zone).

(3) No long-term restrictions in food consumption: This is interpreted so that after a postulated core melt accident, based on the analysed consequences of the accident, agricultural products beyond the sheltering zone should generally be consumable after the first year following the accident.

(4) Sufficient time: Sufficient time is interpreted so that protective measures should be initiated early enough. Especially the evacuation shall be carried out already when there is a threat of a significant radioactive release into the environment. Sufficient time to implement these protective measures is different for each measure and for each accident scenario and depends on the location of the reactor. Sufficient time for each measure shall be estimated and considered in the design of a reactor and during the site licensing.

Table 3 below summarises the interpretation of limited protective measures of evacuation, sheltering and iodine prophylaxis to be applied as goals in the design phase of new reactors. The zones for emergency preparedness planning, which take into account plant location and population living nearby, are usually larger because they are based on conservative approaches to protect people (for example, it could be assumed that some DiD level 4 provisions could partially fail).

Table 3. Design goals for areas where limited protective measures may be needed.

Measure	Evacuation zone	Sheltering zone	Beyond sheltering zone
Permanent relocation	No	No	No
Evacuation	May be needed	No	No
Sheltering	May be needed	May be needed	No
Iodine Prophylaxis	May be needed	May be needed	No

As for doses to the public or level of contamination to foodstuff, the definition of quantitative values associated to qualitative goals of Objective O3 is difficult since the analysis methodologies of radiological consequences might be based on different national regulations and practices including calculation models and hypothesis.

In addition to the design goals related to limited protective measures, ALARA principle shall be applied and any reasonably achievable measure which would further reduce the radiation doses of workers or the population or environmental consequences should be implemented.

03.5 Position 5: Practical elimination

Introduction

WENRA has issued safety objectives for new reactors including Objective O3 “Accidents with core melt”:

- reducing potential radioactive releases to the environment from accidents with core melt, also in the long term, by following the qualitative criteria below:
    - accidents with core melt which would lead to early or large releases have to be practically eliminated;
    - for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

Here the scope of “core melt” includes the nuclear fuel at fuel storage locations, as described in the WENRA publication on safety objectives: “Core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool. It has to be shown that such accident scenarios are either practically eliminated or prevented and mitigated”. Here, “core melt” also includes severe degradation due to mechanisms other than melting, since radioactive releases can occur without melting (e.g. severe reactivity increase accidents).

Accident sequences that are practically eliminated have a very specific position in the Defence-in-Depth approach because provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design. The justification of the “practical elimination” should be primarily based on design provisions where possible strengthened by operational provisions (e.g. adequately frequent inspections). All accident sequences which may lead to early or large radioactive releases must be practically eliminated.

An early release means a release that would require off-site emergency measures but with insufficient time to implement them. A large release means situations that would require protective measures for the public that could not be limited in area or time.

Means of Practical Elimination

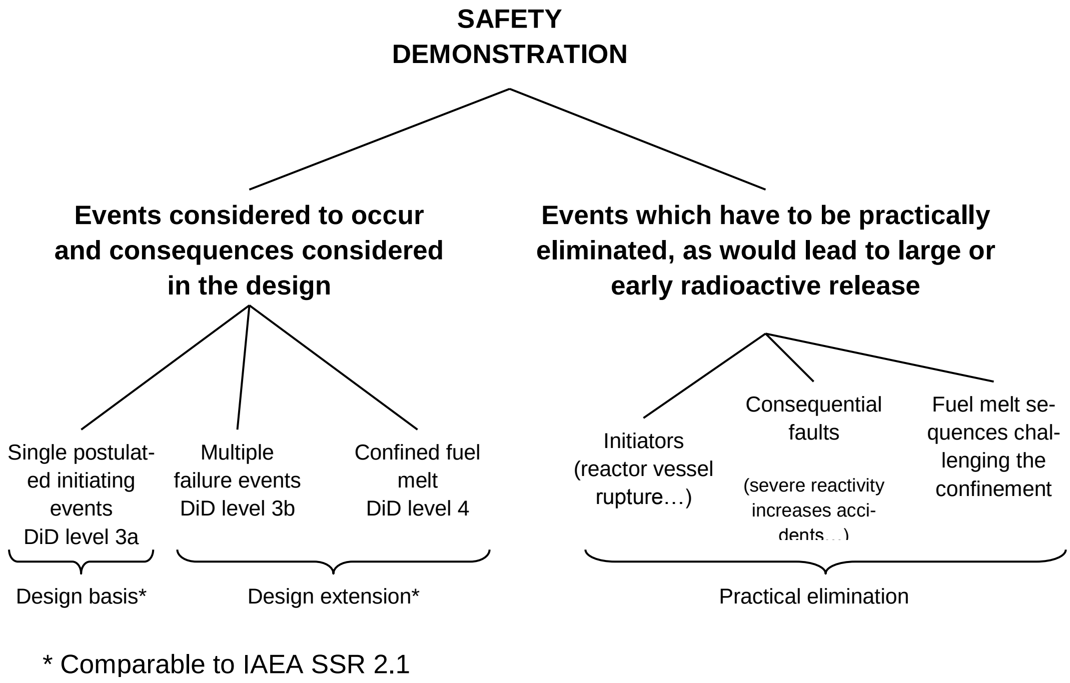
Accident sequences with a large or early release can be considered to have been practically eliminated:

(1) if it is physically impossible for the accident sequence to occur or
(2) if the accident sequence can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA SSR-2/1).

In each case the demonstration should show sufficient knowledge of the accident condition analysed and of the phenomena involved, substantiated by relevant evidence.

To minimize uncertainties and to increase the robustness of a plant's safety case, demonstration of practical elimination should preferably rely on the criterion of physical impossibility, rather than the second criterion (extreme unlikelihood with high confidence).

Accident sequences to be considered for Practical Elimination



Identification of accident sequences that have the potential to cause a large or early release should be based on deterministic analyses, supported by engineering judgment, and probabilistic assessment. These analysis approaches in the safety justification have to be adapted to each particular situation.

Important examples where consideration of severe accidents conditions should be aimed at practically eliminating large or early releases include those:

Unacceptable initiating faults:

- rupture of major pressure retaining components, e.g. reactor vessel.

Unacceptable consequential faults:

- large reactivity insertions directly leading to severe core degradation;
- internal hazard leading to severe core degradation (heavy load drops or internal flooding);
- fuel melt in an unconfined spent fuel pool situation[21].

[21] Also in confined spent fuel pool situations, fuel melt should be practically eliminated unless it can be demonstrated that there will be no large or early releases.

Fuel melt sequences challenging the confinement

- whilst at load that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;
- whilst at load that could damage the containment in a late phase as a result of base mat melt-through or containment over pressurization;
- when in the shutdown state whilst the containment is open or severe accident mitigating measures are out of service;
- at all times when loss of confinement is caused by containment bypass, e.g. rupture of a steam generator tube, isolation valves are open or an interfacing system LOCA.

Different mechanisms and phenomena that could threaten the containment integrity during an accident with fuel melt, or due to the initiating event, have to be studied. It has to be shown that the failure of the containment function resulting from these events is practically eliminated. This requires reliability of the complementary safety features to manage the threats, as well as deterministic analyses of each mechanism and phenomenon, which need to be supported by adequate experimental results. Deterministic analyses are used to show that the containment function is fulfilled under design conditions of the containment including the expected conditions for the sequences which have not been practically eliminated, leading only to limited protective measures. Deterministic and probabilistic analyses are used to show that conditions leading to failure of the containment function due to physical phenomena or system failures are practically eliminated.

Deterministic analyses shall cover the expected course of severe accident scenarios. They are carried out with realistic assumptions and best estimate methodologies. Parameter variations have to be utilised in order to show the robustness and reliability of the approach. The probabilistic risk assessment is an essential supplement for the deterministic analyses. Analyses shall cover all the plant states (power operation, refuelling outages, maintenance, etc.) as well as different initiating event classes (internal events, fire, seismic events …).

Accident sequences with core melt resulting from external hazards which would lead to early or large releases should be practically eliminated.

Safety demonstration

Demonstration of Practical Elimination via Physical Impossibility

Demonstration of physical impossibility, based on engineered provisions, can be difficult. Care must be taken to recognize that some claims for practical elimination may be based on assumptions (e.g. non-destructive testing, inspection) and those assumptions need to be acknowledged and addressed. For engineered provisions this can be done by excluding the certain feature from the design making further development of accident scenario impossible (accident sequence cut-off).

A very simple example of a physically impossible situation is the case of a 10 m high load drop to ground level which is not possible from a crane at ground level with a maximum lift height of 5m. Most cases however are not so simple to consider, but representative examples are:

- in the reactor core design negative reactivity feedback protects the plant against a self accelerating reactivity accident;
- eliminating from the design those component features and/or failures which may initiate specific accident sequences. For example designing the spent fuel pools in such a way that the coolant cannot escape the pools.

Demonstration of Practical Elimination as extremely unlikely with a high degree of confidence

(1) The degree of substantiation provided for a practical elimination demonstration should take account of the assessed frequency of the situation to be eliminated and of the degree of confidence in the assessed frequency (uncertainties associated with the data and methods shall be evaluated in order to underwrite the degree of confidence claimed). Appropriate sensitivity studies should be included to confirm that sufficient margin to cliff edge effects exist. For engineered provisions the practical elimination can be done for instance by providing substantial increase of the protective means reliability.

(2) Practical elimination of an accident sequence cannot be claimed solely based on compliance with a general cut-off probabilistic value. Even if the probability of an accident sequence is very low, any additional reasonably practicable design features, operational measures or accident management procedures to lower the risk further should be implemented.

(3) The most stringent requirements regarding the demonstration of practical elimination should apply in the case of an event/phenomenon which has the potential to lead directly to a severe accident, i.e. to pass from DiD level 1 to level 4. For example demonstration of practical elimination of a heterogeneous boron dilution fault would require a detailed substantiation. Good examples of such detailed substantiation already exist in the form of cases made to exclude reactor vessel failure.

(4) It must be ensured that the practical elimination provisions remain in place and valid throughout the plant lifetime. For example, in-service inspection and other periodic checks may be necessary.

(5) All codes and calculations must be validated against the specific phenomena in question and verified.

03.6 Position 6: External hazards

Introduction

This section provides a common position on the consideration of external hazards for new reactors. The purpose is to provide high level guidance on regulatory expectations on how external hazards should be considered in the design and siting of new reactors.

Here the external hazards of concern are those natural or man-made hazards to a site and facilities that originate externally to both the site and its processes, i.e. the licensee may have very little or no control over the initiating event. Malicious actions are not included in the scope of this study.

The assessment of natural external hazards requires knowledge of natural processes, along with plant and site layout. In contrast with almost all internal faults or hazards, external hazards may simultaneously affect the whole facility, including back up safety systems and non-safety systems alike. In addition, the potential for widespread failures and hindrances to human intervention may occur. For multi-facility sites this makes the generation of safety cases more complex and requires appropriate interface arrangements to deal with common equipment or services as well as potential domino effects.

Safety Expectations

The safety assessment for new reactors should demonstrate that threats from external hazards are either removed or minimised as far as reasonably practicable.

This may be done by showing that all relevant safety Structures, Systems and Components (SSCs)[22] required to cope with an external hazard are designed and adequately qualified to withstand the conditions related to that external hazards.

External Hazards considered in the *general design basis*[23] of the plant should not lead to a core melt accident (Objective O2 i.e. level 3 DiD).

Accident sequences with core melt resulting from external hazards which would lead to early or large releases should be practically eliminated (Objective O3 i.e. level 4 DiD). For that reason*, rare and severe external hazards*[24], which may be additional to the *general design basis*, unless screened out (see “Screening of External Hazards” below), need to be taken into account in the overall safety analysis.

For new reactors external hazards should be considered as an integral part of the design and the level of detail and analysis provided should be proportionate to the contribution to the overall risk.

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[22] The words “all relevant safety Structures, Systems and Components (SSCs)” has the same meaning as “items important to safety” in IAEA’s terminology.

[23] The *general design basis* is that used to define the events that have been taken into account in the design and associated design basis analysis

[24] Rare and severe external hazards are additional to the *general design basis*, and represent more challenging or less frequent events. This is a similar situation to that between Design Basis Conditions (DBC) and Design Extension Conditions (DEC); they need to be considered in the design but the analysis could be realistic rather than conservative.

Safety Demonstration

A number of stages are envisaged:

- Identification
- Screening
- Determination of hazard parameters
- Analysis

Identification of External Hazards

The first step in addressing the threats from external hazards is to identify those that are of relevance to the site and facility under consideration. Any identified external hazard that could affect a facility should be treated as an event that can give rise to possible initiating events.

The list of external hazards to be considered should be as complete as possible and include all of the hazards mentioned in the relevant IAEA sources[25]. These sources have been combined to produce a consistent and coherent list which is included in the end of this section. This generic list is a starting point and it is expected that it would be augmented by any site specific hazards not included. The overall demonstration should include justification that the list (generic + site specific) is complete and relevant to the local site.

Screening of External Hazards

Screening is used to select the External Hazards that should be analysed. The screening process should take as a starting point the complete list discussed in the previous section. Each external hazard on the list should be considered and selected for analysis if:

a. It is physically capable of posing a threat to nuclear safety, and
b. the frequency of occurrence of the external hazard is higher than pre-set criteria.

The pre-set frequency criteria may differ depending on the nature of the analysis that is to be undertaken. Typically for the *general design basis*, where the analysis will be done using traditional conservative methods, assumptions and data, the criterion will be higher than the frequency criteria used for analyses of *rare and severe external hazards* or PSA that could employ realistic, best estimate methods and data. Therefore the screening process may lead to separate, but compatible lists of external hazards for the range of analyses to be undertaken and there should be a clear and consistent rationale for the differences in the lists.

In all cases the pre-set frequency criteria used should be stated and justified taking into account the way the hazards are going to be analysed in the safety demonstration.

The degree of confidence of the estimated frequency of occurrence should be stated and justified taking into account the related uncertainties according to the state of knowledge.

The screening process should explicitly consider correlated events and combinations of events.

[25] See Safety Series Standards NS-R-3, NS-G-3.1, NS-G-3.3, NS-G-3.6, NS-G-1.5, NS-G-1.6 and relevant events in SSG-3 and SSG-18

Determination of hazard parameters

All of the candidate external hazards that are selected should be characterised in terms of their severity and/or magnitude and duration. The characterisation of the external hazard will depend on the type of analysis that is to be carried out and shall be conservative for the *general design basis* analysis and could be realistic/best estimate for *rare and severe external hazards* analysis and PSA. It should be noted that for external hazards PSA, a range of frequencies and associated hazard parameters is often required. All relevant characteristics need to be specified and the rationale for their selection justified. For some external hazards:

- the ability to forecast the magnitude and timing of the event, and the speed at which the event develops may be relevant and should be considered;
- several parameters could be relevant to characterize severity and/or magnitude.

Analysis Considerations

The external hazards analysis includes the design of SSCs which are relevant to ensuring that the fundamental safety functions are fulfilled, development of probabilistic models where necessary, and the consideration of *rare and severe external hazards*. The following should be considered when undertaking this analysis:

- Minimising the risk from external hazards by initial siting of the facility
- Designing plant layout to minimise impact of external hazards (this is particularly important for multi unit facilities – also where units are of different generation)
- Justification of the lists of identified external hazards
- Justification of any hazard screening
- Combinations of external hazards that can occur simultaneously or successively within a given period of time[26] including correlated hazards and those combinations which occur randomly
- Consideration of consequential events, such as fire or flooding following a seismic event
- External hazard induced multiple failure of safety systems and/or their support systems
- Cliff edge effects – where a small change in a parameter leads to a disproportionate increase in consequence.
- In addition to considering the impact of external hazards on the systems and components, the reliability of the buildings and structures responding to an external hazard should be taken into account.
- The PSA for external hazards should include consideration of building and structural reliability as well as system and component fragilities and should take account of the potential for human response to be affected by the external event.

[26] The given period of time means that subsequent hazards occur within the mission time of the induced fault sequence. The mission time is the time necessary to reach pre defined safe, stable condition and not an arbitrarily assumed value.

- Impact of climate change and other potential time related changes that might affect the site should be considered
- Consideration should also be given to the impact of external hazards on the ability to support (emergency services) the site damaged by that external event (relevant to DiD).
- The design of the plant should reflect the external hazards analyses. Similarly the operating and maintenance procedures as well as the training etc. should take account of the external hazards analyses.
- Care must be taken where the definition of the hazard levels is imprecise, and claims are made based on the accuracy of calculations which have an accumulation of assumptions and conservatisms (or lack of)
- A clear methodology is important, along with an understanding of the associated uncertainties, both epistemic and aleatory. This is particularly important where the work also supports numerical PSA based approaches and where it is used to screen out hazards.
- The use of generic fragilities should be treated with care, as failure mechanisms may not be similar for similar types of plant, despite appearances
- Large uncertainties in characterisation of the hazards, particularly those selected for *general design basis*, need to be addressed as part of "cliff edge" considerations and margin assessments
- Multiple unit sites may need additional consideration for common plant areas and mitigation

Standards and guides

The following documents provide appropriate information to guide detailed consideration of external hazards.

(1) IAEA Safety Standards Site Evaluation for Nuclear Installations Safety Requirements No. NS-R-3

(2) IAEA Safety Guide - External Events Excluding Earthquakes in the Design of Nuclear Power Plants - Safety Guide No. NS-G-1.5

(3) IAEA Safety Guide - Seismic Design and Qualification for Nuclear Power Plants Safety Guide Safety - Standards Series No. NS-G-1.6

(4) IAEA SSG 9 Specific safety guide: Seismic Hazards in Site Evaluation for Nuclear Installations. Aug 2010

(5) IAEA Safety Guide - Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants Safety Guide - Safety Standards Series No. NS-G-3.6

(6) IAEA SSG-18 Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations. (replaces NS-G-3.4 and NS-G-3.5)

(7) IAEA Safety Guide NSG 3.1 External Human induced events in site evaluation

(8) IAEA Safety Guide NSG 3.3 Evaluation of Seismic Hazards for Nuclear Power Plant

(9) IAEA SSG 3 Development and Application of Level 1 PSA for Nuclear Power Plants

(10) IAEA SSR-2/1 Safety of Nuclear Power Plants: Design

Generic list of External Hazards

Category	Hazard
Seismotectonic	Ground motion
	Long period ground motion
	Liquefaction
	Dynamic compaction
	Tsunami
	Volcano (includes other effects than seismic)
	Meteorite (includes other effects than seismic)
Flooding	Extreme Rainfall (note links to other meteorological phenomena)
	Tidal Effects
	Storm Surge
	Seiche
	Tsunami
	Dam Failure
	Watercourse containment failure
	Wind generated waves
Meteorological	High Wind (Tornado, Hurricane, Cyclone Typhoon) and wind blown debris
	Extreme Drought
	Extremes of Air Temperature
	Extremes of Ground Temperature
	Extremes of Sea (or river) Temperature
	Lightning
	Snow (snow pack and snow melt)
	Icing
	Hail
	Humidity
	Air pressure
	Sandstorm, dust storm
	Saltspray/saltstorm
	Snow avalanche
	Waterspouts
	Ice flows / Frazil
	Mist/Fog
	Solar flares
Man Made	Accidental Aircraft Impact
	Gas Clouds (toxic, asphyxiates, flammables)

Category	Hazard
	Liquid Releases (flammables, toxic, radioactive)
	Fires
	Explosions (blast waves, missiles)
	Missiles (turbines, bottles BLEVE)
	Structural Failure
	Transport (road, sea, rail)
	Electromagnetic Interference
	Pipelines (Gas, Oil, Water)
	Vibrations
	Space Debris
	Flotsam/ Jetsam
	Log jam
	Pollution (ground or water course)
	Electrical Eddy currents into ground
	Military Activity (Accidental)
	Residual artefacts from previous use (i.e. munitions)
Biological	Seaweed
	Fish
	Jellyfish
	Marine growth
	Crustaceans, molluscs (shrimps, clams, mussels, shells)
	Birds
Infestation	Airborne swarms
	Infestation by rodents and other animals
Geological	Settlement
	Ground heave
	Mining (inactive or active)
	Caverns/ natural cavities
	Groundwater
	Leeching
	Contaminated land
	Landslides
	Radon
	Fissures
	Faults
	Soluble Rocks
	Unstable Soils (quick clays etc.)
	Permafrost

03.7 Position 7: Intentional crash of a commercial airplane

Introduction

Accidental crashes of airplanes have been considered in the design of reactors for several decades. However, according to the estimated frequencies of crashes, only crashes of small airplanes and/or military airplanes were generally taken into account. After the September 11th, 2001 attack, the consequences of an intentional crash of a commercial airplane were then considered.

Despite measures taken to prevent the intentional crash of a commercial airplane[27], this event should be considered in the design of new reactors.

This event is considered by WENRA as a very significant example of the expectations regarding the improvement of the interface between safety and security issues, as stated in Objective O5.

Expected safety level

The general expectation is that such a crash should not lead to core melt and therefore not cause more than a minor radiological impact as stated in Objective O2. Nevertheless, in this specific situation, it is recognized that releases of radioactive materials could exceed those considered in other events not involving core melt. However, the consequences of this specific situation should remain within Objective O2.

Safety functions required to bring and maintain the plant in a safe state after such a crash shall be designed and protected adequately.

In particular, the following shall be ensured:

- Reactivity control, including reactor scram;
- Residual heat removal (including in the long term) from the core in the vessel and the fuel pool in order to exclude core or fuel melt;
- Confinement of radioactive materials, consistent with radiological consequences of Objective O2.

Key aspects of the safety demonstration which is expected from the licensees

Direct and indirect effects of the airplane crash shall be considered, in particular:

- effects of direct and secondary impacts on mechanical resistance of safety structures and systems required to bring and maintain the plant in a safe state after airplane crash;
- effects of vibrations on safety structures and systems required to bring and maintain the plant in a safe state after airplane crash;
- effects of combustion and/or explosion of airplane fuel on the integrity of the necessary structures and on the systems required to bring and maintain the plant in a safe state after airplane crash.

[27] Characterized by load/time curves.

Buildings or appropriate part of the buildings containing nuclear fuel and housing key safety functions should be designed to prevent airplane fuel from entering them. Fires caused by airplane fuel shall be assessed as different kinds of fire ball and pool fire combinations. Other consequential fires due to the airplane crash shall be addressed.

A realistic approach can be followed, using best estimate material properties and state-of-the-art analytical methods. Realistic failure criteria could be used. In addition it is not necessary to consider other coincident failure of plant and equipment. Sensitivity analysis shall be performed to confirm sufficient margin to cliff edge effects.

The effect of the event on the ability of plant personnel and off-site services to fulfil necessary actions shall be taken into account.

04
Lessons Learnt from the Fukushima Dai-ichi accident

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A severe accident involving several units took place in Japan at Fukushima Dai-ichi nuclear power plant in March 2011. Even though in-depth analysis of this accident has not yet been completed, some items could be highlighted. The immediate cause of the accident was an earthquake followed by a tsunami coupled with inadequate provisions for tsunamis in the original design. Opportunities to improve protection against a tsunami were not adequately taken, which could have been possible for example as part of the PSR process.

Safety culture and organisational factors, including decision making capabilities, contributed to the inadequate protection of the plants and to the difficulties in accident management.

As a consequence of the tsunami, essential safety functions were lost at the plant, leading to core damage in three units and subsequently to considerable radioactive releases.

The Fukushima Dai-ichi accident demonstrates the importance of properly implementing the Defence-in-Depth principle to ensure safety, getting the design basis for external hazards right, providing adequate protection against external hazards and the need to ensuring strong PSR process together with independent regulatory body to drive it. The accident also confirmed the need to have comprehensive safety analysis using both deterministic and probabilistic methods in a complementary manner to provide as full coverage of all safety factors as possible. In the safety assessment specific considerations are needed for multi-unit sites and to address long term aspects.

The Fukushima Dai-ichi accident also demonstrates the importance of adequate on-site resources that are adequately qualified against external hazards and the effects of core melt accidents.

An important lesson from the accident was the importance of a control room and emergency response centre adequately protected against external hazards. Another key lesson was the need to attend to cooling and integrity of spent fuel pools as well as for the reactors. Siting has design implications, in particular in terms of securing sufficient diverse electrical and cooling supplies.

In general, one has to bear in mind that the specific nature of individual events and challenges can never be completely taken into account in design and operation of a nuclear power plant (or indeed any other industrial facility). However, a robust design based on DiD with sizeable safety margins and diverse means for delivering fundamental safety functions as well as comprehensive operator response plans will help to protect against the unanticipated.

Several studies have already been performed concerning the accident and detailed technical studies are still in progress in Japan and elsewhere. In the following conclusions on some essential safety issues based on or reinforced by the lessons learnt from Fukushima Dai-ichi accident are presented, in relation with the positions detailed in Chapter 3.

04.1 External hazards

The Fukushima Dai-ichi accident has reinforced the need to undertake a comprehensive analysis of all external hazards as part of the design process for new nuclear power stations, and periodic safety reviews. In common with other parts of the safety demonstration, the external hazard analysis should cover all areas with significant amounts of radioactive material on the power station.

The Fukushima Dai-ichi accident has highlighted the need to take account of rare and severe hazards. External hazards are comprehensively considered in **Position 6**.

04.2 Reliability of safety functions

Lessons from the Fukushima Dai-ichi accident show the importance of proper implementation of the Defence-in-Depth (DiD) concept and a need for adequate protection of the plants against rare and severe external hazards.

External hazards are comprehensively considered in **Position 6**.

The defence in depth approach, independence of the levels of defence in depth, and multiple failure events are comprehensively considered in **Positions 1, 2 and 3**.

Decay heat removal

The nuclear power plant shall have arrangements to enable the decay heat removal in rare and severe hazards (**Position 6**). For this situation, protection of necessary electrical power supplies has to be ensured. Consistently with the DiD approach of **Position 1**, loss of the primary ultimate heat sink or access to it should be considered in the design. The primary and alternative means for decay heat removal in an emergency should function independently.

Ensuring the energy supply

Where safety functions of NPPs rely on AC power, diverse emergency AC power supply shall be required as a part of DiD sub-level 3.b additional safety features to cope with common cause failures of the primary emergency electrical power supply (**Positions 2 and 3**). Other actions for increasing the reliability of electrical power supply at NPPs deal with enhanced provisions of long term fuel and lubricating oil reserves for all emergency power units at the site and ensuring possibilities to use mobile power supply units. Adequate battery capacity shall be secured. This requires appropriate capacity of some critical batteries and may require improving possibilities to re-charge them.

The correct fail-safe position of safety related equipment in case of loss of energy supply needs to be considered in the design taking into account potential conflicting demands on this equipment.

04.3 Accidents with core melt

The Fukushima Dai-ichi accident confirms that accidents with core melt need to be considered in the design of NPPs. Complementary safety features (as defined in **Position 2**) which ensure the adequate integrity of the containment in case of an accident leading to a core melt need to be included in the design, as discussed in **Position 4**. Robust complementary safety features (DiD level 4) specifically designed for fulfilling safety functions required in postulated core melt accidents should be independent to the extent reasonably practicable from the SSCs of the other levels of DiD, as discussed in **Position 2** and **Position 4**. Accidents with core melt which would lead to early or large releases should be practically eliminated. Analyses shall cover all the plant states (power operation, refueling, outages, maintenance etc.) as well as different initiating event classes (internal events, fire, seismic events, ...), as discussed in **Position 5**.

Essential design principles related to the Fukushima Dai-ichi accident deal with having a filtering capability for the containment venting if any, containment ultimate pressure strength and hydrogen management, also discussed in **Position 4**.

The need to manage large volumes of contaminated cooling water and filtered containment venting over longer periods of time should be included in the design and accident management considerations.

04.4 Spent fuel pools

The Fukushima Dai-ichi accident also highlighted the need for adequate safety and the design of spent fuel pools. This implies that single initiating events, multiple failure events (see **Position 3**), internal hazards as well as external hazards (see **Position 6**) should be properly addressed. In addition to having adequate instrumentation and control for the spent fuel pool, also under accident conditions, WENRA considers that both the defence in depth approach (see **Position 1**) and the practical elimination of accidents with early or large release (see **Position 5**) are fully applicable for fuel storage pools.

Once spent fuel in a pool is overheated, the further development is very difficult to assess. Thus the primary approach for spent fuel pools shall be to “practically eliminate” the possibility of extensive fuel damage due to mechanical, thermal or chemical effects. To achieve this it is essential to ensure the integrity of the spent fuel pools, and maintain sufficient water level in the pools. In addition, subcriticality of the fuel has to be ensured. The strategy to practically eliminate the fuel damages can take into account that time delays of spent fuel heating up in the case of loss of normal cooling systems usually are relatively long (unless the reactor core has been recently transferred into the pool). Practical elimination is discussed in **Position 5**.

The structural integrity of the spent fuel pools needs to be ensured, as needed to maintain sufficient water level in the pools in case of rare and severe external hazards.

04.5 Safety assessment

A strong periodic safety review (PSR) process is very important for continuous improvement of safety of nuclear power plants. In the event that PSR results indicate the need for improvement measures, it is vital that the measures are defined and implemented in an effective manner.

Long term accident mitigation measures should be considered in deterministic and probabilistic safety assessments and consideration given to the reliability and sustainability of the measures.

On multi-unit sites, the plant should be considered as a whole in safety assessments and interactions between different units need to be analysed. Hazards that may affect several units need to be identified and included in the analysis.

04.6 Emergency preparedness in design

The Fukushima Dai-ichi accident showed that events disrupting the regional infrastructure and affecting several units at the same site can have a significant adverse impact on the implementation of the required accident management actions.

The accessibility, functionability and habitability of the control room and of the emergency response centre have to be ensured. This will require adequate protection against rare and severe external hazards. Suitably shielded and protected spaces shall be provided to house necessary workers under postulated core melt accident conditions. The accessibility of local control points required for manual actions also has to be ensured.

The reliability and functionality of the on-site and off-site communication systems, equipment measuring releases, radiation levels and meteorological conditions need to be ensured, taking into account conditions related to rare and severe external hazards.

Annex 1
WENRA Statement on Safety Objectives for new Nuclear Power Plants, November 2010

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Foreword

One of the objectives of WENRA, as stated in its terms of reference, is to develop a harmonized approach to nuclear safety and radiation protection issues and their regulation.

A significant contribution to this objective was the publication, in 2006[28], of a report on harmonization of reactor safety in WENRA countries. This report addresses the nuclear power plants that were in operation at that time in those countries; it includes about 300 "Reference Levels"[29].

Since then, the construction of new nuclear power plants has begun or is being envisaged in the short term in several European countries.

Hence, it has been considered timely for WENRA to define and express a common position on the safety objectives of new nuclear power plants, so that:

- new nuclear power plants to be licensed across Europe in the next years will be safer than the existing ones, especially through improvements of the design;
- regulators press for safety improvements in the same direction and ensure that these new plants will have high and comparable levels of safety;
- applicants take into account this common position when formulating their regulatory submissions.

A report "Safety objectives for new power reactors – study by RHWG – December 2009" has been published by WENRA in January 2010 for stakeholders' comments. Comments received were considered one by one either in establishing the present statement (e.g. comments on the safety objectives themselves) or as an input for the ongoing WENRA work related to new nuclear power plants. In particular, some clarifications were made to the safety objectives stated in the December 2009 study. These seven safety objectives in their final wording (November 2010), as decided by WENRA, are stated below. They will be the basis for further harmonization work.

Improving the protection of people and of the environment

[28] Harmonization of Reactor Safety in WENRA countries, report by RHWG, January 2006

[29] These "Reference Levels" were updated in January 2008

WENRA considers that the design of new nuclear power plants shall take into account the operating experience feedback, lessons learnt from accidents, developments in nuclear technology and improvement in safety assessment.

The safety objectives for new nuclear power plants have been defined on the basis of a systematic investigation of the Fundamental Safety Principles (SF-1 document issued 2006 by the IAEA). Grounding the safety objectives on the fundamental safety principles has been explained in the December 2009 study[30].

The safety objectives address new civil nuclear power plant projects. However, these objectives should be used as a reference for identifying reasonably practicable safety improvements for “deferred plants”[31] and existing plants during periodic safety reviews.

These safety objectives are formulated in a qualitative manner[32] to drive design enhancements for new plants with the aim of obtaining a higher safety level compared to existing plants. For instance,

- to be able to comply with the qualitative criteria proposed in following Objective O3, the confinement features should be designed to cope with core melt accidents, even in the long term;
- these safety objectives call for an extension of the safety demonstration for new plants, in consistence with the reinforcement of the defence in depth. Some situations that are considered as “beyond design” for existing plants, such as multiple failures conditions and core melt accidents, are considered in the design of new plants.

Based on these safety objectives, WENRA is currently developing positions on selected key issues for the design of new nuclear power plants.

WENRA considers that these safety objectives reflect the current state of the art in nuclear safety and can be implemented at the design stage using the latest available industrial technology of nuclear power plants.

However, since nuclear safety and what is considered adequate protection are not static entities, these safety objectives may be subject to further evolution. As technology and scientific knowledge advance, WENRA deems these safety objectives should be reviewed no later than 2020.

[30] In particular, in line with fundamental safety principle 5 “optimization of protection”, the safety of new reactors will have to be improved as far as reasonably achievable starting from the design stage, taking into consideration the state of the art and by taking into account all circumstances of individual cases, as defined in SF-1, para. 3.23 (related objectives are O1 to O4 and O6)

[31] Plant project originally based on design similar to currently operating plants, the construction of which halted at some point in the past and is now being completed with more modern technology

[32] WENRA considered quantitative safety objectives but concluded that they would not be more informative than qualitative objectives with associated safety expectations. It was also recognized that the use of quantitative safety goals needs some prerequisites, such as the development of standardized methodologies. Furthermore, compliance with a numerical value may not be enough.

WENRA Safety Objectives for New Nuclear Power Plants

Compared to currently operating nuclear power plants, WENRA expects new nuclear power plants to be designed, sited, constructed, commissioned and operated with the objectives of:

O1. Normal operation, abnormal events and prevention of accidents

- reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation.
- reducing the potential for escalation to accident situations by enhancing plant capability to
control abnormal events.

O2. Accidents without core melt

- ensuring that accidents without core melt induce[33] no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation[34]).
- reducing, as far as reasonably achievable,
    - the core damage frequency taking into account all types of credible hazards and failures and credible combinations of events;
    - the releases of radioactive material from all sources.
- providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.

[33] In a deterministic and conservative approach with respect to the evaluation of radiological consequences.

[34] However, restriction of food consumption could be needed in some scenarios.

O3. Accidents with core melt

- reducing potential radioactive releases to the environment from accidents with core melt[35], also in the long term[36], by following the qualitative criteria below:
    - accidents with core melt which would lead to early[37] or large[38] releases have to be practically eliminated[39];
    - for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

O4. Independence between all levels of defence-in-depth

- enhancing the effectiveness of the independence between all levels of defence-in-depth, in
  particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives), to provide as far as reasonably achievable an overall reinforcement of defence-in-depth.

O5. Safety and security interfaces

- ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.

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[35] For new plants, the scope of the safety demonstration has to cover all risks induced by the nuclear fuel, even when stored in the fuel pool. Hence, core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool. It has to be shown that such accident scenarios are either practically eliminated or prevented and mitigated.

[36] Long term: considering the time over which the safety functions need to be maintained. It could be months or years, depending on the accident scenario.

[37] Early releases: situations that would require off-site emergency measures but with insufficient time to implement them.

[38] Large releases: situations that would require protective measures for the public that could not be limited in area or time.

[39] In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).

O6. Radiation protection and waste management

- reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities:
    - individual and collective doses for workers;
    - radioactive discharges to the environment;
    - quantity and activity of radioactive waste.

O7. Leadership and management for safety

- ensuring effective management for safety from the design stage. This implies that the licensee:
    - establishes effective leadership and management for safety over the entire new plant project and has sufficient in house technical and financial resources to fulfil its prime responsibility in safety;
    - ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of new plants demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety.

WENRA
WESTERN EUROPEAN NUCLEAR
REGULATORS ASSOCIATION

RHWG
REACTOR HARMONISATION
WORKING GROUP

WGWD
WORKING GROUP ON WASTE
AND DECOMISSIONING

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WENRA
Nuclear Regulators Association

WENRA

Statement

SAFETY OBJECTIVES FOR NEW NUCLEAR POWER PLANTS

November 2010

Western European

WENRA

Nuclear Regulator's Association

November 2010

WENRA STATEMENT
ON
SAFETY OBJECTIVES
FOR NEW NUCLEAR POWER PLANTS

Foreword

One of the objectives of WENRA, as stated in its terms of reference, is to develop a harmonized approach to nuclear safety and radiation protection issues and their regulation.

A significant contribution to this objective was the publication, in 2006[1], of a report on harmonization of reactor safety in WENRA countries. This report addresses the nuclear power plants that were in operation at that time in those countries; it includes about 300 "Reference Levels"[2].

Since then, the construction of new nuclear power plants has begun or is being envisaged in the short term in several European countries.

Hence, it has been considered timely for WENRA to define and express a common position on the safety objectives of new nuclear power plants, so that:

- new nuclear power plants to be licensed across Europe in the next years will be safer than the existing ones, especially through improvements of the design;
- regulators press for safety improvements in the same direction and ensure that these new plants will have high and comparable levels of safety;
- applicants take into account this common position when formulating their regulatory submissions.

A report "Safety objectives for new power reactors – study by RHWG – December 2009" has been published by WENRA in January 2010 for stakeholders' comments. Comments received were considered one by one either in establishing the present statement (e.g. comments on the safety objectives themselves) or as an input for the ongoing WENRA work related to new nuclear power plants. In particular, some clarifications were made to the safety objectives stated in the December 2009 study. These seven safety objectives in their final wording (November 2010), as decided by WENRA, are stated below. They will be the basis for further harmonization work.

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[1] Harmonization of Reactor Safety in WENRA countries, report by RHWG, January 2006

[2] These "Reference Levels" were updated in January 2008

Improving the protection of people and of the environment

WENRA considers that the design of new nuclear power plants shall take into account the operating experience feedback, lessons learnt from accidents, developments in nuclear technology and improvement in safety assessment.

The safety objectives for new nuclear power plants have been defined on the basis of a systematic investigation of the Fundamental Safety Principles (SF-1 document issued 2006 by the IAEA). Grounding the safety objectives on the fundamental safety principles has been explained in the December 2009 study[3].

The safety objectives address new civil nuclear power plant projects. However, these objectives should be used as a reference for identifying reasonably practicable safety improvements for "deferred plants"[4] and existing plants during periodic safety reviews.

These safety objectives are formulated in a qualitative manner[5] to drive design enhancements for new plants with the aim of obtaining a higher safety level compared to existing plants. For instance,

- to be able to comply with the qualitative criteria proposed in following safety objective O3, the confinement features should be designed to cope with core melt accidents, even in the long term;
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Based on these safety objectives, WENRA is currently developing positions on selected key issues for the design of new nuclear power plants.

WENRA considers that these safety objectives reflect the current state of the art in nuclear safety and can be implemented at the design stage using the latest available industrial technology of nuclear power plants.

However, since nuclear safety and what is considered adequate protection are not static entities, these safety objectives may be subject to further evolution. As technology and scientific knowledge advance, WENRA deems these safety objectives should be reviewed no later than 2020.

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[3] In particular, in line with fundamental safety principle 5 "optimization of protection", the safety of new reactors will have to be improved as far as reasonably achievable starting from the design stage, taking into consideration the state of the art and by taking into account all circumstances of individual cases, as defined in SF-1, para. 3.23 (related objectives are O1 to O4 and O6)

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- providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.

O3. Accidents with core melt

- reducing potential radioactive releases to the environment from accidents with core melt[8], also in the long term[9], by following the qualitative criteria below:
  - accidents with core melt which would lead to early[10] or large[11] releases have to be practically eliminated[12] ;
  - for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

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[10] Early releases: situations that would require off-site emergency measures but with insufficient time to implement them.

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Safety Objectives for New Power Reactors

December 2009

Western European

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Safety Objectives for New Power Reactors

Study by

WENRA Reactor Harmonization Working Group

December 2009

RHWG

Reactor Harmonization Working Group

Table of content

1. INTRODUCTION

One of the objectives of WENRA, as stated in the policy statement signed in Stockholm in December 2005, is to develop a harmonized approach to nuclear safety and radiation protection issues and their regulation.

A significant contribution to this objective was the publication, in 2006[1], of a report on harmonization of reactor safety in WENRA countries. This report addresses the nuclear power plants that were in operation at that time in those countries.

Since then, the construction of new nuclear power plants has begun or is being envisaged in the short term in several European countries. Furthermore, some plants whose construction had been halted several years ago are now under completion. Despite all these plants were not addressed in the study published in 2006, it is expected that, as a minimum, they should meet the corresponding "Safety Reference Levels".

These "Safety Reference Levels" were designed to be demanding for existing reactors. However, in line with the continuous improvement of nuclear safety that WENRA members aim for, new reactors are expected to achieve higher levels of safety than existing ones, meaning that in some safety areas, fulfillment of the "Safety Reference Levels" defined for existing reactors may not be sufficient.

Hence, it has been considered timely for WENRA to define and express a common view on the safety of new reactors, so that:

- new reactors to be licensed across Europe in the next years offer improved levels of protection compared to existing ones;
- regulators press for safety improvements in the same direction and ensure that these new reactors will have high and comparable levels of safety;
- applicants take into account this common view when formulating their regulatory submissions.

In addition, this common view could provide insights for the periodic safety reviews of existing reactors.

2. MANDATE

In March 2008, the Reactor Harmonization Working Group (RHWG) reported to WENRA on a proposal for a study on new reactors. Such a study would consist of the following tasks:

- identify and review the existing relevant documentation on new reactors (IAEA and NEA documents, national regulations and other relevant documents);
- on this basis, select from this documentation a justified set of safety objectives, safety principles and specific considerations which are relevant for new reactors;
- review these safety objectives, safety principles and specific considerations against the "Safety Reference Levels" for existing reactors, and indicate where the "Safety Reference Levels" may need completion or updating.

After discussion, WENRA members mandated the RHWG to perform a pilot study, which should concentrate on safety goals and a limited test of the proposed methodology. This would give WENRA the necessary elements to make a decision on the continuation of the study. In October 2008, WENRA members asked RHWG to consider potential quantitative safety goals to complement the qualitative safety objectives.

It is worth mentioning that unlike the 2006 study, the objective of the present study is not to develop reference levels for new reactors nor to benchmark projects or designs.

[1] Harmonization of Reactor Safety in WENRA countries, report by RHWG, January 2006

3. SCOPE OF THE PRESENT STUDY ON NEW REACTORS

The RHWG has had many discussions on the definition of "new reactors", the main difficulties being that it is not a static concept in time and that it can refer to various states of development of a project (from not at all designed up to just commissioned). The case of "deferred plants", that are plant projects originally based on reactor design similar to currently operating plants, the construction of which halted at some point in the past, and now being completed with more modern technology, is particularly illustrative of these difficulties.

The main intent of the present study, because it corresponds to a need for WENRA members, is to address the civil nuclear power reactors projects that are under way or planned in the short term, at the time of completion of the study. These projects are based on designs that are largely completed. Two plants are even under construction in WENRA countries (in Finland and France).

However, since technology and scientific knowledge advance, RHWG suggests that the proposed safety objectives be reviewed no later than 2020, and before if appropriate.

As regards deferred plants, the objectives proposed in the study may not be fully applicable. However, these objectives should be used as a reference for identifying reasonably practicable safety improvements.

4. OVERVIEW OF THE METHODOLOGY

The methodology is based on identification of the existing relevant documentation and, by consensus, selection and rewording of the items relevant for the purpose of the study. As for the previous RHWG study (Harmonization of reactor safety in WENRA countries, 2006), the IAEA documents were a major input to this process.

4.1. Identification and review of the existing relevant documentation

In order to comprehensively consider the safety aspects relevant to new reactors, the following key documents were identified to be analyzed:

- the most advanced safety standards from the IAEA and INSAG publications;
- the regulations and guidance published in WENRA countries or other countries and explicitly addressing new reactors;
- the publications that have been issued in various contexts in order to improve the safety of operating power reactors and to find innovative approaches.

The main documents reviewed for the purpose of this study are listed in Annex 1.

4.2. Definition of safety objectives

The safety objectives for new reactors have been defined on the basis of a systematic investigation of the Fundamental Safety Principles (SF-1 document issued 2006 by the IAEA).

Each Fundamental Safety Principle has been investigated to check whether, on the basis of the review of the existing documentation, safety objectives related to this principle needed to be further expressed.

4.3. Investigation of quantitative safety targets

RHWG investigated the implications of defining common quantitative safety targets associated with the safety objectives. To this aim, current experience on the use of such quantitative safety targets was investigated by issuing a questionnaire on qualitative and quantitative safety targets used in WENRA countries. The answers provided included considerations on the usefulness of using numerical values and on the prerequisites and conditions for their use.

On the basis of the answers to the questionnaire, each safety objective defined in step 4.2 was considered to check if quantitative safety targets used in at least one WENRA country could help support the corresponding common safety expectation.

The RHWG discussed each candidate target and aimed to reach a consensus on whether to retain the target for the study.

The RHWG also discussed, on the basis of the answers to the questionnaire and of the existing documentation, the conditions for the use of such targets.

4.4. Identification of areas of improvements in meeting the proposed safety objectives

Starting from the proposed safety objectives, RHWG identified examples of areas of improvements compared to existing reactors that could be considered or taken into account at the design stage and duly assessed in the safety demonstration. As this exercise was very time consuming, it was decided to start with a limited generic list.

It was outside the mandate of the present study to make a comprehensive, systematic search of generic safety issues of the present generation plants in order to check the extent to which they might be effectively addressed in the design of the new plants.

4.5. Review of the reference levels for existing reactors

RHWG has carried out a limited pilot exercise to categorize each of the "Safety Reference Levels" (January 2008 version) into the following groups:

A. Fully applicable, safety expectation not greater for new reactors

B. Wording acceptable for new reactors, but greater safety expectation

C. More stringent description is necessary

In addition, the missing "safety issues" or topics were identified (D).

This exercise is only preliminary.

5. CONSIDERATIONS FROM THE REVIEW OF THE DOCUMENTATION

5.1. Studies related to safety improvements for new reactors since 1990

Starting from the end of the 1980s, several milestones characterized the evolution of nuclear safety:

- many designers worldwide launched proposals for evolutionary and innovative new plants[2];
- a resolution of the IAEA General Conference in 1991[3] invited the Director General to start activities on safety principles for the design of future plants, INSAG 5[4] was issued;
- OECD NEA launched studies and produced documents on regulatory requirements for advanced nuclear power plants[5];
- in US, the utilities started an industry-wide effort to establish the technical foundation for the design of what they named "Advanced Light Water Reactors"[6] and the US NRC issued policies[7] related to evolutionary LWR issues in relation to current regulatory requirements;
- in Europe, the French GPR and German RSK issued a proposal for a common safety approach for future pressurised water reactors[8] and the European Commission issued a "consensus document"[9] on the safety of European LWR;
- the European Utilities Requirements[10] document was issued, as a result of a large common effort by utilities to produce a complete set of requirements for plants to be built in Europe;
- Several organisations - AVN (Belgium), AEA Technology (United Kingdom), ANPA (Italy), CIEMAT (Spain), GRS (Germany), IPSN (France) - took part in a European project[11], which examined in detail some relevant safety issues and analysed the European Utilities Requirements.

The overview of these studies shows a general consensus on "defense in depth" continuing to be the fundamental means of ensuring the safety of nuclear plants, and on the fact that it should be reinforced as far as possible.

It also provides the following overall picture of the general lines of evolution to be taken into account at the design stage for new reactors:

- address to the possible extent the recognized issues of the present generation plants (mid-loop operation, fire protection, intersystem LOCAs, common cause failures etc.), optimizing the balance of different safety measures (also by early performance of PSA) and enhancing the use of operating experience,
- increase the level of independence of the defence in depth levels,
- extend the design beyond traditional design basis, based on best estimate calculations and sound engineering practices, in the area of core melt prevention and mitigation, with particular emphasis on reducing the challenges and strengthening the capability of the containment. A

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2 ALWR, AP 600, PIUS, SBWR, MHTGR, ISIS …
3 Resolution GC(XXXVI)/RES/553 1991
4 INSAG 5 "The safety of Nuclear Power", 1992
5 e.g: NEA/CNRA/R(94)2 OECD-NEA " A review for regulatory requirements for advanced nuclear power plans" 1994
6 EPRI ALWR Utility Requirements Document (URD), 1985
7 e.g.: Secy – 90 – 016 "Evolutionary LWR Certification issues and their relationship to current regulatory requirements", 1990.
8 GPR/RSK common proposal for a safety approach for future pressurized water reactors. May 25, 1993
9 ISSN 1018 – 5593 Report EUR 16803 EN "1995 consensus document on safety of European LWR"
10 European Utilities Requirements for LWR nuclear power plants, Rev C, 2001
11 EUR 20163 EN "TSO study project on development of a common safety approach in EC countries for large evolutionary PWRs" – 2001.

fourth level of defence (systematic consideration of severe accidents) is called for from the beginning of the design process. The corresponding protection is supposed to be achieved primarily by design measures,

- reduce the necessity for off-site measures such as evacuation, and the potential for long term and large scale land contamination,
- increase the protection against external hazards, since the contribution of the risks coming from such hazards increases when the level of protection against internal failures and hazards has been substantially improved.

In many studies, numerical (mainly probabilistic) safety goals have been proposed, in order to provide an as broad as possible picture of the overall safety of the plants and acceptance criteria for the improvements to be made. Nevertheless, in the international debate, difficulties in defining sound quantitative criteria have been recognized and it was emphasized that meeting quantitative criteria should not preclude continuing efforts in seeking safety enhancement[12].

5.2. National regulations or guidance on new reactors

The main documents reviewed were those Finnish YVL guides that have been recently updated to be applicable for new plants, the French-German "Technical guidelines for the design and construction of the next generation of nuclear power plants with PWRs", the UK "Safety Assessment Principles" revised in 2006 and the Bulgarian regulations for new reactors.

The conclusions of this review are consistent with the general picture described in 5.1. In particular, the required approach mainly relies on a reinforcement of the defence-in-depth, both of each level and of their independency. The need to take into account severe accidents as part of the design is also stressed.

However, these regulations / guidance put emphasis on increasing the diversity of safety systems, on improving security driven design features (such as protection against large airplane crash) and on safety management in the design and construction phases.

5.3. Safety Pillars as expressed in IAEA documents

Recently, a large effort was started by IAEA to revise the "safety standards". In particular, a new IAEA document on Safety Fundamental Principles (SF-1) was published in 2006.

The analysis of the SF-1 document in light of the above mentioned studies shows that evolutions in safety for new reactors concentrate on strengthening the implementation of some of the Fundamental Safety Principles.

[12] IAEA TECDOC 905 "Approaches to Safety of Future Nuclear Power Plants" (1995)"

6. PROPOSED SAFETY OBJECTIVES

6.1. Foreword

The proposed safety objectives for new reactors have been selected to further improve the protection of people and of the environment.

However, since nuclear safety and what is considered adequate protection are not static entities, the safety objectives that are proposed in this report may be subject to further evolutions.

They have been formulated in a qualitative way, so that European citizens can easily understand WENRA's expectations in terms of:

- protection of the public (consequences of accidents) and workers (radiation protection);
- protection of the environment (discharges);
- protection of future generations (waste and dismantling).

6.2. Grounding the safety objectives on the fundamental safety principles

The fundamental safety principles, from IAEA SF-1 document published in 2006, were recognized to be a good basis for the present study. These fundamental safety principles have been used to ground the proposed safety objectives for new reactors. In this context, the following fundamental safety principles have been found to be especially relevant for improvement of safety of new reactors:

❑ **In line with fundamental safety principle n°5 "*optimization of protection*"**, the safety of new reactors will have to be improved as far as reasonably[13] achievable starting from the design[14] stage, with due consideration given to insights gained from:

- experience feedback from existing reactors ;
- deterministic and probabilistic safety assessments ;
- state-of-the art technologies, analysis methodologies and techniques ;
- results of safety research.

For existing plants, improvement potentials from those insights are implemented on a pragmatic basis within the limits of what is reasonably practicable, for example in the process of periodic safety reassessment, taking due account of the original design.

For new reactors, more significant improvements in the design over what has been done before become now reasonably achievable, in particular concerning prevention and mitigation of severe accidents, including in the long term phase.

PSA shall be used as part of the design process for new reactors.

➢ *See Objectives O1 to O4 and O6.*

❑ **In implementing the fundamental safety principle n°8 "*prevention of accidents*"**, the defence-in-depth concept remains the key safety approach for new reactors. Therefore, for new reactors, strengthening of the implementation of the concept has to be aimed for:

- reinforcement of each level of the defence in depth concept,

[13] By taking into consideration the state of the art and by taking into account all circumstances of individual cases, as defined in SF-1, para. 3.23
[14] beyond back fitting measures taken for existing plants

- improvement of the independence between the levels of defence in depth.

Based on this principle, security features for new reactors should also be considered consistently with safety ones.

It is also stressed that quality assurance and management of safety are key elements of the prevention of accidents.

➢ *See Objectives O1 to O5 and O7.*

❑ **In line with fundamental safety principles n°6 "*limitation of risks to individuals*" and n°7 "*protection of present and future generations*"**, the radiological and non radiological impact of normal and abnormal operation, potential accidents and decommissioning activities will have to be reduced at the design stage.

➢ *See Objectives O2, O3 and O6.*

❑ **In line with fundamental safety principle n°3 "*leadership and management of safety*"**, due consideration has to be given to safety management from an early stage coherently with security requirements.

➢ *See Objectives O5 and O7.*

The set of hereafter proposed safety objectives is based on these considerations.

6.3. Proposed safety objectives

Compared to currently operating reactors, new ones are expected to be designed, sited, constructed, commissioned and operated with the objectives of:

O1. Normal operation, abnormal events and prevention of accidents

- reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation.
- reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events.

O2. Accidents without core melt

- ensuring that accidents without core melt[15] induce[16] no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation[17]).
- reducing, as far as reasonably achievable,
    - the core damage frequency taking into account all types of hazards and failures and combinations of events;
    - the releases of radioactive material from all sources.
- providing due consideration to siting and design to reduce the impact of all external hazards[18] and malevolent acts.

[15] For new reactors, the scope of the defence-in-depth has to cover all risks induced by the nuclear fuel, even when stored in the fuel pool. Hence, core melt accidents (severe accidents) have to be considered when the core is in the reactor, but also when the whole core or a large part of the core is unloaded and stored in the fuel pool.

[16] in a deterministic and conservative approach with respect to the evaluation of radiological consequences.

[17] However, restriction of food consumption could be needed in some scenarios.

[18] As defined in Reference Level E 5.2., January 2008 version.

O3. Accidents with core melt

- reducing potential radioactive releases to the environment from accidents with core melt, also in the long term[19], by following the qualitative criteria below:
    - accidents with core melt which would lead to early[20] or large[21] releases have to be practically eliminated[22] ;
    - for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

O4. Independence between all levels of defence-in-depth

- enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives) to provide, as far as reasonably achievable, an overall reinforcement of defence-in-depth.

O5. Safety and security interfaces

- ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.

O6. Radiation protection and waste management

- reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities :
    - individual and collective doses for workers;
    - radioactive and non radioactive discharges to the environment;
    - quantity and activity of radioactive waste.

O7. Management of safety

- ensuring effective management of safety from the design stage. This implies that the licensee:
    - establishes effective leadership and management of safety over the entire new plant project and has sufficient in house technical and financial resources to fulfil its prime responsibility in safety;
    - ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of new reactors demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety.

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[19] Long term: considering the time over which the safety functions need to be maintained. It could be months or years, depending on the accident scenario.

[20] early releases : situations that would require off-site emergency measures but with insufficient time to implement them.

[21] large releases : situations that would require protective measures for the public that could not be limited in area or time.

[22] In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NSG1.10).

6.4. Impact of these safety objectives

These safety objectives are clearly formulated to drive design improvements for new plants, and hence obtain a higher safety level compared to existing plants. These design improvements are of two kinds:

- Improvements that are in technological continuity with currently operating plants. These improvements are mainly based on optimised and evolutionary features derived from the lessons learned from operating experience and probabilistic studies performed for existing plants;
- Improvements that represent a significant step in the safety level compared to existing plants. These improvements are based on innovative design features derived from research and development, only achievable if considered at the design stage.

In particular, to be able to comply with the qualitative criteria proposed in objective O3, the confinement features should be designed to cope with core melt accidents, even in the long term, which typically is not the case for currently operating reactors.

Moreover, these safety objectives call for an extension of the safety demonstration for new reactors, in consistence with the reinforcement of the defence in depth. Some situations that are considered as "beyond design" for existing reactors, such as multiple failures conditions and core melt accidents, are considered as "design basis" situations for new plants (see annex 2).

RHWG considers that the design improvements called by these safety objectives are at the same time demanding and reachable by the latest available industrial technology of power reactors.

The next two chapters present the work performed by RHWG to provide insights to drive implementation of the proposed qualitative safety objectives for new reactors. Chapter 7 investigates quantitative safety goals that could be used to harmonize expectations derived from these qualitative safety objectives, while chapter 8 provides examples of potential relevant areas of safety improvements expected from the design stage, either as part of the design process or as design features.

7. QUANTITATIVE SAFETY TARGETS TO DRIVE COMPLIANCE WITH THE PROPOSED SAFETY OBJECTIVES

The RHWG considers that there is merit for countries to use quantitative safety targets along with the proposed qualitative safety objectives. As *safety targets*, these values are useful to drive in-depth technical discussions with the applicants aimed at identifying real safety improvements, rather than being used as stand-alone *acceptance criteria*.

Candidate quantitative safety targets to drive compliance with the proposed safety objectives are discussed below. However, no consensus values were identified at this stage. The RHWG emphasises the need to be aware of differences in methodologies as well as terminology when making comparisons between numerical results in different countries.

Normal operation, abnormal events and prevention of accidents (O1)

Safety indicators on abnormal event occurrences are sometimes used for the supervision of operating nuclear power plants.

No reference numerical value having practical application for improving safety of new reactors as regards objective O1 was identified among WENRA countries. However, RHWG recommends European licensees to have their own ambitious quantitative safety targets[23] on the reliability of systems and components involved in normal operation.

The compliance with the qualitative safety objective O1 is expected to be appreciated through:

- the demonstration that all operational experience feedback has been used to identify the safety issues of existing plants that could be relevant for the envisaged new design;
- the verification that appropriately validated means have been designed to address these issues;
- the implementation of extended operational margins.

Accidents without core melt (O2)

❑ ***Reducing the core damage frequency***

WENRA countries already make a large use of level 1 PSA and widely refer to the core damage frequency (CDF) as a probabilistic safety target for currently operating plants. Some WENRA countries refer to a CDF target less than 10^{-5} per year for new reactors. This is in line with INSAG-12 recommendations, which state that the CDF target for new reactors should be reduced by a factor of at least ten compared to the target for existing ones (10^{-4} per year as recommended by INSAG), all plant states and all types of initiating events being taken into account.

However, two arguments were put forward not to adopt such a common target:

- in some counties, this value is considered as being already reached by some existing reactors;

- the methodologies to calculate the CDF may differ from one country to another.

[23] Not to be mistaken with a plant availability criterion for electricity production.

❑ ***No or only minor off-site radiological impact***

The results of the questionnaire mentioned in §4.3 show that a significant number of WENRA countries use dose / frequency criteria as design targets.

To achieve the objective O2, it is expected that off-site radiological impact of accidents without fuel melt is less than the intervention levels for iodine prophylaxis, sheltering and evacuation.

These intervention levels, which are used in the 5th level of the defence in depth, have already been enforced by EU members in their national regulation to comply with Directive 96/29/Euratom - 13 may 1996 – article 50.2., and are consistent with the ICRP recommendations. For instance, in ICRP-63, the intervention level for sheltering is 5-50 mSv in 2 days.

Design targets should be set below these intervention levels.

Accidents with core melt (O3)

❑ ***Practical elimination***

The possibility of certain accident conditions to occur can be considered as practically eliminated "*if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise*".[24]

As regards conditions that can not be physically excluded, it must be underlined that a justification for extreme unlikelihood has to be provided with high confidence. This means that the practical elimination of a condition cannot be claimed solely based on compliance with a general cut-off probabilistic value. Even if the probability of a condition is very low, any additional reasonable design features to lower the risk should be implemented.

The justification should include demonstration that there is sufficient knowledge of the accident condition analyzed and of the phenomena involved (e.g. DCH, steam explosion, hydrogen behaviour). Furthermore, uncertainties associated with the data and methods should be quantified.

❑ ***Limited protective measures in area and time***

Regarding radiological criteria associated with core melt accidents, a significant number of WENRA countries use release / frequency criteria. Some WENRA countries refer to Caesium release criteria in case of a severe accident. The aim of such criteria is to require that accidents have a limited impact on food consumption and land use. However, it is not easy to make a link between a relevant numerical value for Cs releases and the safety objective O3.

To achieve the objective O3, it is expected that the off-site radiological impact of accidents with core melt only leads to limited protective measures in area and time (no permanent relocation, no long term restrictions in food consumption, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering).

These protective measures are associated with intervention levels, which are used in the 5th level of the defence in depth. Such intervention levels have already been enforced by EU members in their national regulation to comply with Directive 96/29/Euratom - 13 may 1996 – article 50.2., and are consistent with the ICRP recommendations. For instance, in ICRP-63, the intervention level for sheltering is 5-50 mSv in 2 days.

Considering these intervention levels, design targets should be set so that only limited protective measures in area and time are needed. These design targets should take due account of the uncertainties associated with the use of best estimate methodologies for core melt accidents.

[24] IAEA document NS-G-1.10, para 6.5, footnote 14.

Independence between all levels of defence-in-depth (O4)

No relevant quantitative goal has been identified to drive compliance with this safety objective.

Safety and security interfaces (O5)

No relevant quantitative goal has been identified to drive compliance with this safety objective.

Radiation protection and waste management (O6)

- ❑ **Individual and c*ollective doses for workers***

The questionnaire mentioned in §4.3 gathered information on the use of individual and collective dose limits or targets in WENRA countries.

Regarding individual radiation dose limits, such limits are introduced in the 96/29 EU directive. For design purposes, RHWG considers that lower design targets should be set up at the design stage.

Regarding the average annual collective doses of workers, only Finland has published a quantitative target for new reactors. This target (0.5 man.Sv/GW) is technology neutral. However, the performance achievable in the field of radiation protection significantly depends on reactor technologies, even if based on the same optimization principles.

Hence, RHWG considers that to drive compliance with the qualitative safety objective on radiation protection, no single value for the average annual collective doses will be at the same time a reasonable and demanding safety target relevant for all reactor technology : it will be up to the licensee to justify the average annual collective doses reference value used from experience feedback of comparable operating plants and propose for its new reactor a reduced goal based on a comprehensive and ambitious optimization process.

- ❑ ***Radioactive and non radioactive discharges in the environment***

No relevant quantitative goal has been identified to drive compliance with this safety objective.

RHWG considers that to drive compliance with the qualitative safety objective on radioactive and non radioactive discharge, no single value will be at the same time a reasonable and demanding safety goal relevant for all reactor technologies: it will be up to the licensee to justify the discharge reference value used on the basis of experience feedback from comparable operating plants and to propose for its new reactor a reduced goal based on a comprehensive and ambitious optimization process.

- ❑ ***Quantity and activity of radioactive waste***

No relevant quantitative goal has been identified to drive compliance with this safety objective.

RHWG considers that to drive compliance with the qualitative safety objective on quantity and activity of radioactive waste, no single value will be at the same time a reasonable and demanding safety goal relevant for all reactor technologies: it will be up to the licensee to justify the reference value used on the basis of experience feedback from comparable operating plants and to propose for its new reactor a reduced goal based on a comprehensive and ambitious optimization process.

Management of safety (O7)

No relevant quantitative goal has been identified to drive compliance with this safety objective.

8. AREAS OF IMPROVEMENTS IN MEETING THE PROPOSED SAFETY OBJECTIVES

Starting from the proposed safety objectives, RHWG identified examples of areas of improvements compared to existing reactors that could form, after appropriate validation, the basis for a list of items to be either considered or taken into account at the design stage and duly assessed in the safety demonstration. Such examples are given for objectives O1 to O6 in annex 3.

As explained in section 4.4, it was not the task of the group to provide an exhaustive analysis, nor to perform a systematic investigation of the expected technical improvements. Examples of areas of improvements are listed on the basis of the information given in the documents reviewed (see §4.1 and annex 1), and after a group discussion. However, the development of a systematic list would require involvement of significantly greater resources.

The examples have been chosen to be, as far as possible, technology neutral and are not intended to be a prescribed list of safety improvements. It is for the designers to develop those improvements when meeting the objectives developed in this report. However, they may ultimately be used by the regulators to challenge applicants.

The main learnings of this exercise are the following:

- the examples given in the list are of various nature. Some of them are related to material or components, some other to operation, some other to the safety demonstration;
- some items of the list could also be relevant for existing reactors. However, they are expected to be dealt with in a better way for new reactors, since considered at the design stage;
- to properly illustrate the added value of the proposed safety objectives, in some areas, one may need to go into a greater level of details which was impossible to do as a pilot exercise;
- the group was not in a situation to fully check whether the examples given in the list are technology neutral. Furthermore, entering to a certain level of details appeared contradictory with staying technology neutral, which would cause some difficulties when going further in this exercise.

9. CHECK OF THE CURRENT SET OF REFERENCE LEVELS FOR ADAPTATION FOR NEW REACTORS

The reference levels developed by WENRA for the existing reactors are widely applicable also to new reactors.

However, as the practicability of safety improvements at design stage is greater than that for an operating plant, more stringent application of several of the reference levels is expected for new reactors.

In addition, there is room for safety improvements that go beyond the intent of the reference levels for existing reactors and which reflect the use of state-of-the art methodologies and techniques and the results of safety research.

To get a more precise picture of this general situation, and to identify those reference levels or issues that would have to be revised or completed to reflect the safety objectives for new reactors, an exercise has been performed by a subgroup to categorize each reference level published in 2008 into following groups:

A. Fully applicable (wording of the reference level does not need to be changed)

B. Applicable (wording does not need to be changed) but greater expectation (for instance, greater expectation on the practicability for new reactors)

C. More stringent description is necessary (wording needs to be changed)

The categorization was made without explicit criteria, on the basis of expert judgment.

In some cases the categorization between B and C was difficult to determine.

In addition, each issue was evaluated and identified the missing topics (signed by D).

At this stage, the RHWG has not yet been able to review this categorization.

The main results presented by the subgroup are the following.

- Almost all reference levels in issues H, LM, and P are considered as fully applicable also for new reactors.

- For issues A, B, C, D, G, I, J, K, N, O, Q and R, the applicability of several reference levels would need to be extended, either to refer to the license applicant (and not only to the licensee) or to the vendor organization and its subcontractors, and to cover other phases of the plant life cycle than operation (for instance, for issue K, requirements would need to be developed for inspection and testing in the pre-commissioning phase).

- Many reference levels in issues E, F, and to a less extend S, would need to be applied with greater expectations for new reactors, or even re-written.

- Missing topics were identified in issues D, E, F, I, K, N, O, R, S.

However, this is only a preliminary analysis not yet validated by RHWG as a whole.

10. CONCLUSIONS AND RECOMMENDATIONS ON THE USE OF THE PROPOSED SAFETY OBJECTIVES

According to the mandate given by WENRA, a pilot study on new reactors has been performed. In particular, safety objectives for new reactors have been proposed on the basis of the IAEA Fundamental Safety Principles and of a review of the existing relevant documentation.

These safety objectives are formulated as expected improvement compared to existing reactors, which is in line with the commitment of WENRA members to continuously improve safety. They are formulated in a qualitative manner, so that they can be more easily understandable by the public.

The RHWG also discussed some proposals for quantitative safety targets. However, no consensus values were identified at this stage. The need to be aware of differences in methodologies as well as terminology, when making comparisons between numerical results in different countries, was emphasised.

ANNEX 1

List of the documents reviewed

Wenra documents (www.wenra.org)

(1) RHWG Harmonization of Reactor Safety in WENRA Countries *Report by* WENRA Reactor Harmonization Working Group – 2008
(2) RHWG Probabilistic Safety Assessment Explanatory Note, Mar. - 2007

IAEA documents (www.iaea.org)

(3) IAEA Safety Standard Series No. SF-1 "Fundamental Safety Principles" (2006)
(4) IAEA Safety Standard Series No. NS-R-1 "Safety of Nuclear Power Plants: Design Safety Requirements" (2000)
(5) IAEA Safety Standards Series No. NS-G-10 "Design of Reactor Containment Systems for Nuclear Power Plants Safety Guide" (2004)
(6) IAEA General Conference Resolution - GC(XXXVI)/RES/553 1991
(7) IAEA Proceedings of the Conference "The safety of Nuclear Power. Strategy for the Future" (1991)
(8) IAEA TECDOC 682 "Objectives for the Development of Advanced NPP" (1993)
(9) IAEA TECDOC 712 "Safety aspects of design for future LWR (evolutionary design)" (1993)
(10) IAEA TECDOC 801 "Development of safety principles for the design of future nuclear power plants" (1995)
(11) IAEA TECDOC 905 "Approaches to Safety of Future Nuclear Power Plants" (1995)
(12) IAEA TECDOC 1362 "Guidance for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles: Report of Phase 1° of the International Project on Innovative Nuclear Reactors and Fuel Cycle (INPRO)" (2003)
(13) IAEA TECDOC 1366 "Considerations in the development of safety requirements for innovative reactors: application to modular high temperature gas cooled reactors" (2003)
(14) IAEA TECDOC 1570 "Proposal for a technology neutral safety approach for new reactor design" (2007)
(15) IAEA ES CS 94 "Safety aspects of design for future LWR (innovative design)" (1993)
(16) IAEA ES CS 14 "The IAEA Safety Standards for Design. Application to Small and Medium Size Reactors " (2002)
(17) INSAG 5 "The safety of Nuclear Power" (1992)
(18) INSAG 10 "Defense in depth in Nuclear Safety" (1996)
(19) INSAG 12 "Safety Principles for Nuclear Power Plants" (1999)

OECD NEA documents (www.nea.fr)

(20) NEA/CNRA/R(94)2 " A review for regulatory requirements for advanced nuclear power plans" 1994

(21) NEA/CSNI/R(2007): Proceedings of the Workshop on Future Control Station Designs and Human Performance Issues in Nuclear Power Plants - Halden, Norway 8-10 May 2006

(22) NEA/CSNI/R(2006): Proceedings of the Workshop on Better Nuclear Plant Maintenance: Improving Human and Organisational Performance, Canada 3-5 October 2005

(23) NEA/CSNI/R(2006): Draft Pilot Report on Approaches to the Resolution of Safety Issues

(24) NEA/CSNI/R(2007): Use and Development of Probabilistic Safety Assessment

(25) NEA/CNRA/R(1994)2: A Review for Regulatory Requirements for Advanced Nuclear Power Plants

(26) NEA/CSNI/R(2002): Passive System Reliability - A Challenge to reliability engineering and licensing of advanced nuclear power plants - Proceedings of an international work-shop hosted by the CEA 4-6 March 2002 Cadarache - France

(27) NEA/CSNI/R(1999): Fuel Safety Criteria Technical Review - Results of OECD / CSNI / PWG2 Task Force on Fuel Safety Criteria

(28) NEA/CSNI/R(1995): Summary and conclusions: Specialist Meeting on Severe Accident Management Implementation (1995 : Niantic, Conn.)

(29) NEA "The Regulatory Goal of Assuring Nuclear Safety", 2008

(30) NEA/CSNI WGRisk Task (2006) 2 "Probabilistic Risk Criteria"

European documents (ec.europa.eu/publications/)

(31) EUR 20163 EN "TSO study project on development of a common safety approach in the EU for Large Evolutionary pressurized water reactors" Oct. 2001

(32) EC Study Project of Safety Issues for Future Reactors in the frame of EC-RF Cooperation - Rev. a, 23.09.1997

(33) ISSN 1018 – 5593 Report EUR 16803 EN "1995 consensus document on safety of European LWR"

Utilities documents

(34) European Utilities Requirements for LWR nuclear power plants, Rev C, 2001 (www.europeanutilityrequirements.org)

(35) Electric Power Research Institute - Advanced Light Water Reactor - Utility Requirements Document (URD) Rep. EPRI NP- 6780, Palo Alto, CA (1990) (**www.epri.com**)

(36) TVO - Teollisuuden Voima Oyj General Presentation -1 October, 2008 Esa Mannola - Senior Vice President, Nuclear Engineering (**www.tvo.fi**)

Individual countries documents

(37) BNRA - Bulgarian Regulation on ensuring the safety of Nuclear Power Plants, 2004, Sofia (www.bnra.bg/en/documents-en/legislation /regulations)

(38) BNRA - Conditions of the permits for designing a nuclear facility – unit 1 of Belene NPP, 2007, issued by the Bulgarian Nuclear Regulatory Agency (www.bnra.bg/en/nuclear-facilitie-belene-licensing)

(39) STUK - Finnish Regulatory Guides on nuclear safety (YVL Guides), (http://www.stuk.fi/julkaisut_maaraykset/viranomaisohjeet/en_GB/yvl/)

(40) GPR/RSK common proposal for a safety approach for future pressurized water reactors. May 25, 1993 – adopted during the GPR/RSK common meeting on may 25,1993

(41) ASN - Technical Guidelines for the Design and Construction of the Next Generation of Nuclear Power Plants with Pressurized Water Reactors - adopted during the GPR/German experts plenary meetings held on October 19th and 26th, 2000 - sent by ASN to EDF on September 28th, 2004 (www.asn.fr/sites/default/files/files/technical_guidelines_design_construction.pdf)

(42) SKI Report, 2007-06 "Probabilistic Safety Goals"

(43) NKS - Nordic nuclear safety research – NKS 172 – Probabilistic safety goals, phase 2 – Status report

(44) HSE - Safety Assessment Principles for Nuclear Facilities, 2006 Edition, Revision 1, HSE, January 2008 (http://www.hse.gov.uk/nuclear/saps/saps2006.pdf)

(45) HSE - Nuclear Power Station Generic design Assessment - Guidance to Requesting parties. HSE August 2008 (http://www.hse.gov.uk/newreactors/ngn03.pdf)

(46) US NRC Secy – 90 – 016 "Evolutionary LWR Certification issues and their relationship to current regulatory requirements", 1990 (www.nrc.gov)

(47) CSNC - Canadian Draft Regulatory Document RD 337 "Design of New Nuclear Power Plants" (www.csnc-ccsn.gc.ca/eng/lawsregs/regulatorydocuments/published/rd337/)

ANNEX 2

Discussion on the evolution of the defence-in-depth approach for new nuclear power plants

This annex reflects the discussions among RHWG on the evolution of the defence in depth approach for nuclear power plants. The here under presented elements are to be considered as a contribution to the reflection on the topic, and are in no way a conclusive proposal.

1. Historical development of the defence-in-depth as regards currently operating reactors

The concept of "defence in depth" (DiD) was present in the design of nuclear power plants since their inception. This concept was gradually refined to constitute an increasingly effective approach combining both prevention of a wide range of postulated incidents and accidents and mitigation of their consequences. Incidents and accidents were postulated on the basis of single initiating events selected according to the order of magnitude of their frequency, estimated from general industrial experience.

The definitions of the levels were set as to mirror accident escalation: shall one level fail, the next level comes into force. The approach was intended to provide redundant means to ensure the fulfilment of the basic safety functions of controlling the criticality, cooling the fuel and confining radioactive material. In the early stage, the concept of defence in depth included three levels:

	Level of defence in depth	Objective of the level	Essential means	Associated Plant condition categories
Original design of the plant	Level 1	Prevention of abnormal operation and failure	Conservative design and high quality in construction and operation	Normal operation
	Level 2	Control of abnormal operation and failure	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences
	Level 3	Control of accident within the design basis	Engineered safety features and accident procedures	Design basis accidents (postulated single initiating events)

Then, the concept of defence in depth for the current operating reactors was further developed to take into account severe plant conditions that were not explicitly addressed in the original design (hence called "beyond design conditions"), in particular lessons learned from the development of probabilistic safety assessment (PSA) and from the Three Mile Island accident (USA 1979) which led to a severe core melt accident and from the Chernobyl accident (Ukrainian Republic of USSR 1986). At this stage of development of the defence in depth concept, two additional levels were added (see INSAG 10 – 1996):

	Level of defence in depth	Objective of the level	Essential means	Associated Plant condition categories
Original design of the plant	Level 1	Prevention of abnormal operation and failure	Conservative design and high quality in construction and operation	Normal operation
	Level 2	Control of abnormal operation and failure	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences
	Level 3	Control of accident within the design basis	Engineered safety features and accident procedures	Design basis accidents (postulated single initiating events)
Beyond design situations	Level 4	Control of severe plant conditions that were not explicitly addressed in the original design of currently operating plants owing to their very low probabilities	Complementary measures and accident management	Multiple failures Severe accidents
Emergency planning	Level 5	Mitigation of radiological consequences of significant releases of radioactives materials	Off-site emergency response	-

In SF-1 published in 2006, the IAEA stressed that independency effectiveness of the different levels of defence is a necessary element of DiD concept.

2. New reactors design and associated evolution of the defence-in-depth concept

For new reactors, there is a clear expectation, and not only an opportunity, to address in the original design what was "beyond design" for the previous generation of reactors, such as multiple failures situations and core melt accidents.

This is a major evolution in the list of postulated initiating events considered in the initial design to prevent accidents, control them and mitigate their consequences, and in the corresponding design features of the plant.

It implies that the meaning of "beyond design basis accident" is not the same for existing reactors and for new reactors. Several scenarios that are considered beyond design basis for existing reactors are now included in the design basis for new reactors (multiple failures accidents, core melt accidents…), even if the safety assessment rules may vary depending on the kind of accident considered.

Furthermore, for the existing plants, the defence-in-depth was mainly considering the nuclear fuel when loaded in the reactor vessel. For new reactors, the scope of the defence-in-depth has to cover all risks induced by the nuclear fuel, even when stored in the fuel pool.

The proposed revised structure of the levels of DiD discussed by RHWG is as follows:

<table>
<tr><th></th><th>Level of defence in depth</th><th>Objective of the level</th><th>Essential means</th><th>Associated plant condition categories</th><th>Radiological consequences</th></tr>
<tr><td rowspan="5">Original design of the plant</td><td>Level 1</td><td>Prevention of abnormal operation and failure</td><td>Conservative design and high quality in construction and operation</td><td>Normal operation</td><td>Regulatory operating limits for discharge</td></tr>
<tr><td>Level 2</td><td>Control of abnormal operation and failure</td><td>Control, limiting and protection systems and other surveillance features</td><td>Anticipated operational occurrences</td><td>Regulatory operating limits for discharge</td></tr>
<tr><td rowspan="2">Level 3 (1)</td><td>Control of accident to limit radiological releases and prevent escalation to core damage conditions (2)</td><td>Safety systems

Accident procedures</td><td>DiD Level 3.a

Postulated single initiating events</td><td rowspan="2">No off-site radiological impact or only minor radiological impact (see NS-G-1.2/4.102)</td></tr>
<tr><td>Control of accident to limit radiological releases and prevent escalation to core melt conditions (3)</td><td>Engineered safety features (4)

Accident procedures</td><td>DiD Level 3.b

Selected multiples failures events including possible failure or inefficiency of safety systems involved in DiD level 3.a</td></tr>
<tr><td>Level 4</td><td>Practical elimination of situation that could lead to early or large releases of radioactive materials

Control of accidents with core melt to limit off-site releases</td><td>Engineered safety features to mitigate core melt

Management of accidents with core melt (severe accidents)</td><td>Postulated core melt accidents (short and long term)</td><td>Limited protective measures in area and time</td></tr>
<tr><td>Emergency planning</td><td>Level 5</td><td>Mitigation of radiological consequences of significant releases of radioactives materials</td><td>Off-site emergency response
Intervention levels</td><td>-</td><td>Off site radiological impact necessitating protective measures</td></tr>
</table>

(1) Even though no new safety level of defence is suggested, a clear distinction between means and conditions is lined out

(2) Accident conditions being now considered at DiD Level 3 are broader than those for existing reactors as they now include some of the accidents that were previously considered as "beyond design" (3b). However, acceptance criteria for level 3a are not relinquished compared to those required in level 3 for currently operating reactors. For instance, pin integrity is required for the most frequent conditions.

(3) Acceptance criteria have to be defined according to a graded approach, based on probability of occurrence.

(4) Highest safety requirements should be imposed for safety system used for 3a. Requirements for systems used for 3b may be not as stringent as for 3a if appropriately justified.

3. Rationale for the updated defence-in-depth concept

3.1. Consideration on multiple failures situations on level 3

3.1.1. Multiple failures to be addressed by the original design of the plant

The multiple failures to be considered at the design stage should be identified in an iterative process starting with a first list of multiple failures based on the postulated complete loss of safety systems needed to control a postulated initiating event or combination of postulated initiating events. This first list has then to be adapted through experience feedback and the use of PSA.
Safety assessment of the conditions resulting from the selected multiple failures shall be performed deterministically in order to design additional features that aim at preventing core damage conditions. The appropriateness of the foreseen additional design features has to be checked by PSA insights.

Example of multiple failures situations are:

- anticipated transients without scram;
- station blackout;
- total loss of feedwater;
- small break loss of coolant accident and loss of the medium head safety injection trains or of the low head safety injection system;
- small break loss of coolant accident and simultaneous loss of the component cooling water system/essential service water;
- total loss of the spent fuel pool cooling system;
- etc.

While the postulated single initiating events analysis in combination with the single failure criteria gives credit on redundancy in design provisions of safety systems and of their support functions, addressing multiple failures situations emphasises more on diversity in the design provisions of the third level of DiD.

3.1.2. Multiple failures conditions: 3rd or 4th level of DiD?

In the DiD approach, the objective of the different levels of defence are defined as successive steps in escalation of accident situations.

The phenomena involved in accidents with core melt (severe accidents) differ radically from those which do not involve a core melt. Therefore core melt accidents should be treated in a specific level of defence in depth.

In addition, for new reactors, design features that aim at preventing a core melt accident should not belong to the same line of defence as the design features that aim at controlling a core melt accident that was not prevented.

The question has been discussed by RHWG whether for multiple failure events, a new level of defence should be defined, because the safety systems which are needed to control the postulated single initiating events fail and thus another level of defence should take over. However, the single initiating events and multiple failures analysis are two complementary approaches that share the same objective: controlling accidents to prevent their escalation to core melt accidents.

Hence, at this stage of the discussion, it has been proposed to treat the multiple failures conditions as part of the 3rd level of DiD, but with a clear distinction between means and conditions (sub levels 3a and 3b).

3.2. Considerations on practically eliminated situations (level 4)

As stated in IAEA safety standard NS-G-1.10, for new reactors, the design should aim at practically eliminating the following conditions:

- Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;
- Severe accident conditions that could damage the containment in a late phase as a result of basemat melt-through or containment over-pressurization;
- Severe accident conditions with an open containment — notably in shutdown states;
- Severe accident conditions with containment bypass, such as conditions relating to the rupture of a steam generator
  tube or an interfacing system LOCA.

The possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if design provisions have been taken to design them out so that they can be considered to be extremely unlikely to arise with a high degree of confidence.

In this approach, each condition to be practically eliminated has to be assessed separately, taking into account the uncertainties due to the limited knowledge on some physical phenomena, and cannot be considered practically eliminated only on the basis of the compliance with a general "cut-off" probabilistic value. Even if the probability of occurrence of the condition is very low, if some reasonably practicable additional design features can still be implemented to lower the risk, then those design features shall be implemented.

ANNEX 3

Examples of areas of improvements in meeting the safety objectives

For objectives O1 to O6, examples of areas of improvement in meeting the safety objectives are given.

Normal operation, abnormal events and prevention of accidents (O1)

According to the purpose of *reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation*, the following areas for safety improvement can be highlighted:

- Use of advanced materials and manufacturing technologies in order to reduce the frequency of failures;
- More comprehensive identification of ageing mechanisms and effective implementation of ageing management programs from the design stage;
- Larger operational margins based on design provisions in order to reduce the frequency of abnormal events;
- More comprehensive identification of initiators using operating experience and insights from PSA, including initiators originating out of the plant;
- Strengthen human factors engineering (through experience feedback and testing) to improve man-machine interface as regards human failures prevention;
- Design provisions intending to improve in-service inspections, testing and aging monitoring.

According to the purpose of *reducing the potential for escalation to accident situations by enhancing the capability to control abnormal events*, the following areas for technical improvement can be highlighted:

- Increased use of limitation systems in order to avoid unnecessary initiation of protection systems;
- Improved man-machine interface as regards information and diagnostic provided to operators.

Accidents without core melt (O2)

According to the purposes of:

- *ensuring that accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation).*
- *reducing, as far as reasonably achievable,*
    - *the core damage frequency taking into account all types of hazards and failures and combinations of events;*
    - *the releases of radioactive material from all sources.*
- *providing due consideration to siting and design to reduce the impact of all external hazards and malevolent acts,*

the following areas for technical improvement can be highlighted:

- More systematic consideration of initiating events and hazards in all reactor states;
- More systematic consideration of initiating events related to ex-core sources of radioactivity (including waste storage, tank, spent fuel storage...) ;
- More systematic consideration of multiple failure situations;

- Use of PSA at the design stage in order to:
  - o check that the CDF and radiological consequences are indeed reduced;
  - o identify complementary design provisions where needed;
  - o identify where diversity is needed in the design of safety systems, in complement to redundancy;
- Reduction of human-induced failures through:
  - o more automatic or passive safety systems;
  - o longer grace period for operators;
  - o improved man-machine interface;
- Use of improved materials such as thermal insulation materials, to reduce the clogging phenomena on the sump filters.

<u>Accidents with core melt (O3)</u>

According to the purpose of *reducing potential radioactive releases to the environment from accidents with core melt, also in the long term, by following the qualitative criteria below:*

- *accidents with core melt which would lead to early or large releases have to be practically eliminated ;*
- *for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures,*

the following areas for technical improvement can be highlighted:

- Considering the different possible failures of this function for accidents with core melt, substantial design improvements of the containment function such as:
  - o effective reduction of loads on the containment arising from severe accidents situations and/or increased resistance of the containment to such loads
  - o leaktightness of the containment in case of severe accident, including in the long term
  - o provisions to avoid penetration of the corium through the containment basemat
  - o systematic review and suppression of potential containment by-passes
- Use of PSA for verifying that safety objectives are met or to identify complementary design provisions where needed
- Use of improved materials:
  - o for PWR steam generators, to reduce the probability of tube failures during core melt, before core relocation,
  - o for vessel internals and reactor cavity (sacrificial and basemat materials) to take the core melt phenomena into account.

<u>Independence between all levels of defence-in-depth (O4)</u>

According to the purpose of *enhancing the independence between all levels of DiD, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous four objectives) to provide, as far as reasonably achievable, an overall reinforcement of DiD*, the following area for improvement can be highlighted:

- Use of dedicated systems to deal with core melt accidents, so that the independence of the 4th level of the DiD is better ensured.

Considering safety and security interfaces (O5)

According to the purposes of *ensuring that safety measures and security measures are designed and implemented in an integrated manner and of seeking synergies between safety and security enhancements,* the following area for improvement can be highlighted:

- Aircraft crash protection against large civil airplanes

Radiation protection and waste management (O6)

According to the purposes of *reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities, individual and collective doses for workers*, the following area for improvement can be highlighted:

- Improved fuel cladding integrity to avoid release of fission products;
- Minimization of the production and buildup of radionuclides by careful selection of materials and appropriate chemistry control to :
    - Reduce activation;
    - Minimize the spread of activated corrosion products,
    - Ease surface decontamination of components;
- Improved reliability of the systems and components which are currently or are expected to be the major contributors to worker exposure;
- Extensive use of quickly removable and reusable thermal insulation;
- Improved components and system design to minimize the number of welds to be inspected in high dose rates and to avoid corrosion products deposits (traps, pockets);
- Optimization of the plant layout with regard to radiological conditions (dose rate and contamination) by considering:
    - Appropriate shielding of location where workers are daily working or where workers would be required to work in the event of an accident;
    - Workers access needs, taking into account jobs (maintenance, periodic testing, in service inspection) to be performed and access control rules;
    - Improved accessibility to components, including for future decommissioning or component replacement;
- More systematic use of remote handling or control or operating technologies, including for in service inspection.

According to the purposes of *reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities, radioactive and non radioactive discharges to the environment*, the following area for improvement can be highlighted:

- Minimization of the use of hazardous substances;
- Minimization of the production and buildup of radionuclides by careful selection of materials and appropriate chemistry control
- Use of best available technologies to collect, treat and discharge liquid and gaseous effluents ;
- Provisions to allow sufficient time for radioactive decay of short lived radionuclides;
- Filtration/purification (mechanical, ion exchange, activated carbon filter, centrifugation, evaporation…) to reduce significantly the toxicity of the effluents.

According to the purposes *of reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities, quantity and activity of radioactive waste*, the following area for improvement can be highlighted:

- Minimization of the use of hazardous substances
- Minimization of the production and buildup of radionuclides by careful selection of materials and appropriate chemistry control :
    - Reduce activation;
    - Minimize the spread of activated corrosion products,
    - Ease surface decontamination of components;
- Improved the plant layout to
    - Have appropriate rooms for the collection, sorting, handling, packaging and measurement of radioactive waste;
    - Facilitate control of radioactive material use and storage within the plant;
    - Increase possibilities to decontaminate components;
- Verifying at the design stage that radioactive waste to be produced are compatible with the requirement for final disposal.

ANNEX 4

Categorization of the Reference Levels

This table shows the preliminary results of the exercise issue by issue, not yet validated.

		A	B	C	D
Issue	**Number of RLs**	**Fully applicable**	**Applicable but greater expectation**	**More stringent description is necessary**	**Identified Missing topics**
A: Safety Policy	8	2	1	5	0
B: Operating Organisation	15	4	0	11	0
C: Management System	23	11	0	12	0
D: Training and Authorization of NPP staff	15	10	0	5	1
E: Design Basis Envelope for Existing Reactors	44	25	17	2	3
F: Design Extension of Existing Reactors	12	4	8	0	1
G: Safety Classification of Structures, Systems and Components	7	4	2	1	0
H: Operational Limits and Conditions	19	18	0	1	0
I: Ageing Management	8	3	4	1	1
J: System for Investigation of Events and Operational Experience Feedback	16	7	6	3	0
K: Maintenance, In-service inspection and Functional Testing	20	14	4	2	4
LM: Emergency Operating Procedures and Severe Accident Management Guidelines	14	14	0	0	0
N: Contents and updating of Safety Analysis Report	16	13	3	0	1
O: Probabilistic Safety Analysis	16	15	0	1	1
P: Periodic Safety Review	9	8	1	0	0
Q: Plant Modifications	15	11	4	0	0
R: On-site Emergency Preparedness	18	14	2	2	2
S: Protection against Internal Fires	20	16	4	0	1
	295	**193**	**56**	**46**	**15**

Western European
WENRA
Nuclear Regulators Association

WENRA Report on Benchmarking the European inspection practices

March 2012

Western European

WENRA

Nuclear Regulator's Association

Benchmarking the European inspection practices for components and structures of nuclear facilities

Study by

WENRA Inspection Group

March 2012

March 2, 2012

WIG

WENRA Inspection Group

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March 2, 2012

WIG

WENRA Inspection Group

Benchmarking the European inspection practices for components and structures of nuclear facilities

1. Introduction

WENRA decided to initiate work concerning benchmarking of European inspection practices for components and structures of nuclear facilities in its March 2010 meeting where the chair of WENRA presented a proposal "to benchmark European regulatory practices for verifying design and quality of the NPP structures and components both at new and existing reactors". According to the proposal the first goal was getting clear and reliable information on what are the various inspection practices that are applied in member countries to verify that the structures and components are designed, constructed/manufactured, installed and commissioned to meet their design and quality requirements. In the second phase, harmonisation needs and possibilities in the area of inspection practices should be studied. A WENRA task group (WENRA Inspection Group, WIG) was established to perform the benchmark study.

It was noted that MDEP has inspection co-operation in its vendor inspection working group but it is more focused to find out how inspectors can make use of the work done by inspectors of other regulators. The joint inspections performed in that working group are mainly quality system oriented. The intention was that the work of this new WENRA group would focus on technical issues.

Finland volunteered as the lead country for the task group. It was concluded in the WENRA meeting that a letter including a concrete proposal for the work including a tentative work plan and an invitation to send a representative to the group will be sent by Finland to all WENRA members and observers.

The original work plan of the group was the following

1. STUK makes a proposal for contents of national input reports that give a picture of the national inspection practices in the respective WENRA countries.
2. All WENRA members are asked to nominate their representative to the working group.
3. STUK starts preparing its own national report. That is aimed to be an example of how the information could be provided in the report. STUK's target was to circulate the draft of its national report before the end of May 2010.
4. The national input reports should be provided before the end of August 2010.
5. The working group meets for the first time in September 2010.

STUK prepared a list of questions and provided a model for contents of national reports to other participants in early July 2010. National reports were provided to STUK in early September according to the request. In this phase eleven WENRA countries (Belgium, Bulgaria, Finland, France, Hungary, Lithuania, Slovak Republic, Spain, Sweden, Switzerland and UK) prepared national reports of their regulatory inspection practices.

The purpose of the national reports was to establish the basis of the work of the WIG. The first step was to get clear and reliable information on what are the various inspection practices that are applied in WENRA countries. The national reports were prepared to describe briefly the main regulatory inspection practices without going into great depth on the technical details of different types of structures and components.

After the delivery of the national reports the first working group meeting was held on 22-24 September 2010 in Helsinki to clarify the contents of national reports. The national reports were used to select issues for discussion in the meeting and to perform initial comparisons between the participating countries. However, it was noticed that the national reports were so heterogenious that they could not be used for detailed benchmarking.

Therefore, in the meeting in Helsinki, tables were developed to gather information concerning inspection practices. In the tables the activities required by the regulations to be performed by the licensees, inspection bodies and the regulatory bodies were presented related to the lifecycle of the components or structure to enable comparisons between countries. The completed tables were submitted by Belgium, Bulgaria, Czech Republic, Finland, France, Lithuania, Slovak Republic, Spain, Sweden, Switzerland and the UK.

The results of the first meeting were presented to WENRA in its Bratislava meeting in November 2010. In that presentation some typical national inspection practices based on the national reports were presented. Also first ideas concerning basic regulatory approaches and issues for good practices were presented.

The WIG had its second meeting in Bootle, UK in February 2011. In the meeting the group discussed the summary of tables completed by the participating countries and made initial conclusions concerning different national practices. Also the basic regulatory approaches and good practices were further discussed. The content of the final report of the work was agreed and a sub-group was established to write chapter 2 “Basic regulatory approaches for inspection of components and structures” and chapter 4 “Good practices for inspection of components and structures” of this final report of the group. Sweden accepted leadership of the sub-group. The sub-group had a meeting in Stockholm in early June. The other participating countries of the sub-group were Finland, France and UK.

For the final report the participating countries agreed to write short (some pages) national summaries of their inspection practices. The content of these summaries was agreed. These national summaries contributed to the chapter 3 “Benchmarking of the national practices” of this final report. The national summaries were provided by the same countries that provided the completed tables with the addition of Russia. The national summaries are attached to this report.

The final meeting of the group was in Helsinki in September 2011 where the draft report of the benchmark was reviewed by the group.

2. Basic regulatory approaches for inspection of components and structures

Basic roles and responsibilities within the nuclear field are clearly defined and accepted by all concerned. The European Nuclear Safety Directive[1] requires that Member States shall ensure that the prime responsibility for the safety of a nuclear installation rests with the license holder. This responsibility cannot be delegated. The Directive also requires that Member States shall establish and maintain a competent regulatory body in the field of nuclear safety. The regulatory body shall have the powers and resources to verify compliance with national nuclear safety requirements and the terms of the relevant license through regulatory assessments and inspections.

Regulatory bodies are responsible for finding effective and efficient[2] approaches for their regulatory work including assessment and inspection activities. Finding effective and efficient approaches is a difficult task, and will also depend on the national regulatory regime. Regulators need, for example, to establish a clear boundary between regulatory responsibilities for safety and licensee's responsibilities for safety. In selecting approaches regulators consider not only how a strategy may affect safety directly, but also possible indirect effects. Indirect effects may include such things as impacts on resources for the regulator or changes to the safety culture of the licensee. Regulators also have to reassess and adjust approaches to respond to legal, economic and technological changes.

Approaches applied have been discussed among regulatory bodies and by researchers. In an exploratory study[3] that the Swedish Nuclear Inspectorate (SKI) conducted between 2003 and 2005, the use of six different regulatory approaches for oversight of commercial nuclear power plants were studied and compared: prescriptive, case-based, goal-setting or outcome-based, risk- or hazard-informed, process-based, and self-assessment approaches. One main finding regarding the experiences of using different regulatory approaches was that regulators tend to use combinations of at least two, often three and at times four different approaches for specific examples of oversight issues.

In the area of systems, structures and components (SSC) there is a tradition in many countries to apply approaches which focus on the prescriptive element while regulators in other countries use combinations which focus on goal-setting or outcome-based approaches.

In a prescriptive approach the regulator establishes relatively detailed requirements for functions and properties of systems, components and structures in a plant. A prescriptive approach can also include relatively detailed requirements for conducting specific activities.

In a goal-setting, non-prescriptive approach the regulator establishes specific goals or outcomes for licensees to attain but does not specify how licensees attain these goals.

In the area of SSC, and in particular pressurized components, there is also a tradition in many countries to use independent inspection organizations (IO) and other conformity assessment bodies, to review, assess and supervise different activities during design, manufacture, construction and commissioning. The use of such conformity assessment bodies can be prescribed by the regulator and contracted for its task by either the regulator or by the licensee or the vendor. In other countries the regulator does not prescribe the use of independent conformity assessment

[1] COUNCIL DIRECTIVE 2009/71/EURATOM of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations

[2] In many instances these terms are interchanged quite freely, but in essence have quite different meanings. The Committee on Nuclear Regulatory Activities (CNRA) of the OECD Nuclear Energy Agency (NEA) has in its Guidance Book "Improving Nuclear Regulation" agreed that regulatory effectiveness means "to do the right work", whereas regulatory efficiency means "to do the work right".

[3] Regulatory Strategies in Nuclear Power Oversight, SKI Report 2005:37.

bodies but expresses expectations that the licensee contracts conformity assessment bodies for review, assessment or supervision of important aspects during design, manufacture, construction and commissioning.

Regulatory bodies can thus choose to have an emphasis in either of the two basic approaches, prescriptive and goal-setting/outcome based, for inspection of SSCs or to combine them in an appropriate manner and to differing degrees use independent conformity assessment bodies as part of the work. It should be noted that the use of independent conformity assessment bodies by its nature requires a major element of prescription in using a contract and specification to define the scope and extent of the work expected, particularly when the purpose is to assess conformity with regulations and other requirements.

3. Benchmarking of practices

3.1 Introduction

The working group was expected to describe the practices being applied in each WENRA country for arranging inspection practices of mechanical equipment, steel structures and concrete structures. The countries were to

- explain possible formal approvals of the involved third party organizations
- explain possible correlation with safety classes
- explain if two redundant inspections have to be done where responsibility is with the licensee and with the regulatory body.

As a second step of the work, the group was expected to

- discuss the satisfaction of each WENRA country with their respective approach and the possible needs/plans to modify the approach
- consider good practices that could become harmonized European practices.

The scope of the benchmarking was defined so that it covers the various review and inspection practices arranged in WENRA countries concerning

1. Mechanical components
2. Steel structures
3. Concrete structures

to provide adequate assurance that the components and structures are

- designed
- manufactured/constructed
- installed and
- commissioned

to meet their respective design and quality requirements.

Pre-service and in-service inspections and testing (non destructive testing) were left outside the scope of this study. Pre-service inspections and testing take place during commissioning and their successful performance is one of the prerequisites for starting the operation of the systems and components.

The working group defined the following additional objectives for the work
- learn from others practices to develop your own practices (according to WENRA ToR)
- discuss the added value of different basic regulatory approaches
- assure similar degree of involvement by the industry
- make use of foreign IOs easier in the long term (interchangeability, accreditation)

Because the original national reports could not be used for detailed benchmarking, the group developed, in its first meeting, tables to collect information on inspection practices in a systematic way. The tables were created for pressure equipment, steel structures and concrete structures. Regulatory and licensee inspections and auditing were filled in separate tables. The table for pressure equipment is presented as an example in the following Figure 1.

Country:

Pressure equipment life cycle																
	Design		Hold point		Manufacturing		Hold point		Installation		Hold point		Commissioning		Hold point	
Safety class	Who	How	P	O	Who	How	P	O	Who	How	P	O	Who	How	P	O

Who does?

RB Regulatory body
L Licensee
IO Inspection organization
UI Utilities inspection organization
CI Contractor insp. organization
MO Mandated organization (Belgium)

How it does? What is the completeness of the activity?

C Comprehensive technical control
T Technical control by sampling
M Management system audit
F Focused audit (e.g. follow up of a specified product)
R Reactive intervention by exception

Hold point:
P Predefined
O Optional

Figure 1. Table to collect information on inspection practices, example related to pressure equipment.

Comprehensive technical control (C) refers to a practice where all relevant technical aspects are reviewed or inspected and the control covers a major proportion of structures and components.

If the control is based on sampling (T), a combination of different factors is typically used to define sample size and scope. Factors which are used as a basis may include safety significance, novelty, complexity and operating experience of the structure or component. Sample size is typically increased if issues are identified.

The completed national tables are not quite uniform. Main differences are as follows:

- Terminology of different kinds of inspection organizations (Inspection Organization, Utilities Inspection Organization, and Contractor's/Manufacturer's Inspection Organization as well as Third-Party Organization) varies and makes it difficult to ensure that inspection and auditing practices are understood in a consistent way in all countries.

- Safety classification differs from country to country. In most of the countries, for pressurized systems the primary circuit belongs to safety class (SC) 1, engineered safety features to SC 2 and other safety-related systems to SC 3. For instance, in some countries the highest (most important) safety class of concrete and steel structures is 1 while in some other countries the corresponding structures belong to safety class 2. Two countries are using four safety classes; one five. In spite of these differences, it was decided to use three safety classes in the detailed comparison (tables).

- Many countries have not reported their practices relating to the class non-nuclear safety (NNS) equipment and structures. It was decided to delete class NNS from the tables because the results could have been misleading.

A target in completing these tables was to cover both licensee and regulatory inspections and auditing. It seems that in general a system exists where the licensees first make their inspection and then the regulators (or their IOs). It also seems that the inspections by the licensees generally bound the scope of the inspections by the regulators.

However, some countries have included in their responses for licensee activities the inspections and auditing performed by contractors/manufacturers, too. Possibly this is done for the reason that the licensees rely on the contractor/manufacturer inspections and auditing in such a way that equipment/structures/organizations are not inspected/audited by the licensees themselves (and the regulators do not require this).

In all NPP projects the contractors/manufacturers perform anyway their own quality control (QC) activities and make the results available to the licensees and via the licensees to the regulators. A question rises whether it is necessary that in all cases (at least for the safety related items) the licensee conducts its own inspections.

The licensee has to satisfy itself that the component or structure meets the respective design and quality requirements and that adequate inspection is done by the licensee or on behalf of the licensee to verify this. These inspections have to be defined in the quality management (QM) system of the licensee which should be audited by the regulatory body. The responsibilities and duties of the licensee are discussed in chapter 4.1 “Licensees’ control, supervision and oversight”.

3.2 Comparison of the national tables

The summary tables were created both for licensee and regulatory inspections and auditing. An initial comparison of the completed tables was provided by STUK for the Bootle meeting of the group. In this comparison, information from the national tables was extracted and restructured. Summary tables were reformulated to have the information commensurable for benchmarking and to make comparisons easier. The summary tables for regulatory inspections and auditing are presented in appendix 3.

The summary tables combine the information from all the participating countries and are based on the following phases for pressure equipment and steel structures:

- design (design basis/detailed design)
- manufacturing
- installation and
- commissioning.

For concrete structures the following phases are used:

- design (design basis/detailed design)
- structural concreting
- commissioning.

An example of a summary table on regulatory inspections and auditing relating to design phase of pressure equipment is presented in the Figure 2. Mandatory inspections and auditing are underlined.

Safety Class	Belgium	Bulga-ria[i]	Czech Republic	Finland	France	Lithua-nia	Slovak Republic	Spain	Sweden	Switzer-land	UK
1 (Who)	RB/MO	RB/-	RB/RB	RB/RB	RB/RB, IO	RB/RB	RB,IO/RB	RB/RB	RB/IO	RB/RB+IO	RB/IO
(How)	T/C	C/-	CM/CM	CMR/CMR	CM/CMR	C/T	CM/CM	CM,TFR/C,TFR	CM/C	C/C	TM/CMRD
2 (Who)	RB/MO	RB/-	RB/RB	RB/RB,IO	RB/IO	RB/RB	RB,IO/RB	RB/RB	RB/IO	RB/RB+IO	RB/IO
(How)	T/C	C/-	CM/CM	CMR/CMR, CR	TM/CTMR	C/T	CM/CM	CM,TFR/C,TFR	TM/C	C/C	TM/CMRD
3 (Who)	RB/MO or IO	RB/-	RB/RB	RB/IO	RB/IO	RB/RB	RB,IO/RB	RB/RB	RB/IO	RB/RB+IO	RB/IO
(How)	T/C	C/-	CM/CM	CR/CR	TM/CTMR	C/T	CM/CM	CM,TFR/C,TFR	TM/C	C/C	TM/CMRD

Figure 2. Summary table on regulatory inspections, example related to design phase (design basis/detailed design) of pressure equipment.

Design basis of components and structures refers to those technical requirements which have to be set on components and structures in order that they would meet the demands based on plant and system level design. Design basis includes functional requirements and loading conditions for normal and accident conditions. Also safety, seismic and quality classification are part of component or structure level design basis.

The practices across countries are fairly uniform for pressure equipment although, for regulatory practices, in this area there are variations between countries concerning especially management system audits and/or focused audits. As concerns the design and commissioning phases comprehensive technical control (C) is performed by almost all regulators. For the manufacturing and installation phases there seem to be more variations between regulators. About the same number of countries uses comprehensive technical control (C) or technical control by sampling (T) in these phases. In all the phases about half of the countries perform management system audits (M) or focused audits (F). Auditing of the QM Systems of most important manufacturers of pressure equipment by the regulatory body or in some cases by an inspection body is considered important. As concerns the design and manufacturing of the class NNS (non-nuclear safety) pressure equipment, there seem to be different inspection practices despite of the EU Directive. This observation may however be due to information being missed in the national tables.

As concerns steel and concrete structures there are larger variations in inspection and auditing practices between regulators. Also for steel and concrete structures the design is reviewed by most regulators using comprehensive technical control (C). As concerns manufacturing and structural concreting typically technical control by sampling (T) is used whereas for installation and commissioning either comprehensive technical control or technical control by sampling is used. About half of the countries perform management system audits or focused audits.

Hold points are used widely, especially for design and commissioning phases. The use of hold points is related to the national inspection approach.

Reactive intervention by exception (R) is probably made by all countries although this is not shown in the tables. This may be due to different interpretations of what to include in the tables.

3.3 Basic regulatory approach (structures and components)

A summary of regulatory oversight practices during different phases is presented in the Appendix 3, where the comparison of the completed national tables is presented phase by phase. Every participating country has regulatory oversight activities in design, manufacturing, installation and commissioning phases although the extent and focus of the inspections and supervision varies depending on the country and the phase.

As presented in chapter 2, the regulator can choose to have an emphasis in either of the two basic approaches (prescriptive or goal-setting) for inspection of SSC or to combine them in an appropriate manner. Some of the countries seem to emphasize prescriptive approach and some of the countries goal-setting approach. The regulatory approach can also vary from phase to phase. Prescriptive approach is typically emphasized in design and commissioning phases. Some regulators also use independent conformity assessment bodies to varying degrees as part of the work.

3.4 Expectations on licensees

Every participating country confirms that primary responsibility for the safety of NPPs and quality of NPP structures and components rests with the license holder. The licensee reviews and approves documents and inspections related to structure and component design, manufacturing, installation and commissioning before presenting them to the regulator for approval or to the IO for conformity assessment. Licensees are expected to take all the necessary steps and actions to fulfil applicable safety requirements and to organize quality control related to design, manufacturing, installation and commissioning of structures and components. Licensees shall verify that all the organizations related to these steps have arrangements to produce appropriate quality; in other words they have recognised quality management systems and qualified personnel.

Regarding non-conformances about half of the countries state that it is required that the licensee's management system includes procedures to process non-conformances. Most of the countries expect that licensees process non-conformances by assessing their significance, identifying the reasons for them and taking corrective and preventive actions. If the above mentioned actions are not required (or they don't show from the national summary report), it is at least required that the regulator is informed about serious non-conformances.

3.5 Regulatory inspections and correlation with safety classes

Regulatory inspections performed using comprehensive technical control (C) in different safety classes and whether they are performed by the regulator itself or are delegated to IOs are presented in the graphs in Figures 3 to 5. The graphs are composed for pressure equipment, steel structures and concrete structures on the basis of national summaries and completed tables. When looking at the graphs for steel and concrete structures it must be kept in mind that concrete and steel structures are not classified equally in the participating countries; in some countries the highest safety class is 1 while in some other countries respective structures belong to safety class 2. Also the number of safety classes differs from 1 to 3.

In the graphs, where inspections made by IOs are presented, combined inspections are shown, which are conducted:

- solely by the IO or
- by either the IO or the regulatory body (RB/IO).

The latter one (inspections made by the IO or the regulatory body) might be due to

- different safety significances of the components in the safety class in question such that the regulatory body is liable for certain components or structures which are considered more important to safety and IO is liable for others
- division between tasks in the safety class in question so that the regulatory body is responsible for some tasks and the IO for others
- the need to carry out a large number of inspections that regulatory body cannot perform by itself for some activities which can be supervised either by the regulator or the IO.

The graphs for pressure equipment are presented in Figure 3, for steel structures in Figure 4 and for concrete structures in Figure 5.

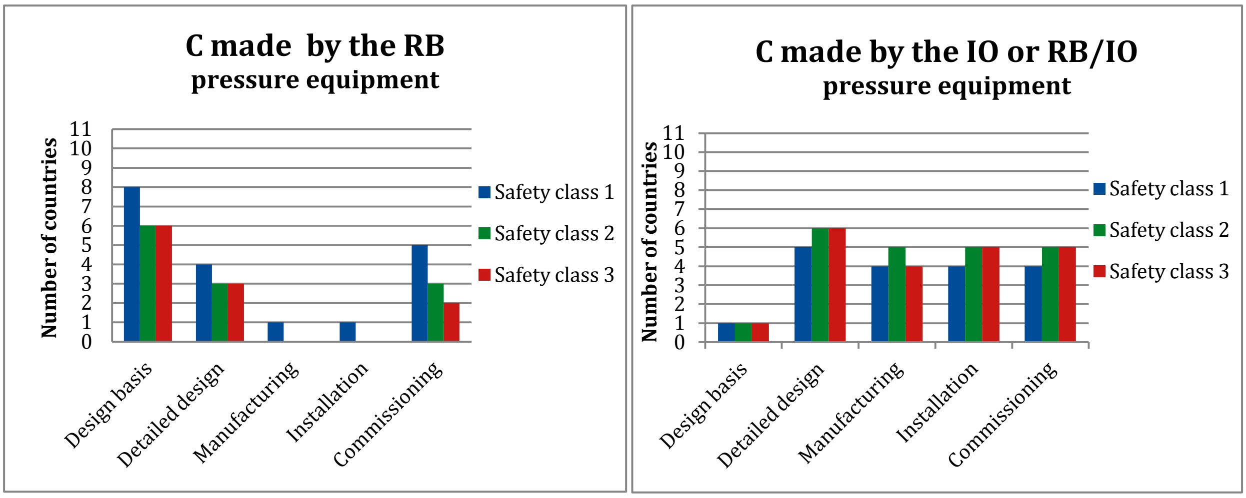

Figure 3. Regulatory inspections of pressure equipment made by regulatory body (RB) and IO in different phases of manufacturing. C = Comprehensive technical control.

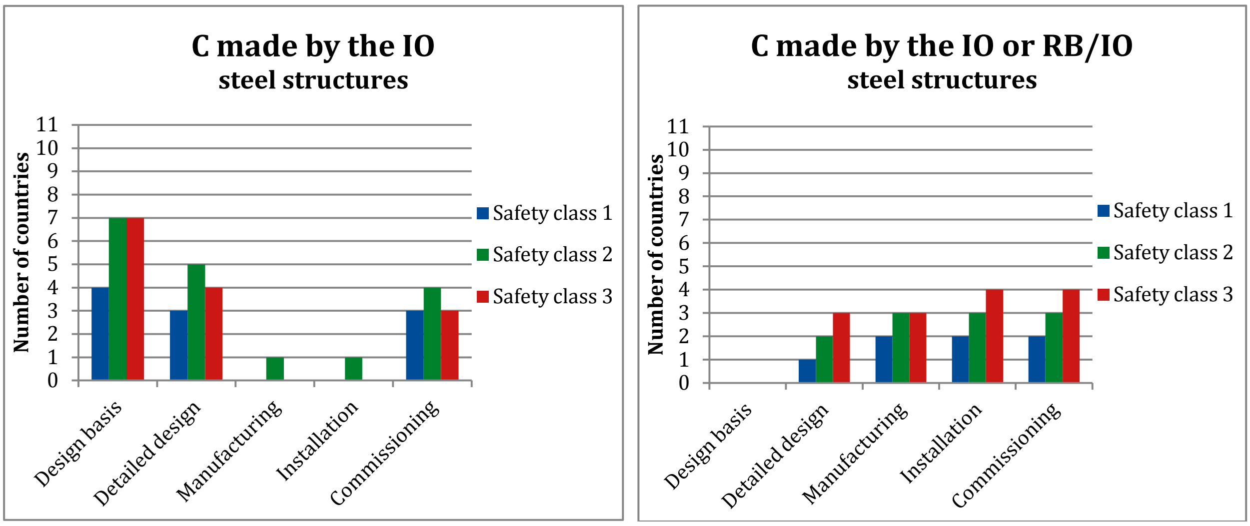

Figure 4. Regulatory inspections of steel structures made by regulatory body (RB) and IO in different phases of manufacturing. C = Comprehensive technical control. *SC1: six (6) countries, SC2: eleven (11) countries, SC3: eleven (11) countries.*

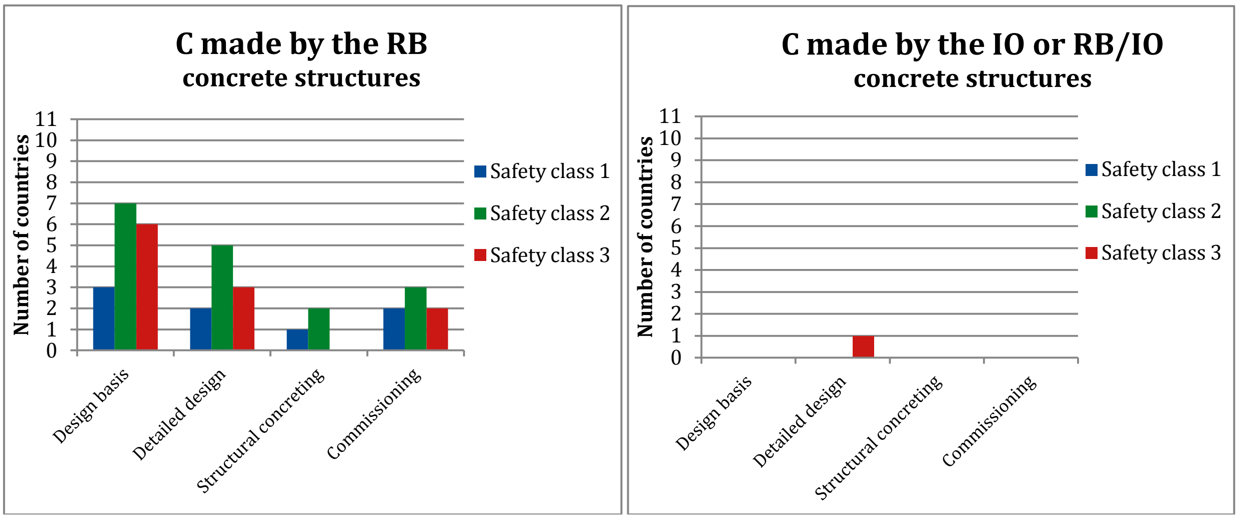

Figure 5. Regulatory inspections of concrete structures made by regulatory body (RB) and IO in different phases of manufacturing. C = Comprehensive technical control. *SC1: five (5) countries, SC2: ten (10) countries, SC3: nine (9) countries.*

Short summaries of countries' regulatory inspection practices concerning structures and components including hold point strategy are presented in the next paragraphs.

Belgium: Regulatory body in Belgium consists of FANC (Federal Agency for Nuclear Control) and Bel-V, which is the subsidiary of FANC and provides technical support. Most of the regulatory inspections of mechanical components are performed together by Bel-V and Mandated Organization (MO), which might be defined as "a regulatory IO". Only one Belgian MO is selected at the moment for this task. For pressurized steam components, the Belgian legislation requires that a MO performs the function of AIA (Authorized Inspection Agency) or the similar IO required by the ASME code. In general Bel-V assesses nuclear safety while MO has the mechanical expertise of components.

Basis for inspections of mechanical components is the ASME code. Inspections for concrete structures are not defined. FANC reviews design phase using technical control by sampling in SC1, SC2 and SC3 and supervises pressure tests of pressure equipment in SC1 and partly in SC2/SC3. Bel-V + MO perform regulatory inspections in SC1,SC2 and SC3 in all phases using comprehensive technical control. In SC2 and SC3 also IOs contracted by licensee are used for components/cases, which are out of the main scope of the MO. Hold points for design, manufacturing and commissioning (Bel-V + MO scope) of pressure equipment are predefined. For installation of pressure equipment and for all phases of steel structures hold points are optional (for concrete structures not defined).

Bulgaria: All regulatory inspections and reviews are on regulatory body's (BNRA, Bulgarian Nuclear Regulatory Agency) responsibility. Review of design basis, construction, installation and commissioning documentation of the structures and components belonging to SC1, SC2 and SC3 are on BNRA's responsibility and BNRA also inspects NPP structures and components to assess whether the requirements in BNRA safety regulations are met. For design basis and commissioning phases comprehensive technical control is used and hold points are predefined. Manufacturing and installation phases/structural concreting are supervised using technical control by sampling and hold points are optional. Commissioning of nuclear power plant/unit can start after issuance of commissioning permit by BNRA.

Czech Republic: All regulatory inspections and reviews are on regulatory body's (SÚJB, The State Office for Nuclear Safety) responsibility. Licensees use IOs widely for all phases of pressure equipment and steel structure inspections. Review of design basis and detailed design of all structures and components in all safety classes is conducted using comprehensive technical control. Other phases are supervised using technical control by sampling. Hold points are predefined.

Finland: Review of design basis and commissioning inspections are on regulatory body's (STUK, Radiation and Nuclear Safety Authority) responsibility in SC1, SC2 and SC3. In SC1 also other inspections and reviews are solely on STUK's responsibility. IOs are used for regulatory inspections for components' supervision/inspections mainly in SC3 and in SC2 depending on equipment's safety significance. SC2 steel and concrete structures and SC3 concrete structures are on STUK's responsibility. Regulatory inspections of SC3 steel structures are on IO's responsibility. Comprehensive technical control is used and hold points are predefined.

France: IOs are used for regulatory reviews and inspections of all nuclear pressure equipment. For N1 (SC1) components, IOs are mandated by regulatory body (ASN) to perform an identified part of conformity assessment. They shall report either monthly and/or punctually for specific reasons and send a final report to ASN about inspections performed. ASN stamps N1 (SC1) pressure equipment. For N2 (SC2) and N3 (SC3) pressure equipment IOs are also used for regulatory inspections and are submitted to ASN's supervision. Comprehensive technical control is used and hold points are predefined for pressure equipment inspections. ASN is responsible for safety relevant steel and concrete structures (all phases) and supervision of them is done by sampling using optional or no hold points.

Lithuania: All regulatory inspections and reviews are on regulatory body's (VATESI) responsibility. Design phase e.g review and approval of technical specification and PSAR is supervised comprehensively, but other phases are supervised using technical control by sampling. Activities related to technical control by sampling are realized by implementing regulatory inspection plans considering safety classification, forthcoming supervision and inspection works of the licensee or results of that work as well as best practices. VATESI can contract Technical Support Organizations for regulatory review and assessment activities when considers it necessary. Hold points for design phase are predefined, otherwise they are optional.

Slovak Republic: All regulatory inspections are on regulatory body's (NRA SR-ÚJD SR) responsibility. NRA SR uses IOs during design phase for design basis assessment. Licensees use IOs widely for all phases of pressure equipment, steel structure and concrete structure inspections. Regulatory body supervises design and commissioning phases using comprehensive technical control, but otherwise supervision is conducted by sampling. Hold points for design phase and commissioning are predefined. Also during manufacturing, installation and structural concreting hold points for technical control are predefined although supervision is conducted by sampling.

Spain: All regulatory inspections and reviews are on regulatory body's (CSN) responsibility. No IOs are used for nuclear regulatory inspections and assessments except inspections according to PED that are performed by IOs under the supervision of Ministry of Industry. Comprehensive technical control is used in all safety classes for review of design (design basis and detailed design) and commissioning inspections of pressure equipment and steel structures if the structure or equipment modification entails a modification of the NPP licence. In all other cases the regulatory activities are supervised using technical control by sampling that can imply reactive and focused inspections or further regulatory activities if the results of sampling are not satisfactory. Regulatory hold points depend on a type of permit granted.

Sweden: Review of design basis is on regulatory body's (SSM) responsibility in SC1, SC2 and SC3 (and SC4/NNS). Reviews of the detailed component design documentation are delegated to IOs. SSM inspects commissioning tests of components/steel structures after major plant modifications in SC1 and SC2, IO in SC3. In minor component replacements or modifications IO supervises manufacturing, installation and commissioning of pressure equipment, components and steel structures in SC1, SC2 and SC3. SSM makes regulatory inspections on sample basis for concrete structures in SC2 and SC3 (SC1 not applicable for concrete structures). When regulatory inspections are carried out by IOs, comprehensive technical control is used. SSM uses technical control by sampling for inspections focusing on major plant modifications. Technical control by sampling is used for concrete structures. Hold points are mostly predefined; only structural concreting phase is optional.

Switzerland: Regulatory body (ENSI) reviews design basis of structures and components in all safety classes. ENSI also supervises commissioning phase with IO (see tables) in all safety classes. Regulatory inspections during manufacturing, installation and final testing (e.g. pressure tests) of SC1 and SC2 pressure equipment are conducted by IO although ENSI takes part in the inspections. IO supervises pressure tests and repairs during manufacturing of SC1 to SC3 (4) pressure equipment as well as manufacturing and installation of SC1 to SC3 (4) steel structures. All phases of concrete structures (BK I, BK II, unclassified buildings) are solely on ENSI's responsibility, but ENSI usually contracts engineering companies to conduct inspections. Hold points are predefined and mostly comprehensive technical control is used. For SC3 (and SC4) pressure equipment and steel structures manufacturing is supervised using technical control by sampling.

UK: All regulatory inspections and assessments are on the responsibility of regulatory body (ONR). Technical support contractors are used to support assessment activity and, less typically, inspection activity, but the responsibility lies with ONR. No IOs are used for direct regulatory inspection or assessment. Regulatory inspections and assessments are undertaken on a sampling basis, with high levels of sampling for safety class 1 SSCs and proportionally less or limited samples for lower safety class SSCs. A limited number of hold points require regulatory permission, many others are defined and controlled by the licensee, and the regulator can specify any of these additional hold points to be subject to regulatory control if appropriate.

3.6 Authorization of IOs

A summary of the practices concerning authorization and contracting of IOs in participating countries for pressure equipment is presented in the Figure 7.

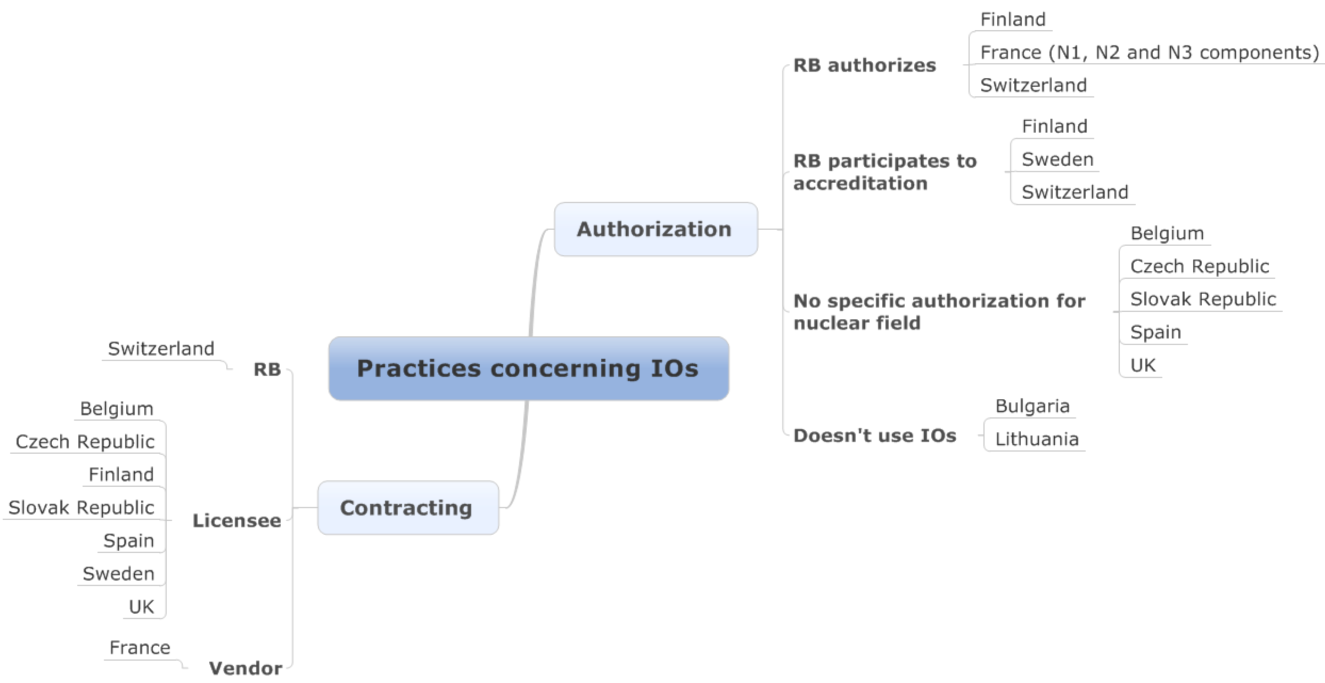

Figure 7. Practices concerning authorization and contracting of IOs in participating countries for pressure equipment.

Short summaries of countries' authorization/contracting practices are presented in the next paragraphs.

Belgium: FANC or Bel-V doesn't authorize IOs. Inspection of nuclear pressurized steam components can only be performed by a Belgian Mandated Organization (MO). MOs are accredited by Belgian Ministry of Labor. Licensee contracts MO and also other IOs, which are used for inspections that are out of the scope of MO. Licensee can also under certain conditions use its own inspection department for inspections out of the regulatory scope of the MO. Licensee shall define prerequisites to contract IO in the quality assurance program. Bel-V supervises implementation of this program.

Bulgaria: BNRA doesn't authorize or use IOs for regulatory inspections. For the purposes of licensing and safety assessment, BNRA may use IO on a specific task. Licensees don't use IOs either, but IOs might work in some cases on behalf of the contractor.

Czech Republic: SÜJB doesn't authorize or use IOs for regulatory inspections. Licensees use IOs and make the contracts with them.

Finland: STUK authorizes IOs, which shall be accredited by FINAS. STUK participates to the accreditation process as an expert. Accreditation is based on the standard EN ISO/IEC 17020, Type A. Licensee contracts IO, which shall be authorized by STUK.

France: ASN authorizes IOs for all nuclear pressure equipment. Before being authorized they shall be accredited for N1, N2 and N3 components. Authorization is based on the ASN guideline which conforms to the standard EN ISO/IEC 17020. For N1 components, manufacturers choose and contract IOs working with inspection programs approved by ASN. For N2 and N3 components IOs work independently (manufacturer contracts), but ASN may supervise their actions.

Lithuania: VATESI doesn't authorize or use IOs or third party organizations for regulatory inspections. Current practice is that licensee by its own decision uses UI for supervision of manufacturing and structural concreting phases, but there are no legal or regulatory requirements about the use of UI. If construction of a new NPP starts, VATESI considers possibility to use third party organizations for inspecting and auditing structures and components of lower safety classes.

Slovak Republic: NRA SR-ÚJD SR doesn't authorize IOs and doesn't perform oversight of IOs. NRA SR-ÚJD SR might use third party organization for independent assessment of complicated safety documentation (e.g. competent technical support organizations). Licensee contracts IO and is responsible for that IO is accredited by SNAS (Slovak National Accreditation Service). Accreditation is based on the standard STN EN ISO/IEC 17020. The accreditation decision of the IO has to be available to NRA SR-ÚJD SR on request.

Spain: CSN doesn't authorize IOs. Licensee is responsible for contracting IOs (if used) and contracting shall be conducted according to licensees's Quality Assurance Manual. IOs involved with PED's requirements are accredited by ENAC (according to standard EN ISO/IEC 17020). Ministry of Industry or equivalent autonomous government organ authorizes these IOs.

Sweden: IOs shall be accredited by SWEDAC. SSM participates to the accreditation process and the accreditation decision is made in consultation with SSM. Accreditation is based on the standard EN ISO/IEC 17020, Type A. Licensee contracts IOs. In some major plant modification projects the licensee requires that the main vendor contracts IOs on behalf of the licensee.

Switzerland: ENSI authorizes IOs, which can also be an engineering company who act as an IO. For supervising/inspecting nuclear pressure equipment IO shall have accreditation in accordance with the standard EN ISO/IEC 17020, Type A. ENSI participates to the accreditation process. Contracts and case by case decisions define the details of organizing the reviews, inspections and reporting to ENSI.

UK: The licensee is responsible for contracting IOs if used. Normally, the IO activity is performed by the licensees' internal regulator providing internal inspection, assessment and oversight. For specific structures (Containment) or components (NSSS), a Third Party Inspection Organization would be contracted by the licensee, in addition to the internal regulatory function.

4. Good practices for inspection of components and structures

These good practices for reviews, assessments and inspections during design, manufacture, construction, installation and commissioning of components and structures are a result of discussions where advantages and disadvantages of different approaches have been compared.

The summaries provided for the role and functions of the licensees and the regulators are a brief set of principles as it is judged they are widely understood. The use of conformity assessment bodies is an evolving topic and these sections on good practices provide more description to show how they can be applied in a variety of regulatory regimes.

4.1 Licensees' control, supervision and oversight

As stated in the European Nuclear Safety Directive the prime responsibility for nuclear safety of a nuclear power plant rests with the license holder and this responsibility cannot be delegated. The licensee should consequently have clearly defined design, manufacturing, installation and commissioning acceptance processes which ensure that necessary reviews, inspections and examinations are performed in the different phases. The licensee should also maintain an internal oversight function which provides a comprehensive examination of activities both within the licensee's organization and external organizations. This includes examination that necessary supervision and control of vendors, contractors and suppliers is performed during different phases of a new build nuclear power plant and during modification of an existing plant. The supervision and control should include to:

- ensure that the contractor[4] has sufficient manpower and competence to carry out the assignment in a safe manner,
- ensure that the contractor has the necessary equipment for executing the assignment and that the contractor employs adequate methods and processes where applicable,
- ensure that the contractor employs management and quality systems that provide full control over safety in conjunction with the assignment and that manufactured and assembled structures, systems, components and devices meet stipulated safety requirements,
- continuously supervise the contractor's activities to ensure that all regulatory requirements[5] and licence conditions are satisfied, along with the goals and guidelines for the activity to which the assignment pertains,
- continuously examine the contractor's continuous improvement programme to evaluate and report events to the licensee and ensure that appropriate safety related measures are taken,
- when necessary, instruct the contractor to take suitable corrective measures, or take such measures himself if the contractor does not adhere to the goals and guidelines established for the assignment
- ensure that all safety and quality requirements are fulfilled in each phase, particularly before a system or component is taken into operation.

The licensee should also, in order to fulfil his responsibility, ensure that personnel from the licensee's own organization as well as from the regulatory body and independent IOs (if used) have access to those facilities where safety related components and structures are manufactured, tested and installed, and to the associated documentation.

[4] The term contractor is here used synonymously with vendor and supplier

[5] In France for pressure equipment regulatory requirements are not under the licensee surveillance: licensee only performs supervision on activities which are identified important for safety.

4.2 Regulatory approaches

Depending on the basic regulatory approach or combination of approaches that are applied the scope and focus of reviews, assessment and inspection during different phases may vary. However, some important aspects should always be subject to reviews and inspections by the regulator. These important aspects should as a minimum include:

- reviews of design basis and extended design conditions of the plant and the related design basis of the structures and components
- reviews and inspections of the licensee's organization, resources and management for internal oversight and arrangements for control and oversight of vendors, contractors and suppliers
- confirmation that independent conformity assessment bodies, including IOs , are accredited or approved for their tasks
- inspections of the licensees quality assurance audits of the supply chain
- reviews of the licensees overall process for successive testing and examination of components and structures including related hold points, which have been defined or approved by the regulatory body
- inspections of licensees arrangements for control of non-conformance, design modifications and design change requests
- reviews of functional system testing programs and other commissioning programs.

If conformity assessment bodies are used for comprehensive detailed reviews and assessments the regulator's work in other phases may be limited to inspections and reviews on a sampling basis.

4.3 Use of conformity assessment bodies

The use of different types of conformity assessment bodies, including IOs, can be prescribed by the regulator in regulations or in response to individual safety cases. The regulator can also express expectations that certain tasks should be tested, reviewed, assessed or inspected by an independent conformity assessment body. Conformity assessment bodies can be contracted either by the regulator or by the licensee or by a manufacturer. Conformity assessment bodies communicate with and report directly to the licensees, the manufacturer or the regulator depending on who has ordered the tasks. Results from their work should however always be available to the regulator.

Depending on national regulatory approach, a good practice in many situations is to apply the following stepwise and sequential approach where independent conformity assessment bodies review, inspect, test and issue certificates or approvals as a basis for further work. Similar sequential approaches may be developed and managed by the licensee. The specific type of document needed to get clearance to move from one phase to another may vary depending on the safety case. This example is adapted to mechanical components and steel structures, but can with some modification also be applied to concrete structures.

<u>Design</u>

A conformity assessment body performs comprehensive reviews of the detailed design documentation based on the regulators review and assessment of the design basis. This detailed design documentation typically includes standards and criteria adopted, structural and other analysis, structural and isometric drawings, material specifications, welding and fabrication/manufacturing processes and their qualification, control/examination plans and procedures for destructive and non-destructive testing. If the conformity assessment body's review, which may include their own verification analysis, show that relevant requirements are met, the body may issue a *design examination certificate or equivalent*.

Manufacturing

The design examination certificate or equivalent should in principle be a prerequisite to start of manufacturing. During manufacturing independent bodies perform supervision and testing in different phases according to the control and examination plans including testing at works, for example visual examination and hydrostatic test. Qualification of welding procedures and welding personnel are supervised and evaluated by a certification body. After testing is completed the results are evaluated with outliers assessed by the licensee and the conformity assessment bodies. Deviations and non-conformances are also reviewed and the results evaluated. If these reviews and evaluations show that relevant requirements are met, the conformity assessment body may issue a *manufacturing examination certificate or equivalent*.

Installation

The manufacturing examination certificate or equivalent should in principle be a prerequisite to the start of installation. During installation independent bodies perform supervisions, examinations and testing according to the control and examination plans. After the installation of a component or structure in the plant an inspection body should verify that

- the component has been installed in accordance with controlled drawings and flowcharts and that performance meets safety requirements,
- deviations and non-conformances identified during installation are reported and evaluated by the licensee and the inspection body as appropriate,
- surface finishes and coatings of installation are finished to the required final state,
- tests have been done to show that the safety valves and other safety equipment operate properly and that the component was not exposed to harmful vibrations or other loads, for which no account is taken when designing the control. The inspection body should witness the tests.

If these verifications and tests show that relevant requirements are met, the body may issue an *installation examination certificate or equivalent.*

Commissioning

The design, manufacture and installation certificates or their equivalents are the basis for the inspection bodies' final assessment of conformity with the regulations or other requirements in the specific safety case. These types of conformity assessments include confirmation that all necessary measures have been taken and that the component or structure has been manufactured and installed according to the design documentation and meets all applicable requirements. Included in the assessments is also confirmation that deviations of various kinds have been handled and remedied correctly and that the necessary maintenance and in-service testing can be undertaken.

If these controls show that relevant requirements are met the body may issue a *certificate of conformity or equivalent*. A certificate of conformity should be a condition for taking a system, component or a structure into overall system functional testing to confirm the limits and conditions of operation identified by the design. The system functional testing should be controlled by the licensee and reviewed by the regulatory body.

Satisfactory system level tests are a prerequisite for taking the system into operation which is beyond the scope of this report.

4.4 Accreditation, authorisation and surveillance of independent inspection, testing and certification organizations

4.4.1 Accreditation

The use of independent inspection, testing and certification organizations in support of the regulators and licensees work with verifying compliance with national nuclear safety requirements and license conditions assumes that there is confidence in their inspection activities. This confidence can be supported by an accreditation of such organizations delivered by a national accreditation body.

If bodies that carry out inspection, testing and certification activities are accredited, this should be done for their tasks in question in the nuclear field based on applicable regulations and standards. This means that the bodies should be accredited in accordance with recognized accreditation standards and for those specific inspection, testing and certification tasks that result from applicable national nuclear safety regulations.

The international standards define general requirements for conformity assessment bodies in the accreditation process and they should be followed in the accreditation. These standards are listed in the appendix 5 of this report.

Accreditation decisions should be based on inspections (audits) and comprehensive reviews of the organizations' management systems (including working procedures and professional training programs), technical competence, personnel resources and work practice for verification that the requirements in relevant accreditation standards and relevant national nuclear safety regulations are met. A national accreditation body carries out its inspections and reviews in accordance with standards and specifications agreed with the national nuclear regulatory body, both before an accreditation decision and during subsequent surveillance. A good co-operation and information flow between the national nuclear regulatory body and the national accreditation body are consequently essential. However, it is the national accreditation body that makes the independent accreditation decision.

An accreditation certificate should clearly specify the scope of accreditation, and thus the field in which the body may act in its role as an accredited body. This means for example that an accreditation certificate should include information on those regulations and other rules against which conformity assessment can be made. An accreditation certificate should also:

- include conditions related to reporting and co-operating with the national accreditation body and the national nuclear regulatory body
- define the mandates with regard to results of reviews and inspections.

Depending on the regulatory regime an accreditation certificate may include additional information if an authorization by the national regulatory body is also needed in order to act as conformity assessment body (see section 4.4.2).

4.4.2 Authorisation

Some regulatory regimes also require authorisation for these organizations in addition to accreditation. This is generally applied when the regulator has limited engagement in the accreditation process, but can also be a part of the applied regulatory approach. When existing, the authorisation process may include additional reviews and inspections by the national regulatory body of such organization, its management system with working procedures, adequacy of their resources and technical competence. The process results in an authorisation delivered by the regulatory body in order to allow these organizations to perform defined tasks during specified time periods.

4.4.3 Impartiality, independence and competence

Independent conformity assessment bodies, including inspection organizations, that perform different kinds of reviews, assessments, testing and inspections during design, manufacturing and installation of component and structures in support of the regulators work should meet the requirements of impartiality and independence of a type A body according to the standard EN ISO / IEC 17020. Type B bodies can however have an important role for example in the licensee's inspections. The degree of impartiality and independence should be defined by the national regulatory body depending on the situation and safety case.

Organizations that audit and certify management systems for different kinds of manufacturing processes including welding should meet the requirements of impartiality and independence in accordance with EN ISO / IEC 17 021. Organizations that certify welding personnel should meet the requirements of impartiality and independence in accordance with EN ISO / IEC 17 024. Organizations that certify components, processes and services according to EN 45011 should meet the same requirements of impartiality and independence as a type A body.

Bodies performing destructive testing and non-destructive testing (laboratories) of safety significant components should normally also meet the same requirements of impartiality and independence as type A bodies according to the standard EN ISO / IEC 17025. For other types of components and structures less stringent requirements of impartiality and independence can be accepted.

Inspection, certification and testing bodies should have systems that specify the knowledge and skills needed for their personnel to carry out the inspection, certification and testing tasks in question. Competency should be determined by examination at the individual level and result in a personal certificate that specifies the tasks to be performed. The validity of certificates should be limited for a certain time period.

Examples of personnel knowledge and skills needed for a conformity assessment body active in the area of nuclear mechanical components is given in Appendix 2.

4.4.4. Non- conformances and reporting

An accredited conformity assessment body should be able to handle and decide on acceptance of non-conformances within those limits that are given in their working procedures, and which have been approved in the accreditation process. When non-conformances outside the limits are observed a certificate of conformity with national regulations cannot be issued, and the components should consequently not be taken into operation. In such situations a licensee or a manufacturer has to take measures to correct the non-conformance or justify and apply to the national regulator for an exception from the relevant requirement. In some regulatory regimes the inspection organization should also report non-conformance to the regulatory body.

It is also important that the terms of accreditation or authorization include conditions requiring accredited organizations to report general problems and serious deviations direct to the national regulatory body.

4.4.5. Collaboration

Collaboration between the relevant stakeholders is important in regulatory systems in which accredited bodies carry out important inspection activities.

The national accreditation body and the national regulatory body should collaborate, both in relation to accreditation and subsequent follow-up and surveillance activities. The national accreditation body should immediately inform the national nuclear regulatory body if a surveillance reveals such deficiencies in an accredited body's activities that accreditation may be withdrawn.

Accredited organizations should be involved in such experience feedback meetings that the national accreditation body and nuclear regulatory body organize. This should also be stated in their terms of accreditation.

4.4.6 Mutual recognition of accreditations

Generally, accreditation is performed according to harmonized standards for specific tasks according to product or facility requirements. In the nuclear safety field requirements concerning systems, structures and components may vary between different countries. Mutual recognition of accreditations is therefore not possible as a general rule. However, for specific areas where there are similar national safety requirements, mutual recognition of accreditations can be made by agreements between accreditation bodies after consultation and agreement of the regulatory bodies.

5. Summary

WENRA decided to initiate work concerning benchmarking of European inspection practices for components and structures of nuclear facilities at its March 2010 meeting**.** A working group was established for this purpose. The working group was expected to describe the practices being applied in WENRA countries for arranging inspection activities of mechanical equipment, steel structures and concrete structures. The countries were to:

- explain possible formal approvals of the involved third party organizations
- explain possible correlation with safety classes
- explain if two redundant inspections have to be done where responsibility is with the licensee and with the regulatory body.

As a second step of the work, the group was expected to:

- discuss the experience gained in WENRA countries with their respective approach
- discuss possible future development plans and needs to modify the approach
- consider good practices that could become harmonized European practices.

The detailed objectives of the group were defined as follows:

- to learn from others practices to develop your own practices
- to discuss the added value of different basic regulatory approaches
- to assure similar degree of involvement by the industry
- to make use of foreign IOs easier in the long term (interchangeability, accreditation)

In the early phase of the work national reports were provided by the participating countries. These were used to make initial comparisons between the countries and to focus the work of the group on the most essential issues. An overall scheme of the inspection practices was developed by the group for the structures and components which were studied. The group developed in its first meeting tables to collect information on inspection practices in a systematic way. The tables were created for pressure equipment, steel structures and concrete structures. Regulatory and licensee inspections and auditing were completed in separate tables. In the final phases of the work countries provided short national summaries of their inspection practices. All this information provided by the participating countries was crucial in developing this unique benchmarking report.

Every participating country confirms that the primary responsibility for the safety of NPPs and quality of NPP structures and components rests with the license holder. The licensee reviews and approves documents and inspections related to component and structure design, manufacturing, installation and commissioning before presenting them to the regulator for approval. Licensees are expected to take all the necessary steps and actions to fulfil applicable safety requirements and to organize the necessary quality control. Licensees shall verify that all the organizations related to these steps have arrangements to produce appropriate quality.

A general conclusion is drawn that all countries see it necessary that the most important SSCs have inspections undertaken by the regulatory body. As the nuclear safety significance and the safety classification of a SSC reduces, the role of the regulatory body tends to reduce and that of conformity assessment bodies, and in particular inspection bodies tends to increase. Similarly the independence of the conformity assessment bodies is at its highest level for the most important SSCs and tends to reduce as the nuclear safety significance and the safety classification of the SSC tends to reduce.

The practices across countries are fairly uniform for pressure equipment although, as concerns regulatory practices, also on this area there is variation between countries concerning especially management system audits and/or focused audits. As concerns the design and commissioning phases comprehensive technical control is performed by almost all regulators. In some cases the regulatory body reviews only the design basis of components and structures and independent IOs are used to review the detailed design. For the manufacturing and installation phases there seem to be more variations between countries. Auditing of the QM systems of the manufacturers of the most important pressure equipment by the regulatory body or an IO is generally considered important.

There are larger variations in inspection and auditing practices between regulators as concerns steel and concrete structures. Also for them a comprehensive technical review of the design or the design basis is done by most regulators. As concerns manufacturing and structural concreting typically technical control by sampling is used whereas for installation and commissioning either comprehensive technical control or technical control by sampling is used.

Hold points between the different lifecycle phases of structures and components are used widely in most countries. Especially for design and commissioning phases all participating countries use regulatory hold points.

There is a tradition in many countries to use independent conformity assessment bodies to review, assess and supervise different activities during design, manufacture, construction and commissioning of components and structures, especially the pressurized components. The use of such conformity assessment bodies can be prescribed by the regulator and contracted for its task by either the regulator or by the licensee or the vendor. In some countries the regulator does not prescribe the use of independent conformity assessment bodies but expresses expectations that the licensee contracts conformity assessment bodies for review, assessment or supervision of important aspects during design, manufacture, construction and commission. Regulatory bodies can thus choose to have an emphasis in either of the two basic approaches, prescriptive and goal-setting/outcome based, for inspection of components and structures or to combine them in an appropriate manner and to differing degrees use independent conformity assessment bodies as part of the work. The regulatory approach may have influence on the use of independent IOs as well on their regulatory oversight.

One of the basic tasks of the group was to “consider good practices that could become harmonized European practices”. For this purpose a subgroup was established which also studied the harmonized European practices applied on the conventional side. Based on the work of this subgroup good practices for inspection of components and structures are presented in chapter 4 of this report. These good practices cover the following issues:

-licensees control, supervision and oversight

-regulatory approaches

-use of conformity assessment bodies in different phases

-accreditation, authorization and surveillance of independent inspection, testing and certification organizations

Especially concerning the use of independent inspection, testing and certification organizations good practices are presented for the following issues:

-accreditation

-authorization

-impartiality, independence and competence

-non-conformances and reporting

-collaboration

-mutual recognition of accreditations.

The good practices are generic in nature. They were developed especially for safety-related pressurized equipment but can be applied to all types of components and structures.

These good practices should be adopted by all WENRA countries when they are developing their inspection practices either by introducing them in the national regulations or by applying them into individual safety cases.

Definitions

For this benchmarking it was defined that

- ✓ Concrete structures meant in this report include buildings and other concrete, reinforced concrete and post-tensioned concrete structures.
- ✓ Steel structures include, for instance, the liner of the containment, liners and structures of spent fuel and reactor internal pools, hoisting equipment, pipe whip restraints, and fire doors.
- ✓ Mechanical components include, pressure vessels, piping, pump units, valve units, reactor internals, diesel generators, etc.
- ✓ Conformity assessment is a demonstration that **specified requirements** relating to a **product, process, system, person or body** are fulfilled.
- ✓ Conformity assessment body is a body that performs conformity assessment services. In this report following conformity assessment bodies are mentioned:
  - o Testing organization means organizations which conduct non-destructive or destructive testing.
  - o Inspection body or inspection organization means organizations that conduct activities (other than testing organizations) intended to verify safe design and achievement of specified quality of structures and components and to audit quality management systems.
  - o Certification body or certification organization is a body operating a product certification system. The word "product" includes also processes and services.
  - o In this report IO is used for inspection body or certification body. IO can be accredited also for both activities.
  - o Notified body is a conformity assessment body fulfilling requirements in Decision No 768/2008/EC of the European Parliament and of the Council of 9 July 2008 on a common framework for the marketing of products, and repealing Council Decision 93/465/EEC.
- ✓ Constructor inspection organization means a type B (second-party) organization of the constructor. Constructor might also contract a third-party organization for specific conformity assessment tasks.
- ✓ Utilities inspection organization means type B (second-party) organization that conducts inspections and assessments to certain group of equipment and structures (e.g. one licensee)
- ✓ Mandated organization is an organization mandated with the inspection of pressure retaining equipment (including NPP pressure equipment) and steel structures defined in ASME XI in Belgium. It has to be accredited by the Belgian Ministry of Labor.
- ✓ Third-party organizations are understood as organizations that make conformity assessment activities which are independent of the person or organization that provide the object and also independent of the end user of the object. They have proven professional competence and properly verified qualification system to conduct independent inspections. (Conformity assessment body may fulfill these requirements or it may fulfill second-party assessment body’s requirements.)
- ✓ Accreditation is a third-party attestation related to conformity assessment body conveying formal demonstration of its competence to carry out specific conformity assessment tasks.

Examples of knowledge and skills needed for a conformity assessment body active in the field of nuclear mechanical components and structures

A conformity assessment body working with mechanical components in nuclear power plants should have competence in

- Structural integrity including aspects of design, manufacturing and installation
- Safety systems and process engineering
- Safety and quality classification systems used in nuclear power plants
- National nuclear safety regulations for components

Within the organization there should therefore be sufficient personnel who have demonstrable knowledge and competence in the following areas. For each main area (design/materials, manufacturing and inspection/testing), there should also be at least one person who has expertise competence in the area.

Design and materials

- Mechanical properties of materials
- Weldability and heat treatment of materials
- Environmental impact on material and degradation mechanisms
- General design rules for machine elements, plates, shells, pressure vessels, piping, valves, pumps, supports
- Methods for deriving load input data from design basis
- Design rules according to European harmonized standards for pressurized components
- Design rules according to recognised international standards such as ASME, RCCM and KTA for nuclear mechanical components
- Drawing Rules
- Strength analysis according to European harmonized standards for pressurized components
- Strength analysis according to recognised international standards such as ASME, RCCM and KTA for nuclear mechanical components
- FEM analysis

Manufacturing technology

- Methods for forming, moulding, bending, forging, surface preparations - possibilities and limitations
- Welding methods - possibilities and limitations for different material and material combinations
- Welding qualification procedures for welding procedures and personnel[6]

[6] Certificate to assess welding personnel and welding procedures should be issued on the basis of education, training and proficiency testing, and experience that apply to the International Welding Engineer (IWE) or *equivalent qualification requirements.*

Inspection and testing

- Manufacturing and installation inspection and control planning according to European harmonized standards for pressurized components
- Manufacturing and installation inspection and control planning according to recognised international standards such as ASME, RCCM and KTA for nuclear mechanical components
- Visual and dimensional inspection techniques
- Pressure and leak testing techniques
- Non-destructive testing methods - possibilities and limitations
- Functional system testing methods and commissioning

Conformity assessments

- Review and assessment of non-conformance
- Issuing of certificates

Summary tables
Regulatory inspections and auditing

Definitions to abbreviations used in the tables:

Who does?

RB	Regulatory body
IO	Inspection organization
CI	Contractor inspection organization
MO	Belgian mandated organization (IO)

How it does? What is the completeness of the activity?

C	Comprehensive technical control
T	Technical control by sampling
M	Management system audit
F	Focused audit (e.g. follow up of a specified product)
R	Reactive intervention by exception
D	Control of inspection documentation

Mandatory inspections and auditing are underlined.

Comprehensive technical control (C) refers to a practice where all relevant technical aspects are reviewed or inspected and the control covers a major proportion of structures and components.

If the control is based on **sampling (T)**, a combination of different factors is typically used for sampling. Factors which are used as a basis may include safety significance, novelty, complexity and operating experience of the structure or component. Sample size is typically increased if problems occur.

Design basis of components and structures refers to those technical requirements which have to be set on components and structures in order that they would meet the demands based on plant and system level design. Design basis includes functional requirements and loading conditions for normal and accident conditions. Also safety, seismic and quality classification are part of component or structure level design basis.

Pressure equipment

Design phase (design basis/detailed design)

Safety Class	Belgium	Bulgaria[7]	Czech Republic	Finland	France	Lithuania	Slovak Republic	Spain[8]	Sweden	Switzerland	UK
1 (Who) (How)	RB/MO T/CMF	RB/- C/-	RB/RB CM/CM	RB/RB CMR/CMR	RB/RB, IO CM/CMR	RB/RB C/T	RB,IO/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/IO CM/C	RB/RB+IO C/C	RB/IO TM/CMRD
2 (Who) (How)	RB/MO, IO T/CMF	RB/- C/-	RB/RB CM/CM	RB/RB,IO CMR/CMR, CR	RB/IO TM/CTMR	RB/RB C/T	RB,IO/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/IO TM/C	RB/RB+IO C/C	RB/IO TM/CMRD
3 (Who) (How)	RB/MO, IO T/CMF	RB/- C/-	RB/RB CM/CM	RB/IO CR/CR	RB/IO TM/CTMR	RB/RB C/T	RB,IO/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/IO TM/C	RB/RB+IO C/C	RB/IO TM/CMRD

Manufacturing

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak[9] Republic	Spain	Sweden	Switzerland	UK
1 (Who) (How)	MO CMF	RB T	RB TMFR	RB CMR	RB, IO CMR	RB T	RB,IO TMFRD	RB TM	IO C	RB,IO C	RB,IO TMFR
2 (Who) (How)	MO,IO CMF	RB T	RB TMFR	RB, IO CMR, CR	IO CTMR	RB T	RB,IO TMFRD	RB TM	IO C	RB,IO C	RB,IO TMFR
3 (Who) (How)	MO,IO CMF	RB T	RB TMFR	IO CR	IO CTMR	RB T	RB,IO TMFRD	RB TM	IO C	IO TD	RB,IO TMR

Installation

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak[9] Republic	Spain	Sweden	Switzerland	UK
1 (Who) (How)	MO CMF	RB T	RB TMR	RB CMR	RB, IO CMR	RB T	RB,CI TMFRD	RB TM	IO C	RB,IO C	RB,IO TMR
2 (Who) (How)	MO,IO CMF	RB T	RB TMR	RB, IO CMR, CR	IO CTMR	RB T	RB,CI TMFRD	RB TM	IO C	RB,IO C	RB,IO TMR
3 (Who) (How)	MO,IO CMF	RB T	RB TMR	IO CR	IO CTMR	RB T	RB,CI TMFRD	RB TM	IO C	IO CD	RB,IO TMR

[7] Detailed design is supervised by the licensee.
[8] For design basis CM in all safety classes when associated to a modification licence; otherwise TFR. For detailed design: C in all safety classes when associated to a modification licence; otherwise TFR.
[9] Quality management and quality requirements documentation (D) for manufacturing and installation phase are approved by RB.

Commissioning

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France		Lithuania	Slovak Republic	Spain	Sweden[10]	Switzer-land	UK
1 (Who) (How)	MO and RB[11] CMF	RB CTMFR	RB CM	RB CMR	RB, IO CTMR		RB T	RB,IO,CI CMD	RB CM	RB/IO TM/C	RB,IO CD	RB,IO TMFR
2 (Who) (How)	MO,IO and RB[11] CMF	RB CTMFR	RB TM	RB CMR	IO CTM	RB TMR	RB T	RB,IO,CI CMD	RB CM	RB/IO TM/C	RB,IO CD	RB,IO TMFR
3 (Who) (How)	MO,IO CMF	RB CTMFR	RB TM	RB CMR	IO CTM	RB TMR	RB T	RB,IO,CI CMRD	RB CM	RB/IO TM/C	RB,IO CD	RB,IO TMFR

[10] IO = focus on functional tests in safety classes 1, 2 and 3.
[11] RB= pressure tests and pre-operational tests in safety classes 1, 2 and 3.

Steel structures

Design phase (design basis/detailed design)

Safety Class	Belgium	Bulgaria[12]	Czech Republic	Finland	France	Lithuania	Slovak Republic	Spain[13]	Sweden	Switzerland	UK
1 (Who) (How)	RB/MO,IO T/C	RB/- C/-	-	-	-	RB/RB C/T	-	RB/RB CM,TFR/C,TFR	-	RB/RB CD/C	RB/CI TMFR/ CMRD
2 (Who) (How)	RB/MO,IO T/C	RB/- C/-	RB/RB CM/CM	RB/RB CMR/CMR	RB/RB TMFR/TMFR	RB/RB C/T	RB/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/IO TM/C	RB/RB CD/C	RB/CI TMFR/ CMRD
3 (Who) (How)	RB/MO,IO T/C	RB/- C/-	RB/RB CM/CM	RB/RB[14],IO CMR/CMR, C R	RB/RB TMFR/TMFR	RB/RB C/T	RB/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/IO TM/C	RB/RB CD/C	RB/CI T/CRD

Manufacturing

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak[15] Republic	Spain	Sweden	Switzerland	UK
1 (Who) (How)	MO,IO C	RB T	-	-	-	RB T	-	RB TM	-	IO CD	RB,CI TMFR
2 (Who) (How)	MO,IO C	RB T	RB TM	RB CMR	RB TMFR	RB T	RB,CI TMD	RB TM	IO C	IO CD	RB,IO TMFR
3 (Who) (How)	MO,IO C	RB T	RB TM	RB, IO CMR	RB TMFR	RB T	RB,CI TMD	RB TM	IO C	IO TD	RB,IO T

Installation

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak[15] Republic	Spain	Sweden	Switzerland	UK
1 (Who) (How)	MO,IO C	RB T	-	-	-	RB T	-	RB TM	-	IO CD	RB,CI TMFR
2 (Who) (How)	MO,IO C	RB T	RB TM	RB CMR	RB TMFR	RB T	RB,CI TMD	RB TM	IO C	IO CD	RB,CI TMFR
3 (Who) (How)	MO,IO C	RB T	RB TM	RB, IO CMR	RB TMFR	RB T	RB,CI TMD	RB TM	IO C	IO CD	RB,CI T

12 Detailed design is supervised by the licensee.
13 For design basis CM in all safety classes when associated to a modification licence; otherwise TFR. For detailed design: C in all safety classes when associated to a modification licence; otherwise TFR.
14 STUK reviews the plans related to physical protection of NPP's.
15 Quality management and quality requirements documentation (D) for manufacturing and installation phase are approved by RB.

Commissioning

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak Republic	Spain[16]	Sweden	Switzer-land	UK
1 (Who)	MO,IO	RB	-	-	-	RB	-	RB	-	RB,IO	Not applicable
(How)	C	CTMFR				T		CM,TFR		CD	
2 (Who)	MO,IO	RB	RB	RB	RB	RB	RB	RB	RB/IO	RB,IO	Not applicable
(How)	C	CTMFR	T	CMR	TMFR	T	CTMFRD	CM,TFR	TM/C	CD	
3 (Who)	MO,IO	RB	RB	RB	RB	RB	RB,IO	RB	RB/IO	RB,IO	Not applicable
(How)	C	CTMFR	T	CMR	TMFR	T	CTMFRD	CM,TFR	TM/C	CD	

[16] For design basis CM in all safety classes when associated to a modification licence; otherwise TFR. For detailed design: C in all safety classes when associated to a modification licence; otherwise TFR.

Concrete structures

Design phase (design basis/detailed design)

Safety Class	Belgium	Bulga-ria[17]	Czech Republic	Finland	France	Lithua-nia	Slovak Republic	Spain[18]	Sweden	Switzer-land	UK
1 (Who) (How)	RB/RB Not defined	RB/- C/-	-	-	-	-	-	RB/RB CM,TFR/C,TFR	-	RB/RB CT/C	RB/CI TMFR/ CMRD
2 (Who) (How)	-	RB/- C/-	RB/RB CM/CM	RB/RB CMR/CMR	RB/RB TMFR/TMFR	RB/RB C/T	RB/RB CM/CM	RB/RB CM,TFR/C,TFR	RB/RB[19] TM/TM	RB/RB CT/C	RB/CI TMFR/ CMRD
3 (Who) (How)	-	RB/- C/-	RB/RB CM/CM	RB/RB[20], IO CMR/CMR, CR	RB/RB TMFR/TMFR	RB/RB C/T	RB/RB CM/CM	RB/R CM,TFR/C,TFR	RB/RB[19] TM/TM	-	RB/CI T/CRD

Structural concreting

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak[21] Republic	Spain	Sweden	Switzer-land	UK
1 (Who) (How)	RB Not defined	RB T	-	-	-	-	-	RB TM	-	RB CT	RB,CI TMFR
2 (Who) (How)	-	RB T	RB TM	RB CMR	RB TMFR	RB T	RB,CI TFD	RB TM	RB[19] TM	RB CT	RB,CI TMFR
3 (Who) (How)	-	RB T	RB TM	RB TMR	RB TMFR	RB T	RB,CI TFD	RB TM	RB[19] TM	-	RB,CI T

Commissioning

Safety Class	Belgium	Bulgaria	Czech Republic	Finland	France	Lithuania	Slovak Republic	Spain	Sweden	Switzer-land	UK
1 (Who) (How)	RB Not defined	RB CTMFR	-	-	-	-	-	RB T	-	RB CT	RB,CI TRF
2 (Who) (How)	-	RB CTMFR	RB T	RB CMR	RB TMFR	RB T	RB TFD	RB T	RB TM	R CT	RB,CI TRF
3 (Who) (How)	-	RB CTMFR	RB T	RB CMR	RB TMFR	RB T	RB D	RB T	RB TM	-	Not audited

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17 Detailed design is supervised by the licensee.
18 For design basis CM in all safety classes when associated to a modification licence; otherwise TFR. For detailed design: C in all safety classes when associated to a modification licence; otherwise TFR.
19 Using IOs for reviews of detailed design and other inspection tasks during construction and facility modifications are being prepared. An investigation is underway on how SSMs regulations should be changed.
20 STUK reviews the plans for reactor island, fuel and safety buildings, circulation water structures and physical protection of NPP's (airplane crach).
21 Quality management and quality requirements documentation (D) for structural concreting phase are approved by RB.

Appendix 4

National summary reports

WENRA, Inspection Working Group

BELGIUM

NATIONAL SUMMARY

Foreword

The Belgian inspection practices as discussed here focus on present practices related to plant modifications (most of them are repairs and replacements).

Those modifications are dealt with, conform with the Belgian regulations, according the prescriptions of ASME-code, section XI, as amended by Belgian regulatory documents.

The regulatory oversight of the few, recently build concrete structures has been limited to a general overview of those activities by Bel V. This should not be considered as standard practice. Therefore, no detailed answers for concrete structures are provided hereafter.

A Basic regulatory approach in the country (structures and components)

Organizations competent at regulatory level: Belgian Regulatory Body and Mandated Organization

The FANC (Federal Agency for Nuclear Control) is the competent authority in the field of nuclear applications. Its subsidiary Bel V provides the technical expertise for carrying out inspections in licensed facilities. Together, FANC and Bel V form the Regulatory Body (RB).

Additionally, all NPP pressure retaining component are subject to inspections by a Belgian Mandated Organization (MO) in charge of inspection of steam components according to the Belgian legislation. The MO investigates the mechanical safety by verifying that the requirements of ASME III and XI are met. The MO has its primary expertise in mechanical safety whereas Bel V also assures the global nuclear safety, taking into account radiation protection and probabilistic safety assessments (PSA).

Applicable regulations in Belgium

Generally speaking, Belgium has chosen the American rules for the design and construction of its nuclear power plants, i.e. the requirements of the Code of Federal Regulations (10CFR50), as well as of the ASME code, of the ANS/IEEE standards and of the documents issued by the US-NRC such as the Regulatory Guides, the Standard Review Plans, the NUREGs…Safety classes are defined according to the US rules (R.G. 1.26, R.G. 1.29 & R.G. 1.143)

Historically, the basic Belgian pressure equipment regulation has been the General Rules for protection at Work (RGPT/ARAB), which are still the legal basis. It has evolved with time, e.g. to endorse the Pressure Equipment Directive 97/23/CE. These regulations do not address explicitly the production of steam by a nuclear reactor.

Regarding the pressure vessels which are part of the nuclear installations, a derogation has been established to allow the replacement of the Belgian rules (RGPT/ARAB) by the American ones.
Transpositions in Belgium of the regulatory aspects of the ASME code (sections III & XI) specify the scope of the inspector assignments as defined by the code which are entrusted to the Mandated Organization and to Bel V. The Authorized Inspection Agency assignment may also entrusted to certain independent entities subject to conditions defined in the transposition (ASME, section XI).

The Mandated Organization and those independent entities act thus as third-party Inspection Organizations (IO) having expertise in mechanical components.

Additionally, for major plant modifications implying a possible modification of the license, e.g. the steam generator replacement and associated thermal power up-rates, the licensee has to prepare a Preliminary Safety Analysis Report (PSAR), which must be approved by Bel V before plant modifications may be implemented.

Hold-points and witness points

Hold-points and witness points are as defined by the ASME-code.

B Expectations on licensees

The licensees have the primary responsibility for the safety of NPPs and the quality of their structures and components.

The licensee shall ensure that every organization the activities of which are connected to design, manufacture, installation, testing and inspection of structures and components, have appropriate quality and management systems and qualified personnel for the work they perform. The licensee must review and approve all the documentation, perform inspections and approve tests before presenting those to Bel V or the Mandated Organization. The licensee is responsible that regulatory requirements and guides are followed. The licensee delivers a Certificate of Authorization to Contractors based on a QA audit performed by the MO.

Identified non-conformances are assessed and corrective and preventive actions are be taken. This is governed by the licensee’s management system, which includes procedures to process non-conformances identified in processes and products.

C Regulatory review of design documentation (RB/IO, safety class)

Neither licensing of a design organization nor auditing of the management thereof are foreseen for the Owner or for the Owner’s Agent, i.e. the organizations to which the tasks are entrusted by the Owner to be carried out under the responsibility of the Owner. The manufacturing follow-up agreements passed with the MO cover the Code’s requirements regarding the Owner and the Owner’s Agent.
Design documentation is established, and reviewed and approved by the IO as foreseen by the ASME-code.

Design inputs are evaluated by Bel V as part of its role related to the global nuclear safety.
The design reports are distributed to the RB and IO for inspection for approval or information according to the detailed requirements of the transpositions.

Control of non-conformances

With a view to the practical application of the ASME-code in a non-US context, the requirements of the code may be replaced by requirements that offer at least equivalent guarantees.

The party applying for shall submit a request for derogation in parallel both to the Owner or the Owner’s Agent and to the MO. Should that party be not conform to the advice given by the MO, the latter will enter a reservation on the ”Data Report”. For any aspects that may have an impact on the nuclear safety, Bel V has also to approve the request.

The same approach applies in case of non-conformities (i.e. deviation or deficiency with respect to the adopted code or set of rules).

National summary reports

D Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

All activities at design stage, off-site manufacturing, on-site construction, manufacturing and installations, and commissioning are dealt with according to the ASME-code, sections III and XI, as transposed to the Belgian context. The corresponding regulatory inspections as required by the code are all performed by the AIA, i.e. an IO as described in §§ A & E.

Off-site manufacturing

An approval is required for material, component and structure manufacturers. Auditing of the manufacturers is done by the IO.

Fabrication and inspection methods (welding, hard facing, heat treatment etc.) as well as fabrication and inspection personnel are validated by the IO.

Manufacturing records are reviewed by the IO.

Regulatory inspection of products and witnessing of functional and pressure tests by the manufacturers are done by the IO.

On-site construction, manufacturing and installations & commissioning activities are managed by the Licensee, Owners Agent and by the MO or other IO (construction/inspection inspectors). The nuclear regulator will intervene in case nuclear safety may be affected. Additionally, Bel V monitors all pre-service tests .

The control of non-conformances follows the same rules as at design stage.

E Authorization of IOs

AIB-Vincotte is one of the Belgian Mandated Organization for inspection of pressurized steam components. It is currently selected by the licensee for inspection related to these components in the Belgian NPPs, including assuming the role of AIA as required by the ASME codes. It is accredited by the Belgian Ministry of Labor.

The Owner may also entrust the AIA assignment to an independent entity only in the case of repair or replacement of some class 2 or 3 equipment which are out of the main scope of the MO. The most significant case is the AIA assignment entrusted to the Owner's or the Engineer's inspection department. This possibility is currently not used. The intervention modalities of this inspection department are specified in the Owner's Quality assurance Program. The verification of the proper implementation of this program rests with Bel V.

F Use of management system audits/focused audits

The Owner delivers authorizations after audits or equivalent verifications by the MO or by the IO.
The nuclear regulatory body does not deliver any authorisation.

Appendix 4

National summary reports

WENRA, Inspection Working Group

BULGARIA

NATIONAL SUMMARY

A. Basic regulatory approach in the country (structures and components)

State control on the safe use of nuclear energy and ionizing radiation and safe management of radioactive waste and spent fuel is carried out by the Chairman of the Nuclear Regulatory Agency, named further “Agency”, who is independent specialized body of Government.

The Bulgarian nuclear regulatory body, Bulgarian Nuclear Regulatory Agency (BNRA) issues a number of permits, mentioned as a hold points below and has the responsibility to supervise licensees’ activities by reviewing siting, design, construction, installation and commissioning documentation as well as by inspecting NPP structures and components to assess whether the requirements in BNRA safety regulations are met. Before BNRA reviews any documentation, the licensee must have approved it. The licensee also performs and approves inspections before BNRA inspection. The licensee has the primary responsibility for the safety of NPPs and the quality of their structures and components.

Hold-points

Conceptual design of nuclear power plant and proposed plant site are subject of regulatory review and approval (hold point).

Regulatory review of the design basis of nuclear facilities (nuclear power plant/unit, research reactor, etc.) is conducted as part of the licensing process (design permit and order of approval of technical design) (hold points). Order of approval of technical design is a prerequisite for next step of licensing, namely application for construction permit (hold point). Modifications of systems, structures and components (SSC), important to safety are performed after issuance of permit by BNRA (hold point).

Nuclear power plant construction works start after issuance of construction permit by BNRA (hold point). Construction permit is issued by BNRA only if the submitted documentation by the licensee is in compliance with the requirements of ASUNE and applicable regulations. Extra hold points regarding carrying out construction works and schedule can be put through the conditions of the permit.

Commissioning of nuclear power plant/unit start after issuance of commissioning permit by BNRA (hold point). If nuclear power plant is commissioned at several stages a commissioning permit is required for each stage. Such stages are initial on-site nuclear fuel storage, initial loading of the reactor core and testing at a subcritical condition, initial reactor criticality and low-power testing, initial power start-up of the unit at stage-by-stage power increase, trial-testing operation – for a new type nuclear reactor. Until the beginning of each commissioning stage, a commission of NRA inspectors appointed by the NRA Chairman shall inspect the site for confirming correspondence with stated data and circumstances and preparedness for carrying out the respective stage.

B. Expectations on licensees

The primary responsibility for the safety of NPP rests on the licensee. In order to ensure safety of NPP licensee shall take all appropriate measures to ensure that systems, structures and components are designed, manufactured, installed and commissioned with quality commensurate with their safety significance. Documentation of SSC is reviewed and approved by the licensee before its submission to BNRA.

National summary reports

The holder of permit for design and construction of nuclear facility is obliged to ensure that mechanical components are manufactured in accordance with the approved technical design. To ensure compliance with that requirement the license holder controls the manufacturing of the mechanical components at the place of manufacturing. The control includes verification of the quality system of the manufacturer as well as follow-up inspections. Necessary condition for start of manufacturing of a piece of equipment is the presence of coordinated design and technological documentation. For each piece of equipment is foreseen step by step acceptance of operations in hold points of Quality plans and in main stages called "Key events"(hold points). The inspection includes review of the quality management documents related to the manufacturing of the specific equipment, review and assessment for compliance of the submitted documents about the performed operations until the relevant hold point with the requirements set in the quality assurance documents and the design and technological documentation including reports (quality records), inspection of the real condition of the equipment/equipment's parts in manufacturing plant, check of the markings and interview of the personnel.

<u>Control of non-conformances</u>

Non-conformances in processes and products shall be assessed and documented according to procedures for control of non-conformances that are part of the quality management system of the Licensee. Depending of the significance of detected non-conformances appropriate corrective and preventive actions shall be taken.

C. Regulatory review of design documentation (RB/IO, safety class)

As it is mentioned above regulatory review of the design basis of nuclear facilities (nuclear power plant/unit, research reactor, etc.) is conducted as part of the licensing process (design permit and order of approval of technical design). Subject of review are technical design/project of NPP and Interim safety analysis report (ISAR), including topical reports for innovative design features, such as new passive safety systems or new structures and components. Topics of review for all safety classes SSC are following: component/structure design basis, operating experience and/or type test data, material specifications or construction materials, strength/structural analysis, structural or isometric drawings, coatings, quality assurance programs of licensee and main contractor/vendor of NPP and other subcontracting organizations that are involved in the design, manufacture and construction of NPP.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

Concerning the steel structures and mechanical components, as main practice, arranging a proper supervision with specific hold points during manufacturing is the responsibility of the License Holder. For selected most safety-significant structures and components NRA may conduct certain inspections during manufacturing. Requirements on these inspections and on the related hold points are given in connection with the approval of the technical design documents and construction permit.

E. Authorization of IOs

BNRA does not authorize inspection organizations. However, for the purposes of licensing and safety assessment, NRA may delegate an independent expertise on a given problem to inspecting organizations.

F. Use of management system audits/focused audits

BNRA conducts management system audits and focused audits of the licensee's activities, mainly on supervision of the licensee's procedures, competence and resources for conducting their own assessment of manufacturers and suppliers.

Appendix 4

National summary reports

WENRA, Inspection Working Group

CZECH REPUBLIC

NATIONAL SUMMARY

A. Basic regulatory approach in the country

The State Office for Nuclear Safety (SÚJB), [RB] is a governmental body as stipulated by Act. No.2/1969, Coll. and as regulatory body is responsible for governmental administration and supervision in the fields of uses of nuclear energy and radiation protection. The authority and responsibilities of the RB, as stipulated by Act. No.18/1997, Coll on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (Atomic Act), include the following issues in particular:

- State supervision of nuclear safety of nuclear facilities, nuclear items, physical protection of nuclear facilities, radiation protection, and emergency preparedness of nuclear facilities and workplaces handling ionizing radiation sources,
- Licensing of activities as specified by Act. No.18/1997, e.g. for the sitting, construction, particular stages of commissioning, operation and decommissioning of nuclear facilities
- Reviewing and approving documentation related to nuclear safety and radiation protection as laid down by the Atomic Act, limits and conditions for the operation of nuclear facilities.

The RB controls every stages of the NPP life cycle according to a/m Atomic Act and associated regulations. The RB inspections activities concern classified equipment defined by Regulation No.132/2008, Coll, on quality assurance (in activities related to the utilization of nuclear energy) and Regulation No.309/2005, Coll., on assuring technical safety defining classified equipment specially designed (in compliance with the European Parliament and Council Directive No.97/1997 [PED]).

Classified equipment specially designed – equipment their potential failure can cause release of radionuclides into environment and threaten of human health.

Independent inspection organizations control design, manufacturing and selected installation of classified equipment specially designed and performs assessment of accordance the respective nuclear pressure components, systems and structures with technical requirements and procedures specified executive legislation.

Hold point strategy

The strategy is anchored in principles of licensing process defined a/m Atomic Act and associated regulations.

B. Expectations on licensees (including control of non-conformances)

The licensee holder is responsible for the nuclear safety. Effective management of organization with clearly defined strategic objectives respecting desired level of nuclear safety, established processes assuring principal organizational activities including their supporting activities and feedback effectiveness evaluation are considered as basic organization’s capability.

Accomplishment adequate quality management system of the licensee holder is also substantial assumption for granting a licence and respective documents about quality management system is subject of RB approval before issuing RB a license according to a/m Atomic Act.

Recognised non-conformances are solved according to defined processes within quality management system of the licensee holder and categorised. In case of the category "one" with impact on nuclear safety - including classified equipment (categories "one, resp. two"), evaluation and fixation of non-conformances is submitted to the RB for approval resp. acceptance.
Existence of accredited inspection team of the licensee holder regularly performing inspections during installation, commissioning and operation stages, audits of licensee contractors (including designers and manufacturers) and internal organizational audits are also important aspects regarding to expectations on licensee holder.

The licensee holder demonstrates active co-operation with independent inspection organizations.

C. Regulatory review of design documentation (RB/IO, safety class)

Design documentation of new NPPs are elaborated in compliance with respective regulations and submit by licensee within licensing process, because design documentation of significant parts of the NPP from nuclear safety point of view is part of preliminary safety report and therefore is reviewed by RB. Inspection activities of the licensee holder are focused on quality management system of design organization performed by accredited inspection team.

In case of design documentation of NPP modification, licensee is obliged to categorize respective modification and those modification categorised as "category one" are submitted to the RB for approval. Others are only announced to the RB.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/ after manufacturing/ installation/commissioning/ clarification of Tables)

Regulatory inspections during manufacturing are performed randomly or as reactive inspections after indication of non-conformances related to classified equipment and classified equipment specially designed.

Regulatory inspections during installation, commissioning and operation stages related to classified equipment and classified equipment specially designed are specified as routine inspections performed by resident inspectors or special inspections performed systems inspectors from the RB headquarter. Those inspections activities are managed in compliance with internal RB guidelines and procedures and incorporated into semi-annual inspection plan.

In case of unplanned emergency reactor scram, indications potentially serious deficiencies regarding to quality management system, the QA processes, or non-conformances of components and structures indicating potentially common course failure, the RB applies reactive inspections.

Co-operation of the RB including information exchange with independent inspection organizations brings synergy effect regarding to performance of independent inspection activities.

E. Authorisation of IO

Independent inspection organizations are authorised by the Czech Office for Standards Metrology and Testing. The Office was established by the Czech National Council Act No. 20/1993, Coll, on the Organization of the State Administration in the field of Standards, Metrology and Testing as the state administration body responsible for such activities.

The RB (SÚJB) actively acts the role of advisory body for those Office and the RB inspection activities are regularly focused on overall status or a specific performance, etc. of the independent inspection organizations, including existence of respective contracts with licensee holder. Frequent

communication, e.g. on daily basis according to respective inspection requests is in competence of the licensee holder.
The RB does not receive any copies of documentation and conformance assessments issued by the independent inspection organizations, only on the base of individual requirements are these documents submitted to the RB.

F. Use of management system audits/focused audits

The RB inspections focused on the processes of the quality management system (QMS) of the licensee holder are regularly included into RB semi-annual inspection plans. Majority of those audits are focused audits or audits based on sampling. The RB also regularly performs audits/ inspections of the QMS contractors of the licensee holder and authorised independent inspection organizations.

Appendix 4

National summary reports

WENRA, Inspection Working Group

FINLAND

NATIONAL SUMMARY

A. Basic regulatory approach in the country (structures and components)

The Finnish nuclear regulatory body, Radiation and Nuclear Safety Authority (STUK), has the responsibility to supervise licensees' actions by reviewing design and manufacturing documentation as well as by inspecting NPP structures and components in various inspections defined in YVL Guides. Before STUK reviews any documentation, it must have been approved by the licensee. The licensee also performs and approves inspections before STUK's inspection. The licensees have the primary responsibility for the safety of NPPs and the quality of their structures and components.

Inspection Organizations (IO) carrying out regulatory type reviews and inspections of structures and components on behalf of STUK have to be authorized by STUK. IOs shall have accreditation specifically for activities related to NPPs and their structures and components. IOs can be utilized to review manufacturing and installation documentation, to conduct control of manufacturing and to inspect structures and components in safety class 3 and partly in safety class 2 (division of tasks between STUK and IO is presented in the Guide YVL E.1). Review of design basis is always at STUK's responsibility (SC1, SC2 and SC3).

<u>Hold-points</u>

Regulatory review of system level design basis is conducted as part of the licensing process (construction license and operating license) and when modifications of an existing system are made. System level design basis approval is a prerequisite to construction plan review <u>(hold point).</u>

Construction plans for manufacturing are reviewed. Approval of essential parts of a construction plan (e.g. design basis of component/structure, drawings, strenght/structural analysis, welding, heat treatment) is a prerequisite to start manufacturing of components <u>(hold point)</u>. For concrete structures also concreting plan shall be approved and readiness inspection shall be passed before manufacturing starts <u>(hold point)</u>.

In the construction inspection manufacturing documentation is reviewed, completed component/steel structure is inspected visually, its dimensions are checked and possible tests are supervised (e.g. pressure, leak tightness, functional, loading) (<u>hold point</u>). If inspection becomes more difficult as manufacturing proceeds, an adequate number of parts of construction inspection shall be carried out during manufacturing. Hold points during manufacturing are placed case by case. For concrete structures material test results of concreting are reviewed (witness point).

Installation construction plans of components are reviewed. Approval of the plan is a prerequisite to start installation <u>(hold point)</u>. In the installation construction inspection installed component/structure is inspected visually, possible tests are supervised and installation documentation is reviewed (<u>hold point</u>).

Commissioning inspection is STUK's <u>hold point</u> in safety classes 1, 2 and 3. In the first phase of the commissioning inspection it is verified that all the previous steps have been performed and documented as expected, the remarks and non-conformances from previous steps have been cleared, location of component is in accordance with the approved plans and possible accessories are inspected. For concrete structures concrete work report is reviewed. After this commissioning can proceed to the second phase and temporary permit for starting preoperational performance tests of components/systems can be issued. The second phase can be approved after the performance tests have been conducted and documented. After approved performance tests components/systems are ready for operation.

B. Expectations on licensees

The licensee has the primary responsibility for the safety of NPP structures and components. The licensee shall see to it that every organization the activities of which are connected to design, manufacture, installation, testing and inspection of structures and components, have appropriate quality and management systems and qualified personnel for the work they perform. The licensees must review and approve all the documentation, perform inspections and approve tests before presenting those to STUK for approval. The licensee is responsible that regulatory requirements and guides are followed.

<u>Control of non-conformances</u>

Licensee's management system shall include procedures to process non-conformances observed in processes and products. Significance of detected non-conformances shall be assessed, reasons for those determined and corrective and preventive actions decided. If necessary, modifications of the plant or components shall be made or procedures or management system shall be improved.

C. Regulatory review of design documentation (RB/IO, safety class)

Review of design documentation covers similar topics in safety classes 1, 2 and 3. The contents of reviews do not differ between STUK and IO. Topics of comprehensive review are following:

- manufacturer approval (pressure vessels) or description of manufacturer[22]
- NDT-organization approval[23]
- accreditation certificate of DT-organization[2]
- component/structure design basis
- operating experience and/or type test data
- material specifications or construction materials
- strength/structural analysis
- welding procedure specifications and their qualification[2]
- manufacturing procedures (qualification for demanding jobs)
- structural or isometric drawings
- coatings
- quality control plans
- NDT procedures including also functional, pressure and leak tightness tests.

Safety class 1:	-STUK makes the comprehensive review of components' design documentation. -SC1 is not applicable for steel and concrete structures.
Safety class 2:	-Review of components' design documentation is divided between STUK and IO depending on components safety significance. -Design documentation of steel and concrete structures is reviewed by STUK.
Safety class 3:	-IO makes the comprehensive review of components' and steel structures' design documentation. -Design documentation of concrete structures is reviewed by STUK.

[22] For concrete structures description of manufacturer contains description of the organization chain between the licensee, the plant supplier, the structural designer and the contractor.

[23] Not applicable for concrete structures

National summary reports

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

Safety class 1:
-STUK supervises manufacturing and makes construction inspections, installation construction inspections and commissioning inspections for components.
-SC1 is not applicable for steel and concrete structures.

Safety class 2:
-Components are divided between STUK and IO depending on components safety significance. STUK or IO supervises manufacturing and makes construction inspections and installation construction inspections, but STUK makes always commissioning inspections.
-STUK makes regulatory inspections for steel and concrete structures.

Safety class 3:
-IO supervises manufacturing and makes construction inspections and installation construction inspections for components and steel structures, but STUK makes always commissioning inspections.
-STUK makes regulatory inspections for concrete structures.

E. Authorization of IOs

a. Accreditation

IO applies accreditation from FINAS (the Finnish Accreditation Service) and the accreditation is based on the standard EN ISO/IEC 17020 (type A). When operation area of IO includes review of design documentation, IO shall be accredited also against EN 45011. If IO is situated abroad, FINAS will be the contact between a corresponding foreign organization. The operation area of IO, e.g. pressure equipment, hoisting devices, valves, pumps etc., the related standards and YVL Guides shall be defined in the application.

FINAS administers the accreditation process and STUK's specialist acts as an expert. When accreditation is approved by FINAS, IO can apply authorization from STUK.

STUK approves the IOs for the inspections based on the application submitted to STUK by the IO. The operation area of the IO is defined in the STUK's approval decision and the area is based on the IO's own profile. References in the approval decision are YVL Guides, applicable standards and STUK decisions. The authorized IOs are listed on the STUK web-pages where licensees can choose the suitable IO for their use.

b. Contracting, organizing daily inspection requests

The licensee makes a contract with the IO and invites the IO to the defined inspections.

c. Reporting to RB, oversight of IOs by RB

IOs have to report their actions monthly to STUK e.g. decisions in/decisions out, inspections and significant non-conformances. IOs are obliged to send also annual report and participate to annual experience feedback meetings.

STUK supervises IOs' activities by making observations at the NPPs or at the vendors' premises. STUK analyses IOs' reports as well as NPP plants' operation experience and ISI-reports. STUK also participates to annual evaluations related to the accreditation.

d. Conformance assessment

Conformance assessment is a stepwise process, the steps of which are hold points. These hold points are listed under title "A. Basic regulatory approach in the country (structures and components)" in this summary. For the IOs the hold points related to design base approval and commissioning are not applicable, because STUK is responsible for those phases.

At every hold point conformance is assessed and a certificate of conformance is given by a qualified person. The certificate is a basis to continue work and proves that the requirements related to the hold point are fulfilled.

F. Use of management system audits/focused audits

STUK participates to audits arranged by the licensee as an observer, but has the right to make observations and remarks and issue non-conformances. STUK gets an invitation to the audits mentioned below and attends them widely.

a. Which organizations are audited by the RB?
- safety class 1 and 2 components' design organizations
- safety class 1 and 2 pressure equipment manufacturers and installation companies approved by STUK
- safety class 2 steel structure design organizations, manufacturers and installation companies
- safety class 2 and 3 concrete structure design organizations, manufacturers and installation companies
- commissioning organizations

b. Is this auditing based on sampling?

No.

c. When do you use management system audits and when focused audits?

Mainly management system audits are used although auditing of manufacturing process is usually focused to the deliverable product. Focused audit might be used if there are doubts that some part of the design/manufacturing/ installation/commissioning process is not functioning properly.

STUK also performs own audits, especially in the construction license phase to assess the readiness of the licensee, plant vendor and main contractors to start construction.

Appendix 4

National summary reports

WENRA, Inspection Working Group

FRANCE

NATIONAL SUMMARY, Pressure Equipment

A. Basic regulatory approach in the country

ASN is the regulatory body in charge of oversight activities (Nuclear pressure equipment department and Nuclear Power Plant Department) meeting the Law of 13 June 2006.

ASN mainly involves third party organisations through a mandate. Difference is made between level N1, N2 and N3 equipments defined by the order of 12 December 2005 based on Pressure Equipment Directive (PED).

- N1 equipments: primary and secondary circuits and equipments which are essentials to maintain the nuclear power plant in safe conditions;
- N2 equipments: non classified N1 equipments and those which the failure can lead to radioactive waste higher than 370GBq.
- N3 equipments: non classified N1 and N2 equipments which the failure can lead to radioactive waste higher than 370MBq.

Third party organisation may be mandated by ASN for the N1 equipments.
Third party organisations are in charge of regulatory activities for N2 and N3 equipments.

Hold points are applied according to the inspection plan established by ASN or by the third party organisation. Inspection plan is mainly based on the risk analysis. There is no regulatory text regarding hold points, this is only “good practice”.

Appendix 4

National summary reports

The table below presents the modules issued from Pressure Equipment Directive (PED) applicable for conformity assessment of nuclear pressure equipments.

Pressure equipment categories from I to IV are defined in the PED.

<table>
<tr><th>Level</th><th>N1</th><th>N2</th><th>N3</th></tr>
<tr><td>Pressure vessel or pressure accessories or safety accessories Category I or II</td><td>H+G</td><td>B+F ; B+D ; G ; H1 ; B+C1 ; B1+F ; B+E ; B1+D ; H</td><td rowspan="3">See table 1 below, applicable conformity assessment procedures required by order 21/12/1999.</td></tr>
<tr><td>Pressure vessel or pressure accessories or safety accessories Category III or IV</td><td>H+G</td><td>B+F ; B+D ; G ; H1</td></tr>
<tr><td>Pipes</td><td>H+G
Pipes of primary circuit with nominal size (DN) ≤ 50 and others from categories I or II and for DN ≤ 100 :
B+F ; B+D ; G ; H1</td><td>B+F ; B+D ; G ; H1 ; B+C1 ; B1+F ; B+E ; B1+D ; H</td></tr>
<tr><td>Pressure accessories with CE marking</td><td>Non applicable</td><td colspan="2">If conformity assessment has been performed with module A, the equipment can not be used as nuclear pressurised equipment. For other cases, complementary assessment shall be performed by third organisation or inspection organisation.</td></tr>
<tr><td>Assemblies</td><td colspan="3">- Evaluation of each nuclear pressure equipment if necessary.
- Evaluation of elements integrated to assembly according to the highest level and category included in assembled equipments.
- Evaluation of assemblies between nuclear pressure equipments according to the highest level and category included in assembled equipments.
- Evaluation of the assembly protection according to the highest level and category included in assembled equipments.
- Final assessment and proof test, assemblies of equipment being constituted.</td></tr>
</table>

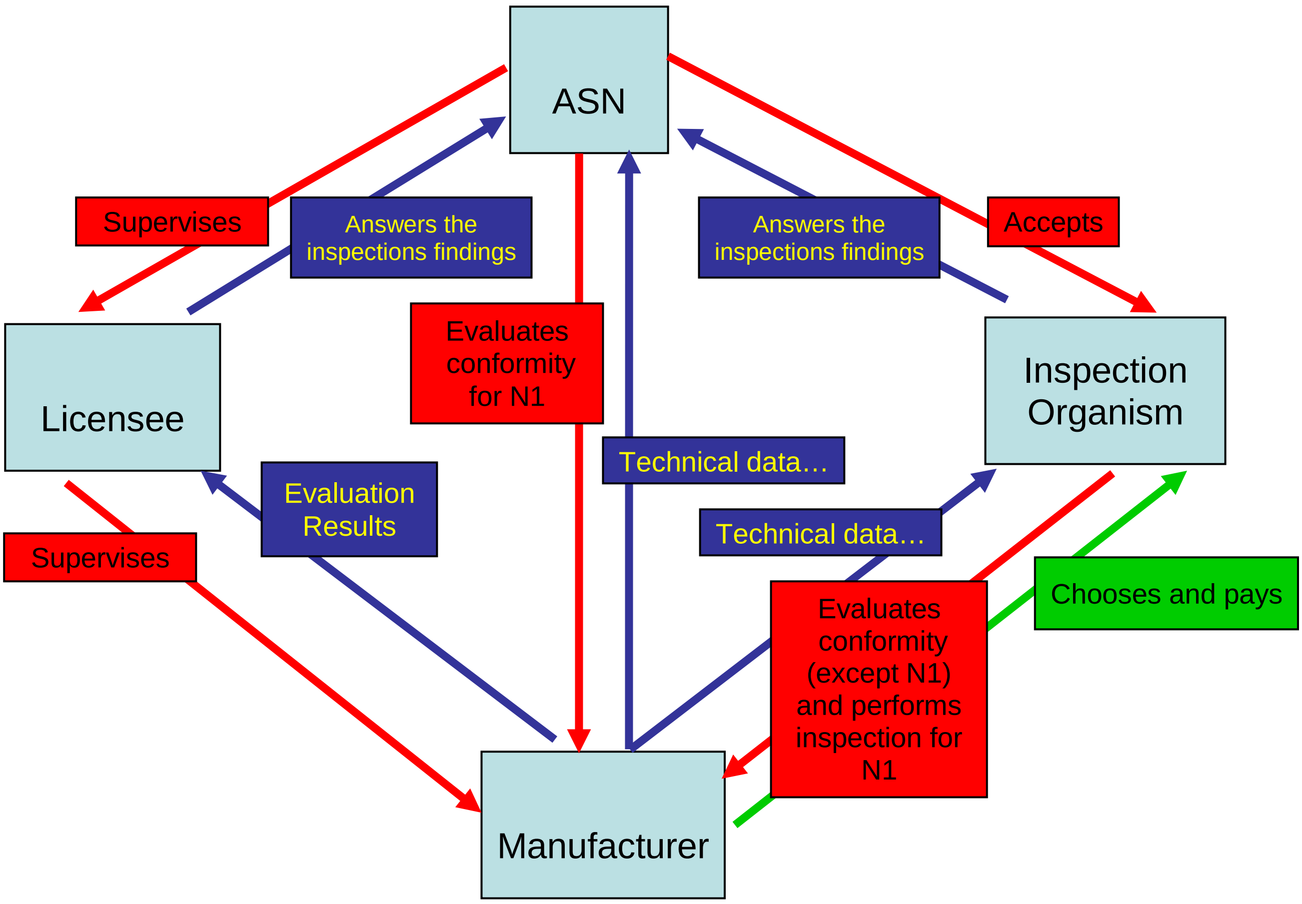
Table 1 :

<table>
<tr><th rowspan="2">Risk categories</th><th colspan="2">Without quality assurance</th><th colspan="2">With quality assurance</th></tr>
<tr><th>In series</th><th>Unit</th><th>In series</th><th>Unit</th></tr>
<tr><td>Cat. I</td><td colspan="2">A</td><td colspan="2">A</td></tr>
<tr><td>Cat. II</td><td colspan="2">A1</td><td colspan="2">D1 (Production) ou E1 (Product)</td></tr>
<tr><td>Cat. III</td><td>B+C1</td><td>B1+F</td><td>B+E or B1+D or H</td><td>H or B1+D</td></tr>
<tr><td>Cat. IV</td><td>B + F</td><td>G</td><td>B +D or H1</td><td>H1</td></tr>
</table>

Appendix 4

National summary reports

Description of the current conformity evaluation for nuclear pressure equipments (revision is on-going):



B. Expectations on licensees

- The licensee shall fulfill the requirements of the order of 10 August 1984 concerning quality assurance system.
- Identify important elements for nuclear safety (structures, equipments, assemblies and working conditions)
- Implement a quality management
- Demonstrate quality achievement for the identified components
- Identify activities concerned by the quality. These dispositions are applicable to all organizations working in the NPP.
- Define licensees' responsibilities, surveillance of sub-contractors and controls of furniture
- Write a manual for quality assurance and demonstrate its used
- Demonstrate adequacy between the human and technical means and the organization install to fulfill requirements
- Identify and qualify the personnel concerned
- Supervise quality concerned activities
- Analyse feedbacks from past experiences

- Inspector shall not be the operator
- Quality assurance system should be defined by independent and competent people. They regularly check its effectiveness and perform corrective actions in case of abnormal situations.
- Manual for quality assurance shall :
  - Define process of control, acceptance criteria and treatment of non conformities
  - Specify that inspections shall be described in reports
  - Describe actions program
  - Demonstrate actions performed
  - Assessment shall be drawn up
- Documents shall be saved and archived
- Describe anomalies and incidents, and their status
- Identify significant findings and notify the French nuclear Safety Authority
- Analyses shall be conducted and shall lead to a feedback. ASN shall be informed.

C. Regulatory review of design documentation (RB/IO, safety class)

Assessment shall be finished before pronouncing the conformity of the equipment assessed.

Documentation to be handed over and reviewed (ASN guide n°8):

- The licensee shall provide the manufacturer with a description of all the situations which may apply to the equipment, in accordance with the safety report of the installation for which it is intended, supplemented by the associated files as well as the loads to be taken into account for each situation.
- The manufacturer shall perform the risk analysis laid down in indent 3 of preliminary comments of annex 1 of Decree of 13 December 1999, taking account of the data provided by the user and the radioactive nature of the fluid that will be contained.
- The list of harmonized norms meeting requirements of ESPN decree of 12 December 2005
- Design and manufacturing drawings.
- Base material certificates
- Specifications dedicated to base and filler materials.
- Strength calculation reports
- Marking procedures
- Test reports for experimental design.
- Welding procedure qualification reports, welders qualification, heat treatment and NDE procedures and certificates of the personnel performing NDE.

National summary reports

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

Licensee

Pressure equipment life cycle																
	Design		Hold point		Manufacturing		Hold point		Installation		Hold point		Commissioning		Hold point	
Safety class	Who	How	P	O	Who	How	P	O	Who	How	P	O	Who	How	P	O
N1	L	TMFR		X	L	TMFR		X	L	TMFR		X	L	TMFR		X
N2	L	M		X	L	M		X	L	M		X	L	M		X
N3	L	M		X	L	M		X	L	M		X	L	M		X

Regulator

<table>
<tr><th colspan="17">Pressure equipment life cycle</th></tr>
<tr><th></th><th colspan="2">Design</th><th colspan="2">Hold point</th><th colspan="2">Manufacturing</th><th colspan="2">Hold point</th><th colspan="2">Installation</th><th colspan="2">Hold point</th><th colspan="2">Commissioning</th><th colspan="2">Hold point</th></tr>
<tr><th>Safety class</th><th>Who</th><th>How</th><th>P</th><th>O</th><th>Who</th><th>How</th><th>P</th><th>O</th><th>Who</th><th>How</th><th>P</th><th>O</th><th>Who</th><th>How</th><th>P</th><th>O</th></tr>
<tr><td>N1</td><td>RB/
RB,IO</td><td>CM/
CMR</td><td></td><td>X</td><td>RB,
IO</td><td>CMR</td><td></td><td>X</td><td>RB,
IO</td><td>CMR</td><td></td><td>X</td><td>RB,
IO</td><td>CTMR</td><td></td><td>X</td></tr>
<tr><td rowspan="2">N2</td><td rowspan="2">RB/
IO</td><td rowspan="2">TM/
CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td rowspan="2">IO</td><td rowspan="2">CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td rowspan="2">IO</td><td rowspan="2">CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td>IO</td><td>CTM</td><td rowspan="2"></td><td rowspan="2">X</td></tr>
<tr><td>RB</td><td>TMR</td></tr>
<tr><td rowspan="2">N3</td><td rowspan="2">RB/
IO</td><td rowspan="2">TM/
CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td rowspan="2">IO</td><td rowspan="2">CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td rowspan="2">IO</td><td rowspan="2">CTMR</td><td rowspan="2"></td><td rowspan="2">X</td><td>IO</td><td>CTM</td><td rowspan="2"></td><td rowspan="2">X</td></tr>
<tr><td>RB</td><td>TMR</td></tr>
</table>

National summary reports

E. Authorisation of IOs (if used)

a. accreditation

Third party organisations shall be qualified as notified organisation (decree of 13 December 1999) and shall be authorized by ASN (ESPN order of 12 December 2005).

They shall be in compliance with an ASN guideline based on ISO 17020 (guide ASN n°5).

b. contracting, organizing daily inspection requests

In practise, for N1 equipment, manufacturers propose a third party organisation per equipment and usually ASN accepts this proposal (but can also refuse).

Manufacturers are also contractors.

ESPN order of 12 December 2005 emphasized manufacturers' duties for pressure equipment.

c. reporting to RB, oversight of IOs by RB

For N1 equipment, latest mandates specify a monthly and a final report for the inspections performed by third party organisations.

Organisations may be subjected to inspection by ASN in the frame of the mandate.

For N2 and N3 equipments, third party organisations are working independently but their actions can also be supervised by ASN to make sure their agreement is still valid.

d. conformance assessment

For N1 equipment third party organisations are performing inspections following their own guide. Inspections reports are provided to ASN.

With regards to results of inspections conducted for each equipment (module G - PED) and to quality management assessment of the manufacturer (module H - PED), ASN would stamp the component.

For N2 and N3 equipments, third party organisations are working independently (except for the description of the situations which may be applied to the equipment, which is reviewed by ASN).

F. Use of management system audits/focused audits

a. which organisations are audited by the RB?

ASN performs agreement of third party organisations and inspection organisations (belonging to a licensee).

The manufacturer's management system is audited by a third party organisation, which is accepted by ASN (module H - PED). ASN can assist to those audits by supervising the third party organisation.

b. is this auditing based on sampling?

All points included in ISO 17020 “General Criteria for the operation of various types of bodies performing inspection » and additional requirements described in ASN guide n°5 should be audited within 3 years.

c. when do you use management system audits and when focused audits?

Audits are performed following ISO 17020.

Appendix 4

National summary reports

WENRA, Inspection Working Group

LITHUANIA

STATE NUCLEAR POWER SAFETY INSPECTORATE SUMMARY

The State Nuclear Power Safety Inspectorate (VATESI) have regulatory oversight experience of operating Nuclear Power Plant including design modification, experience of design, construction and operation of Radioactive Waste Management Facilities, including Spent Fuel Storage Facility, but have no regulatory oversight experience in design, construction and commissioning process of Nuclear Power Plants.

A. Basic regulatory approach in the country

VATESI is the main nuclear safety regulatory institution, which sets safety requirements, controls whether they are complied with, issues licences and permits, performs safety assessments and other regulatory functions.

VATESI performs review and inspection activities which cover all important aspects of site evaluation, designing, manufacturing, construction and commissioning, operation and decommissioning of NPP. An important part of these aspects is using of proven technologies at all stages of NPP life cycle. The oversight of VATESI inspectors focuses in the safety culture, the Regulatory Body (RB) also verifies that the licensee implements its safety and quality management.

The basic objective of the review and assessment is to determine whether the operator's submissions demonstrate that the facility complies throughout its lifetime with the nuclear safety requirements approved by the regulatory body.

Sampling of activities is undertaken to demonstrate conformity with the regulatory requirements.

Hold-points

As VATESI has only regulatory oversight experience of operating NPP the hold points were used only during the plant modifications in accordance with regulatory document "Requirements for modifications in nuclear facilities". Additional to that Ignalina NPP have to agree with VATESI all technical proposals, which are related to safety (nuclear fuel unloading program, safety related components in service inspection manual, programs and methodology and others).

For the new Nuclear Facilities currently it is foreseen some key hold points: approval of technical specification of Nuclear Facility; issue of Construction license; approval of commissioning program. Taking into account expected scope of activities related with new NPP VATESI is going to determine hold points during regulatory oversight of design, manufacturing, installation and commissioning of safety related systems and components of new NPP.

B. Expectations on licensees

Licensee has the prime responsibility for nuclear safety, ensuring the quality, control and supervision of the construction activities and organizations that are involved in the design of structures and components, material production, component manufacturing, installation, construction, testing, and inspection.

National summary reports

In respect to the agreed by VATESI technical design and Construction License requirements Licensee should verify and ensure that all manufactured concrete and steel constructions or mechanical components meet their design requirements, also verify how the manufacturer is complying with its quality management system, handling of base and welding material, calibration of working and testing equipment, management of subcontracting, qualification of personnel etc. In addition to this Licensee should prepare and present to VATESI the schedule of performing works where, according to the license requirements should be determined in details activities, related to safety important systems and components design, off-site manufacturing, construction, installation and commissioning.

According to the presented schedule VATESI determines the most important topics and prepares regulatory inspection programs/plans and oversights, controls how Licensee follows the determined requirements during manufacturing related to structures and components. The Safety Class of structures and components is taken into account on preparing the inspection programs and determining the scope of inspection.

Licensee should report VATESI about the manufacture and supplement of all safety related mechanical components, steel and concrete structures: the list of all contracted manufacturers and providers, information and results of all performed inspections and management system audits, all data about fixed discrepancies and non-conformances according to safety requirements and standards, the reasons of non-conformity, taken corrective measures and administrative actions.

<u>Control of non-conformances</u>

In respect with the regulatory requirements “Requirements of Management System” Licensee must identify the reasons and significance of all discrepancies and non-conformances, provide for and check the conformance, adjustment, other corrective measures and administrative actions to remove the potential and(or) identified reasons of discrepancies in order to prevent any further non-conformances. Once the negative trends of processes or other non-conformances according to the requirements, which adversely affects or could affect nuclear safety, are identified it is necessary to analyze the factors which led to such cases also to establish and carry out corresponding improvement actions.

C. Regulatory review of design documentation (RB/IO, safety class)

VATESI review and approves technical design documentation of mechanical components, steel and concrete structures, also PSAR, which are the bases for Construction License.

In accordance with legislation VATESI issues license only to operating organizations, no other organizations are licensed in the framework of nuclear safety regulation.

Independent design verification is required together with Site Evaluation Report during sitting of NPP and Preliminary Safety Analysis Report before the construction.

Regulatory oversight activities foreseen by VATESI during the design, manufacture, construction and commissioning of steel/concrete structures and mechanical components in general we can describe dividing activities into 3 general stages (this concept of regulatory oversight activities is foreseen in drafted Law on Nuclear Safety):

<u>1 Stage.</u> Safety assessment and review before construction:

VATESI performs review and assessment of design documentation and preliminary safety analysis report (PSAR) of steel/concrete structures and mechanical components against established national safety requirements and good practice (for instance IAEA safety standards) before construction license is issued.

2 Stage. Inspections/safety assessment and review during construction and before the permit for first transportation of fuel into the site:

VATESI performs inspection activities during manufacture, construction and cold performance tests of steel/ concrete structures and mechanical components which are important to safety. The Safety Class of structures and components is taken into account on determining the scope of inspection. Before receiving a permit for first transportation of fuel into the site license holder shall prepare and agree with VATESI updated safety analysis report.

3 Stage. Inspections/safety assessment and review after permit for first transportation of fuel into the site and before commercial operation:

VATESI performs inspection activities of steel/ concrete structures and mechanical components which cover important aspects of plant safety. The Safety Class of structures and components is taken into account on determining the scope of inspection. Before receiving a permit for commercial operation and starting commercial operation license holder shall prepare and agree with VATESI final safety analysis report.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

VATESI has implemented safety classification model in General requirements for design of NPP and Nuclear Safety Requirements for modifications of Nuclear Facilities according to the IAEA safety standards. All structures, systems and components are classified into 4 Safety Classes on the basis of their perform function and significance with regard to safety. Mechanical components, steel and concrete structures are designed, constructed and maintained such that their quality and reliability is commensurate with this classification.

Licensee is responsible for full scope supervision and maintenance of mechanical components, steel and concrete structures of all Safety Classes. VATESI regulatory oversight based on sampling covers NPP structures and components, which are related to safety (safety classes 1-3). Safety Class 4 (non-nuclear safety) is under supervision of other competent institutions according to Lithuania legislation and in respect with EU directive 97/23/EC.

According to the current legislation there is not foreseen obligatory provision of use of any third-party organizations by VATESI nor by Licensee for support in inspection activities. As a result VATESI performs regulatory oversight activities by sampling taking into account the labour force and scope of structures and components or pressure equipment.

The Safety Class of structures and components, the best practise and international recommendations are taken into account on preparing the inspection programs/plans and determining the scope of inspections. Steel and concrete structures are classified according to design assigned safety functions.

E. Authorization of IOs

During regulatory oversight of Ignalina NPP the support of Third-Party Organizations were not used. Design modifications of safety related systems and components, including lower safety classes, during operation of Unit 1 and 2 were implemented under supervision of VATESI. Taken into account expected scope of regulatory oversight activities during design, manufacturing, installation and commissioning of safety related systems and components of new NPP VATESI will consider possibility to use Third-Party Organizations after review of other WENRA countries inspection practice report and recommendations.

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F. Use of management system audits/focused audits

It is foreseen that all management system audits shall be carried by the Licensee according to VATESI requirements „Requirements for Management System", which were approved in 2010. Additionally licensee is responsible for organizing independent reviews and audits of management system. Regulatory Body does not carry any audits, but have the right participate in any audits arranged by Licensee as an observer. Any discrepancies and non-conformances fixed during observation are presenting directly to the Licensee.

According to the „Requirements for Management System" it is required that the documents of management system should determine and foreseen the analysis and justification of nuclear facility design conformity to the nuclear safety requirements, including independent verification of design results performing alternative calculations based on other computational techniques.

Licensee is responsible for ensuring performance of independent verifications (management system audits, license application draft documents review and other independent verification). Licensee must conduct all audits in the process to determine if the management systems and processes meet the requirements for management system and other requirements, if management systems and processes are effectively implemented and improved, processes and management system improvement opportunities.

The independent verification should be carried out under the responsibility of the operating organization by a team of experts who are independent of the designers and those performing the safety assessment. Personnel are considered independent if they have not participated in any part of the design and safety assessment. The aim of the design verification is the comparison of the design results and the original design input. This independent verification is in addition to the quality assurance (QA) reviews carried out within the design organization.

Currently Ignalina NPP (Licensee) is performing audits of their contractors and reporting about these audits results to VATESI. Additional to that VATESI specialists performs inspections in components manufacturing facilities with the aim to verify how Licensee carries out their audits and inspections.

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RUSSIA

NATIONAL SUMMARY

A. Main approaches to regulation in Russia (equipment and components)

Regulatory authority in the field of nuclear and radiation safety of Russia (Rostechnadzor) reviews and assesses main documents justifying safety of NPPs developed by designers of NPPs. When necessary, specialized organizations are engaged by Rostechnadzor to participate in the review and assessment.
Rostechnadzor reviews justifying documents and issues licenses to the Operators for siting, construction, commissioning, operation and decommissioning of NPPs. The review of system design documents forms part of the licensing procedure.

Also, Rostechnadzor or its regional representative authorities issue licenses to designers and manufacturers of equipment for the right to develop design documents for NPP components and for manufacture of equipment.

In order to verify that the license conditions are being fulfilled by organizations and that organizations' activities meet the requirements of safety regulations, Rostechnadzor performs inspections (supervision) at all stages of construction and operation of NPPs and requires to submit the necessary information.

In any case the responsibility for NPP safety rests with the organization holding a respective Rostechnadzor license (i.e. the licensee).

To perform safety assessment and analysis Rostechnadzor engages its subordinate organization, Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS). To control the design and manufacture of equipment, Mandated organizations are appointed. To assess process issues (use of materials, welding, nondestructive and destructive tests), specialized organizations approved by Rostechnadzor are engaged.

The review and approval of design and process documents (drawings, calculations, welding and heat treatment procedures) are performed before the start of manufacture of equipment and its components by engaging Mandated organizations. Supervision of manufacture of 1, 2 and 3 safety classes components is entrusted with Mandated organizations and inspection organizations on contract basis with the licensee. Documents are reviewed and supervision is performed to verify compliance with Rostechnadzor documents.

Supervision during construction of building structures is performed in accordance with Russian federal law "On self-regulating organizations" by appointed organizations who perform certification of the performers of the works. The process of construction is continuously followed-on by the licensee.

During design and manufacture of new or one-of-a-kind equipment, sample tests are performed under supervision of a committee consisting of the licensee, the customer, a Mandated organization, the manufacturer and the designer.

Supervision over manufacture of 1, 2, and 3 safety classes equipment is performed by the Mandatory organization together with the licensee, engaging, when necessary, inspection and specialized organizations, in which case the functions are being divided between the mentioned organizations according to respective Rostechnadzor and licensee documents.

B. Licensing development of documents and manufacture

Licensing is preceded by examination of the prospective licensed organization by specialists of Rostechnadzor and/or Mandated organization.

During examination Quality Management System of the organization is audited and capabilities of the organization to design and manufacture equipment fulfilling the specified safety and quality requirements is assessed.

Non-conformance control

Decisions on non-conformances regarding 1,2, and 3 classes components which have an impact on safety are eliminated. In case it is impossible to eliminate them, the decisions to admit non-conformances are being agreed upon with the designer, the customer, material engineering organization, and are approved by Rostechnadzor.

Decisions on non-conformances which have no impact on safety are taken by the licensee with participation of the designer.

C. Review of Design Documents

Rostechnadzor reviews and approves system documents developed by the designer (Safety Analysis Reports (PSAR, FSAR), Probabilistic safety analysis, and other system documents).

Mandated organization reviews design documents on components to verify that they comply with the safety regulations. Documents on 1 safety class components are reviewed engaging, when necessary, independent specialists and organization.

New components are certified by organizations accredited according to the established procedure and tested in laboratories accredited by the licensee.

D. Inspection of equipment and components during design, manufacture, installation, and commissioning

For components of 1, 2, and 3 safety classes, acceptance is performed at the manufacturer by the specialists of the Mandated organization and the licensee to verify the compliance of the components to Rostechnadzor regulatory requirements and the requirements of the designer. Acceptance procedure is determined by Rostechnadzor and licensee documents.

Supervision over components during their installation and commissioning is performed by the licensee. Rostechnadzor approval is required for the start and completion of individual procedures.

E. Accreditation of inspecting organizations

Inspecting organizations (IO) contracted to perform certain works shall be approved by the licensee and, in some cases, by Rostechnadzor. IOs perform their activities under contracts awarded by licensees.

Regarding foreign companies who develop and manufacture components for Russian NPPs, there is a special control procedure approved by Rostechnadzor.

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IOs report on the progress of work to licensees, and in any case Rostehnadzor maintains full access to IOs reports. Assessment of non-conformances found by IOs is performed according to the procedure mentioned above.

F. Audits of organizations' Quality Management Systems

QMS audits are performed by Rostechnadzor during licensing stage, and by licensees during stage of awarding contracts for design and manufacture of equipment and components, and during the process of manufacture with use of Mandated organizations.

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SLOVAK REPUBLIC

NATIONAL SUMMARY

A. Basic regulatory approach in the country

Nuclear Regulatory Authority of the Slovak Republic (**NRA SR** – **ÚJD SR**) is a central administrative state office of the Slovak Republic responsible for the nuclear regulatory activities. The Licensees are fully responsible for nuclear safety.

hold-point strategy

In Slovak Republic Regulatory body (**RB**) controls every stages of the Nuclear Facilities – design, construction, commissioning and decommissioning. For all stages are applied hold - points as a necessary condition to go into next stage.

Technical and administrative preparedness from design through Construction to decommission permission, lies on the licensee according to §3, §5 and §10 of the Atomic Law. Every violations in technical specifications, non-conformances, design modifications, new procedures (welding, maintenance, repairs, quality inspections, operation) have to be approved by the RB before realization.

Hold points during manufacturing are not applied from the point of RB view, it is in the licensee responsibility. Concrete structure material is tested continuously by accredited test organization, and results are stored by licensee and have to be available to the RB on request.

B. Expectations on licensees

-includes control of non-conformances

According to Atomic Law emphasize responsibility for nuclear safety is putt on the licensee. On the basis of this the licensee is responsible for organizing quality controls. QA and QC are performed by accredited inspection team from utility or accredited 3rd party organizations. This ISI body issue protocols Conformity with quality documents. All of the protocols must be acceptable before unit commissioning. Accredited inspection of the 3rd party bodies communicates with and report directly to the NPP licensee. Results from their work have to be available to the RB on request. Third-party inspection bodies shall report general problems and serious deviations also to the expert meeting for ISI evaluation organized before the NPP starts up with the regulatory participation. Utilities apply the national and international IO with notification for the NPP area.

As independent third-party organisations the licensee during refuelling and overhauling cooperates with the manufacturer for supervisions during maintenance and repair of most important components (manufacturer of RPV, MCT, SG, etc.).

C. Regulatory review of design documentation (RB/IO, safety class)

The RB activity has focus on review documentation according to legislative documents (Law and Regulations), manuals and guides. According to licensee internal rules every documents prepared to submission to the RB shall be approved by the licensee. The quality documentation is listed in

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Regulation No. 56/2006 Col. specify requirements on important documentation which have to be submitted to the RB before realization for receiving decision. During assessment of complicated safety documentation submitted according to §10 of the Atomic Law the RB utilizes 3rd party for independent assessment. For the RB technical support organization (TSO) if could be an expert organization, research institutes, universities, etc. Every design modifications have to be submitted to the RB with the Ministry of environment statement based on European requirements.

The RB according to §4 article 2, letter a), number 2 of the Atomic Law reviews and approves every important documents such as the design basis, categorization of classified equipments into safety classes, Technical specifications, quality assurance documentations, quality requirements on classified equipments, implementations of modifications of nuclear facilities and list of classified equipments. NRA issues agreement or decision for sitting of nuclear facilities or implementation of modifications.

List of documentations which are necessary to present to RB for individual stage of nuclear facilities are in appendix No.1 of the Atomic Law. The documentation which is necessary for following stages of nuclear facilities:

- sitting of the nuclear facility (permition)
- construction (decision)
- commissioning and operation (decision)
- decommissioning (decision)

RB reviews and approves following documentations:

- design documentations and design modifications (with design basis)
- quality plan for each individual classified equipments
- classification of the equipment into safety classes
- pre-service inspection and ISI programs
- technological documentation of the manufacture, assembly, repairs and modifications of classified equipment,

Quality requirements of the classified equipments is reviewed and approved in two stages:

1. Design stage
2. Manufacturing and assembly stage

Review of the components' design documentation for every safety classes is fully in RB competence.

In decision No. 68/2007 are prescribed list of regular announcements for the RB information in which the licensee has a duty to submit to the RB following documentation (status of the nuclear devices during operation – components failures, quarterly and yearly safety assessment of the operation, refuelling and outage plan, ISI evaluation report, and other details) to create picture about safety operation.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

According to legislation the RB does not have competence to supervise on–site manufacturing process. The RB performs on-site inspection during construction and operation of nuclear facilities by sight inspectors and by inspection teams in accordance with the yearly inspection plan. The main purpose of the inspection during the construction is to oversee assembly of classified components mainly primary site, their testing and reviewing documentation. On-site inspections check, by multi professional team focused on procedures for verification, actual equipment status and their conformance verification according to valid documentation. The outputs of these inspections are written reports or, in case when non-conformances are found, the outputs are protocols with the important observations and corrective measures which have to be implemented by the licensee. The RB in complicated cases if it is necessary to employ expert assessment utilized by 3rd party organization for review details (during repairs using welding procedures, NDT tests etc.).

The RB oversees to the IOs activities during overhaul and assesses conformance with valid documents. The Authorised organization certificates conformity with regulations, it is a condition for taking a component or system for operation.

E. Authorisation of IOs

a. Accreditation

SNAS – Slovak National Accreditation Service

SNAS is Governmental Organization and Central State Administration Organization for technical metrology, for standardization and Conformance assessment applying Slovak and European Standards and European Directives, manage and organize accreditation system in SR.

SNAS accredits each organizations base on request for specific activity with time limitation. The RB doesn´t control accreditation of IOs. The licensee is responsible for assuring accredited IO in tender. RB verifies if IO was certified.

The RB defines minimal requirements for qualification of the systems for NDT in nuclear field in Regulatory guide BNS II.5.4/2009 - Qualification of systems for non-destructive examination in nuclear power engineering.

Substantial documents for each of the licensee are Quality Management and Quality Control system (QM), (QC). The RB has been approving QM and QC system. Application of these systems is verified by the RB once in three years and by certified auditing organisation, for example Det Norske Veritas, TÜV SÜD, etc., accredited in accordance with international standard ISO 9001 based on utility request.

b. Contracting, organising daily inspection requests

Organising daily inspection is fully in licensee responsibility and competence. Scope and organization of inspection is based on valid QC procedures for selected components.

c. Reporting to RB, oversight of IOs by RB

During inspection the RB has controlls personnel certificates, ISI group structure and scope of training. The accredited IO submitts its accreditation to licensee during selection process only. The IO accreditation has to be available to RB on request. RB doesn´t perform an oversight of IOs.

d. Conformance assessment

The accreditation is realized fully in conformance with European and national standards and directives.

All requirements which licensee has to observe are specified in the legally binding Slovak Atomic Act and Slovak RB Regulations.

Requirements for a quality management system are specified by the International Standard ISO 9001:2008 documentation.

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F. Use of management system audits/focused audits

a. Which organisations are audited by the RB?

The RB performs audits of licensee only based on international standard ISO 19 011 and IAEA recommendations. Except for RB, licensee is audited by 3rd party - independent organizations with SNAS accreditation base on request.

b. Is this auditing based on sampling?

The RB has audited licensees based on sampling according to regulatory internal procedures.

c. When do you use management system audits and when focused audits?

Licensees organize internal and external audits. Internal audits are managed by self audit team to recognized internal faults, and external audits which identify mainly structural faults. External audits are realized by 3rd party accredited organization. The RB can require audit to the licensee when we need to identify responsibility for an incorrect activity. Management system audit shows complexity of the licensee organization structure. Management system audit is applied mainly during a utility assessment personnel reduction, personnel optimization etc. to offer detailed information.

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WENRA, Inspection Working Group

SPAIN (CSN)

NATIONAL SUMMARY

Discussion of regulatory approach for design modifications on Nuclear Power Plants in operation

A. Basic regulatory approach

For each design modification the **licensee** must perform and analysis to verify if the criteria, rules and conditions on which its permit is based are still met. If from the analysis conducted the licensee concludes that criteria, rules and conditions are still meeting, the licensee may carry out the modification or tests periodically informing the Ministry of Industry and CSN (is the case "B" in the next paragraph). In the event that the modification entails a modification of criteria, rules and conditions on which operating permit is based, the licensee must request for a modification authorization (is the case "A" in the next paragraph .

Consequently **CSN (Spanish Nuclear Regulatory Body)** adopts a proportionate approach to regulation based on the two main circumstances explained in the foregoing paragraph. That is:

(A) If it is granted a permit (authorization) for the design modification.

In this case the actuations of the Regulatory Body are close to a comprehensive one (consult tables to see some exceptions as in manufacturing).

(B) If it is not necessary to grant a permit for the design modification.

In this case sampling of regulatory nuclear activities is undertaken.

Regardless of the aforementioned authorization when in opinion of CSN or Ministry of Industry the modification is far-reaching or entails significant construction or assembly works national regulators shall require the licensee to apply for a modification execution and assembly authorization (is also the case "A" in the foregoing paragraph)

The hold points required by regulators in the different phases of the equipment life cycle depend on the permit or authorization granted. The European Directive on pressure equipment must be complied. This is the only regulator <u>hold point for the equipment.</u>

The other <u>hold points</u> that CSN applies are related to the design modification as a whole process (not to each equipment). That is: the hold point is the permit to be granted itself (obviously this permit implies a review of the documentation of design basis, structural analysis, material specifications, non conformance and corrective actions, etc, but the asses done by CSN staff does not imply the approval of each document in an individual format). In these sense there are two possible hold points in a design modification:

1. <u>Hold point or permit</u> which shall have to be effect prior to the modification coming into service.
2. <u>Regardless of the hold point</u> already mentioned a <u>previous hold point or permit</u> for assembly in the cases when in opinion of Ministry of Industry or CSN the modification is far

reaching or entails significant construction or assembly work. In this case the licensee cannot start activities of assembly or construction prior to the granting of the corresponding authorization

If during sampling inspections the inspectors detect Non Conformances than the licensee has not detected by himself or in the case that the correction actions are not appropriated, CSN performs reactive inspections and can require further activities to the licensee (as strategic plans that CSN asses and audit)

B. Expectations on licensees

Control, analysis, and verification of every step in the different processes in design, manufacturing, installation and commissioning of the equipment and structures in accordance with permits and all nuclear standards applicable.

Non conformances must be analyze and resolved.

C. Regulatory review of design documentation

- *Regulatory body (CSN)*
  - Control and approval of design if it is associated to a modification license. The analysis of design of pressure equipment, steel structures and concrete structures associated to the modification is included in the assessment for approve.
  - If is not require a license the design control is by sampling.
  - No other external verification

D. Inspection and control responsibilities of structures and components

D1 Inspection and control responsibilities in manufacturing and installation

- *Regulatory body (CSN)*
  - Technical control and control of the management system of manufacturers and assemblers (installation in facility) of safety equipment is done by sampling. If has been granted an authorization for assembly the licensee cannot start any activity of assembly prior to the granting of the authorization (hold point).
  - No other external verification under the direct supervision of CSN. CSN does not audit or give an authorization to the IO´s
- *Licensee*
  - His activities are regulated by permits and nuclear Spanish standards in safety equipment. The licensee controls and approves manufacturing and installation. They can contract IO's for these jobs.
  - The IO´s contracted by the licensee to inspect manufacturing and assembling have to be audited and approved (by him) respect to nuclear standards. The IO´s report to licensee. The manufacturers report to licensee.

D2 Inspection and control responsibilities in Commissioning

- *Regulatory body (CSN)*
  - Attend to final proofs of the design modification associated to an authorization and review final documentation and corrective actions of non conformances.

- Attend to specific proofs of pressure equipment and special activities in the manufacturer facility.

There is an external and mandatory administration control of the accomplishment with conventional regulations of pressure equipments (European Directive of Pressure equipment). This is competency of the Ministry of Industry who grand specific authorizations. This job is transferred to autonomic governments and practices can vary, but not in the important things. The Ministry of Industry contracts IO´s accredited, with a quality system certificated. They are audited by governmental personnel.

- *Licensee*
  - His activities are regulated by permits and nuclear Spanish standards in safety equipment. They control and approve the jobs.
  - He (and manufacturer) most report to CSN the defects found in equipment if they are already ensemble in plant.

E. Authorization of IOs

- Not applicable for CSN
- IOs contracted by the Ministry of industry to control the accomplishment with conventional regulations of equipments (European Directive of Pressure equipment) are accredited, and they have a certificated quality system. They are audited by governmental personnel. They report to Ministry of Industry.

F. Use of systems audit/focused audits

- *Regulatory Body (CSN)*
  - *CSN performs specific* audits to the management systems of licensee and principal enterprises of engineering if the modification is associated to a license. If some problem is detected can be programmed a focused audit or a reactive audit. As part of the authorization these audits are hold points.
  - If the modification is not associated to a design modification CSN performs periodical audits to the licensee. If some problem is detected can be programmed a focused audit or a reactive audit, and as consequence of them may be required further activities to the licensee (as strategic plans than CSN asses and audit).

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SWEDEN

NATIONAL SUMMARY

A. Basic regulatory approach in the country (structures and components)

The Swedish nuclear regulatory body, Swedish Radiation Safety Authority (SSM), has the responsibility to supervise licensees' actions by reviewing design, manufacturing, construction, installation documentation as well as by inspecting NPP structures and components to assess whether the requirements in SSM's safety regulations, SSMFS, are met. Before SSM reviews any documentation, it must have been approved by the licensee. The licensee also performs and approves inspections before SSM's inspection. The licensees have the primary responsibility for the safety of NPPs and the quality of their structures and components.

Inspection Organizations (IO) that carrying out regulatory type reviews and inspections of structures and components have to be accredited for their tasks by Swedish Board for Accreditation and Conformity Assessment (SWEDAC) under SWEDAC's general accreditation rules and regulations issued by the Swedish Radiation Safety Authority (SSM). For some mechanical components notified bodies have assessment and inspection tasks during manufacture according to European directives (such as Pressure Equipment Directive, PED). Reviews and inspections by independent IOs are prescribed by SSM for all mechanical components and structures in all safety classes, but with varying extent. Review of design is performed by SSM.

It should be noted that the current Swedish regulations for inspections of mechanical components and concrete structures are based on the present situation with the operation and modification of the ten existing NPP. This means for example that no major changes are made to the reactor containment and other concrete structures. If new nuclear power plants will be built in Sweden, parts of the present control scheme, with different types of accredited organizations, need to be modified.

<u>Hold-points</u>

The design basis including all boundary conditions for design, coming from plant/systems level design, shall be notified to SSM before an IO review component level design documents. The IO will consider in its review comments and observations from SSM on the design basis. If the IOs review, which may include their own control analysis, confirm that relevant requirements are met, the body issues a design examination certificate. (Hold point).

The design examination certificate is in principle a prerequisite to start manufacturing. In practice however, manufacturing starts from time to time at the commercial risk of the vendor or the licensee before they have a design examination certificate. During manufacturing accredited laboratories, or the manufacturing organisation under supervising by an IO, perform testing in different phases. Qualification of welding procedures and welding personnel are supervised and evaluated by an IO. All results are reported to and reviewed by IO. If these reviews confirm that relevant requirements are met, the IO issues a manufacturing examination certificate. (Hold point).

The manufacturing examination certificate is in principle a prerequisite to start installation. During installations accredited laboratories perform testing according to the control and examination plans. After the installation of a mechanical component in the plant shall

- an IO verify that the component has been installed in accordance with examined drawings and flowcharts, and that performance meets safety requirements,
- an commissioning/operation tests have been done to demonstrate that safety valves and other safety equipment operates properly and that the device not is exposed to harmful vibrations or other loads, for which no account has been taken in the design. An IO shall witness the operation tests.

If these verifications and tests confirm that relevant requirements are met, the IO issues an installation examination certificate. (Hold point).

Before new mechanical components and modified plant systems can be taken into operation an IO conducts a final assessment based on previously issued design, manufacture and installation certificates. If all examinations, supervisions and controls confirm that the requirements are met, the IO issues a certificate of conformity with relevant SSM regulation. This certificate of conformity with SSM regulation is a legal condition for taking a component or a system in operation. (Hold point).

This means that any outstanding issues or discrepancies must be resolved at that time. In situations when a certificate of conformity not can be issued for any reason the licensee must apply for exemption, based on necessary safety justifications. Such applications are then assessed and decided by SSM. (Hold point).

For major plant modifications, such as thermal power up-rates, SSM reviews commissioning test programs, inspect and follow the test operation, which normally lasts for an full operational period with subsequent maintenance outage.

B. Expectations on licensees

The licensee has the primary responsibility for the safety of NPP structures and components. The licensee consequently has to take all necessary steps and actions to fulfill applicable safety requirements. This includes audits and controls of the licensee's manufacturers, suppliers and contractors. Particular for contractors used at NPP site the licensee should

- ensure that contractors has sufficient manpower and competence to carry out the assignment in a safe manner,
- ensure that contractors has the necessary equipment for executing the assignment and that the contractors employs adequate methods and processes where applicable,
- ensure that contractors employs management and quality systems that provide full control over safety in conjunction with the assignment and that manufactured and assembled structures, systems, components and devices meet stipulated safety requirements,
- continuously supervise the contractor's activities to ensure that all regulatory requirements and license conditions are satisfied, along with the goals and guidelines for the activity to which the assignment pertains,
- continuously follow up the contractor's evaluation and reporting to the licensee of events and ensure that appropriate safety-related measures are taken,

- when necessary, instruct the contractors to take suitable measures, or take such measures himself if the contractor does not adhere to the goals and guidelines established for the assignment.

The licensees must review and approve all the documentation, perform inspections and approve tests. The licensee is responsible that regulatory requirements and guides are followed.

Control of non-conformances

Identified non-conformances shall be assessed and corrective and preventive actions shall be taken. This shall be governed by the licensee’s management system, which shall include procedures to process non-conformances identified in processes and products. The management system shall include criteria for determine significance of non-conformances so that, if necessary, modifications of the plant or components are made or procedures are improved.

C. Regulatory review of design documentation (RB/IO, safety class)

For major plant modifications, such as thermal power up-rates the license has to prepare a Preliminary Safety Analysis Report (PSAR), which must be approved by SSM before plant modifications may be implemented. This PSAR should be based on the plant's existing Safety Analysis Report (SAR) and provided with

- details of the plant designed and lay out after planned modifications
- details of the planned mode of operation including operating limits
- transient and accident analysis and structural analysis that has been made of planned new or modified parts or functions of the plant and of such parts of the plant that has not changed but are affected by changes
- references to the verifying analysis

Any other plant modifications shall be based on up-to-date design specifications. Before the design specifications can be applied, the design basis on which they are based shall be notified to and reviewed by SSM. The design basis should also contain information on loads and load combinations during normal operating conditions, expected events, minor incidents and design basis accidents.

IOs review component design specifications after SSM has reviewed the design basis. This applies to all safety classes (SC1-SC3) and some non-safety classified components.

Design specifications should, to the appropriate extent, contain information concerning the function of the component, boundaries to other components and loads at these boundaries, requirements on pressure relief, internal and external environments, accessibility and testability as well as, where applicable, any neutron radiation that the components might be exposed to. Furthermore, the following should be included: information concerning safety and quality classification, materials requirements, lists of standards and other documents determining design, lists of valves and seals which during operation shall be locked in an open or closed position as well as referrals to documents that describe criteria for operational readiness. The design specifications for plant modifications should also contain analyses of how the modifications affect loads on and operating limits for existing components in the specific system and components in connecting systems. Detailed component level design documents shall also include

- structural and other analysis
- structural and isometric drawings
- material specifications
- welding and fabrication/manufacturing processes and their qualification
- inspection plans and procedures for destructive and non-destructive testing.

This inspection and review scheme including IOs does not yet apply to concrete structures.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning)

Safety class 1:	IOs supervises manufacturing and installation inspections, including welding qualifications, and commissioning inspections for components. SSM inspect commissioning tests after major plant modifications SC1 is not applicable for steel and concrete structures.
Safety class 2:	IOs supervises manufacturing and installation inspections including welding qualifications, and commissioning inspections for components. SSM inspect commissioning tests after major plant modifications SSM makes regulatory inspections on sample basis for concrete structures.
Safety class 3:	IOs supervises manufacturing and installation inspections including welding qualifications, and commissioning inspections for components. SSM makes regulatory inspections on sample basis for concrete structures.

E. Authorization of IOs

a. Accreditation

IOs as well as testing organizations (laboratories) for destructive and non-destructive testing have to be accredited for their tasks by SWEDAC. Accreditation decisions are based on inspections (audits) and reviews of the organizations management systems (including working procedures, education and professional training programs), technical competence, personnel resources and work practice for verification that SWEDAC's general accreditation rules and relevant regulations issued by SSM are met. SWEDAC do their inspections and audits before accreditation decisions in consultation with SSM.

SWEDAC's general accreditation rules are based on harmonized European and international standards such as EN ISO/IEC 17020 and EN ISO/IEC 17025. SSM states technical and personnel competence, which shall be reflected in the accredited organizations management systems and in their internal competence system to perform different tasks. References in the approval decision are applicable EN ISO/IEC standards and SSM regulations. The accredited IOs and testing organizations (laboratories) are listed on the SWEDACs web-pages where licensees can choose the suitable IO and testing organization for their use.

In the manufacture of mechanical components in another country may other certification and inspection bodies and laboratories carry out certain certification, inspection and testing tasks if they have been accredited under provisions equivalent to those for Swedish accredited organizations. These tasks are related to welding qualifications and testing of materials, welding and components.

b. Contracting, organizing daily inspection requests

The licensee or its vendor makes a contract with the IO and order the IO for the tasks.

c. Reporting to RB, oversight of IOs by RB

IOs communicate with and reports directly to the licensee and in some situations with the vendor/supplier Results from their work shall however be available to SWEDAC and SSM on request. Under the terms of accreditation, IOs shall report general problems and serious deviations also to the SSM. They must also be involved in experience feedback meetings SWEDAC or SSM organize.

SWEDAC and SSM staff perform annual oversight of IOs that have been accredited for inspection and review tasks in NPP under SSM's rules. This includes controls on sample basis of reviews performed by the IO in question as well as witnessing of how they perform inspections.

d. Conformance assessment

Conformity assessments are conducted by IOs stepwise during the process from design over manufacturing and installation to commissioning as have been described above. During these stepwise assessments, with clear hold points, IOs can handle and decide on acceptance non-conformances within those limits that are given in their working procedures, and which have been approved in the accreditation process. When non-conformances outside the limits is observed a certificate of conformity with SSM regulation cannot be issued, and the components can consequently not be taken into operation. In such situation a licensee have to take measures or justify and apply at SSM for an exception from the relevant requirement.

F. Use of management system audits/focused audits

In addition to the described inspection and review scheme SSM conducts management system audits and focused audits of the licensee’s activities on a sample basis.

SSM has no mandate under the current Swedish legislation to audit license’s manufacturers and suppliers. SSM's focus is presently instead on supervision of the licensee’s procedures, competence and resources for their own manufacturers and suppliers assessments and that they carry out audits with good quality.

However, discussions are now underway in Sweden to change the legislation so that the SSM as part of the oversight of licensees' supplier assessments also will be able to monitor the work of manufacturers and suppliers to verify that licensees have made correct audits.

National summary reports

WENRA, Inspection Working Group

SWITZERLAND

NATIONAL SUMMARY

A. Basic regulatory approach in the country (-includes hold-point strategy)

In Switzerland oversight responsibilities and scope are regulated in national laws, ordinances and regulatory guides of the Swiss Federal Nuclear Safety Inspectorate (ENSI). Where nuclear safety and security are concerned, the ENSI is appointed as supervisory authority in accordance with the Federal Act of 22 June 2007 on the Swiss Federal Nuclear Safety Inspectorate. Nuclear safety includes also fire protection and radiation protection in nuclear installations.

Inspection organizations are contracted by ENSI to review the detailed design and manufacturing documentation and to conduct the inspections during manufacturing and concreting on-site and in the workshops. The inspection organization for nuclear pressure equipment has an accreditation as an inspection body Type A in accordance with ISO/IEC 17020.

<u>Hold-point strategy</u>

Modifications and new installations of components and structures important to safety (classified equipment) need permits that are granted in 4 steps (hold points) by ENSI. The areas of Reactor Engineering, Civil Engineering, Systems Engineering, Mechanical Engineering, Electrical and I&C Engineering, Radiation Protection, Security, Organization and Personal are covered depending on the type of project. In principal the 4 steps are: (1) approval of the concept, (2) approval of the design, (3) approval of the implementation, (4) approval of the final documentation.

B. Expectations on licensees (-includes control of non-conformances)

The licence holder is responsible for the nuclear safety. Established or proven high-quality processes, materials, technologies and organisational structures and processes must be used in connection with design, construction, commissioning and operation of nuclear installations. This applies especially to the areas of planning, manufacture, testing, operation, surveillance, maintenance, quality assurance, evaluation of operational experience feedback, ergonomic design as well as basic and advanced training and professional development.

The licensee must have a quality management system that fulfils the requirements mentioned above. Depending on the nature of non-conformances deviations concerning classified equipment have to be registered, evaluated, reported or submitted for approval to ENSI or to the IO, which was contracted to supervise the work. Deviations with respect to the concept or the design left in place have to be approved by ENSI.

C. Regulatory review of design documentation (RB/IO, safety class)

The permits based on the review of design documentation for mechanical components of all safety classes (SK 1 to 4) and for all buildings are issued by ENSI. The review of design documentation is conducted by ENSI, which contracts IOs or engineering service companies for support as necessary. For non-nuclear safety mechanical equipment it is expected that industrial rules and standards are applied by the licensee.

The four review steps in mechanical engineering are: (M1) Design basis, (M2) Design specification, (M3) documentation of manufacturer for construction, manufacture and pre-service testing, (M4) start-up and final documentation. Review of M3 and M4 documentation is delegated to the IO in the case of classified pressure equipment as defined in ENSI-guideline G11. For minor modifications a simplified review process in one step may be used.

The four review steps in civil engineering are: (B1) Layout concepts, (B2) Building layout specifications, (B3) Building arrangement and installation, (B4) Building documentation. The review work is supported by engineering companies (IOs) that are contracted by ENSI.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacturing/after manufacturing/installation/commissioning/clarification of Tables)

Regulatory inspections during manufacturing, installation and final testing of nuclear pressure equipment of safety class SK1 and 2 are conducted by the IO as defined in ENSI-guideline G11 or by ENSI. For safety class SK3 and case by case SK4 the documentation of the utility's inspection is reviewed by the IO. Pressure tests and repairs during manufacturing for all classified pressure equipment (SK 1 to 4) are inspected by the IO.

Regulatory inspections on classified steel structures and on all buildings are conducted by ENSI or by engineering companies (IOs) that are contracted by ENSI.

E. Authorisation of IOs (if used) (-accreditation, -contracting, organising daily inspection requests, -reporting to RB, oversight of IOs by RB, -conformance assessment)

The authorization of IOs or engineering companies that act as IOs is given by a contract with ENSI. The ENSI-guideline G11 and the contract for the supervision of nuclear pressure equipment requires an accreditation in accordance with ISO/IEC 17020 Type A. ENSI takes part in the accreditation process. The contracts and case by case decisions regulate the details of organizing the reviews and inspections and of the reporting to ENSI. For practical reasons the daily inspection work is organized in direct contact between the licence holder and the IO. ENSI receives copies of all documentation and conformance assessments issued by the IO.

F. Use of management system audits/focused audits (-which organisations are audited by the RB? -is this auditing based on sampling? -when do you use management system audits and when focused audits?

ENSI takes part in the audits concerning the accreditation of the contracted IO for nuclear pressure equipment. ENSI inspects the processes of the quality management system of the licence holders based on sampling.

Appendix 4

National summary reports

WENRA, Inspection Working Group

UK

NATIONAL SUMMARY

A. Basic Regulatory Approach in the Country

The UK nuclear safety regulatory authority is the Health and Safety Executive (HSE). Within HSE the Office for Nuclear Regulation (ONR) deals with nuclear facilities. ONR includes the nuclear safety regulator formerly known as the Nuclear Installations Inspectorate. ONR includes nuclear safety, security and transport regulation, it also works closely with the UK Environment Agencies.

Within the UK regulatory system, the operators of nuclear plant are licensed by HSE/ONR. No other organisations are licensed or approved in the framework of nuclear safety regulation.

The basic regulatory approach in the UK is outcome focussed, goal setting regulation. The licensee is responsible for safety.

<u>Hold Point Strategy</u>

The nuclear site licensees have processes which require hold points throughout the design, construction and commissioning cycle. These hold points will be developed in a clear programme along with schedules and quality documents which ensure that all relevant checks and controls have been met before a hold point can be released. The process for hold point clearance will have a high level of control and supervision by the licensee. Internal governance and oversight arrangements are also expected to ensure that an internal independent challenge function confirms that the work necessary to clear a hold point has been completed to an adequate standard.

In addition using the nuclear site licence it is common for ONR to place hold points or to select a subset of the licensees hold points. Generally ONR will have a small number of hold points placed at the start of the project, with options to increase the number of hold points if the performance of the licensee requires this.

B. Expectation on licensees (includes control of non-conformances)

The licensee is responsible for safety.

The nuclear site license has 36 standard conditions which encompass design, construction, commissioning, modification, operation, maintenance and decommissioning activities. One condition of the licence is that the licensee makes adequate arrangements to ensure that they procure equipment and services that are of sufficient quality. The responsibility for defining the standards and expectations for control of design, manufacture, commissioning, etc lies with the nuclear site licensee. ONR inspections consider the licence compliance arrangements.

<u>Control of non-conformances</u>

In accordance with arrangements made under the nuclear site licence the licensee will have robust controls for the control of modifications, change control and non-conformances which arise during the design, manufacture, construction, and commissioning phases. These arrangements will ensure that the safety significance of deviations is considered with increasingly robust controls in place for higher category components or deviations. These arrangements will be controlled by the licensee and will be expected to be mirrored in main suppliers and contractors. The arrangements will be inspected and reviewed by the licensees internal oversight function.

As part of ONRs' inspection processes we would generally request oversight of significant non-conformances. The licence compliance arrangements include reporting of deviations from expected condition – for major deviations, these are reported directly to ONR.

C. Regulatory review of Design Documentation (RB/IO, safety class)

ONR fulfils its regulatory duties through inspection of licence compliance, and inspection and assessment of the adequacy of safety cases throughout all stages of life of nuclear facilities.

As there are expected to be new NPPs constructed in the UK, ONR has set up a division with the express purpose of considering the generic design of potential new plant. Certain aspects of the design are being examined in some detail, with ONR advising whether the proposals are likely to result in a design or operational plant that is licensable. The oversight process is known as Generic Design Assessment (GDA), and involves ONR and the Environment Agency.

At the end of the GDA process, the Regulators will decide if the proposed designs are acceptable for build in the UK. The GDA process is based on sampling of the design.

The GDA process is separate from nuclear site licensing. Following completion of GDA, issues requiring further regulatory assessment and resolution within the licensing and permissioning regime may include:

- site-specific aspects not covered by the generic site envelope;
- other site-specific aspects;
- any other changes to the design or safety documentation since GDA;
- assessment of the licence applicant's organisation;
- consideration of any exclusions in ONR's statement of Design Acceptance

In general there is no independent design verification required, but in some cases ONR would look for independent design verification, specifically when it is a code requirement. However, for those components that form a principal means of ensuring nuclear safety, such as the Reactor Pressure vessel, other principal NSSS components, or the concrete containment structures, the licensee is expected to contract an Independent Third party (normally the ITPIA) to undertake independent design review, verification and certification. ONR may also commission independent design verification.

D. Regulatory inspections of structures and components (RB/IO, safety class, during manufacture/post manufacture/installation/commissioning)

In general, the principle of goal setting regulation expects the licensee to have full control and supervision of the supply chain and all activities which contribute to nuclear safety. The licensees' arrangements should ensure that adequate oversight is in place to procure components qualified to standards defined in specifications and presented in the Pre Construction Safety Report (PCSR). ONR will inspect aspects of these arrangements on a sampling basis taking account of safety significance, degree of complexity, novelty, feedback from UK or international experience, etc.

National summary reports

For those components that form a principal means of ensuring nuclear safety ONR would normally sample inspect quality management systems and arrangements of the licensee and the supply chain. In addition, for critical phases of manufacture and commissioning, ONR may choose to deploy specialist inspectors to inspect and witness specific activities, such as forging, welding, heat treatment, non-destructive testing, the leak testing of the Prestressed Concrete Containment vessel, etc.

ND may choose to inspect the licensee's arrangements for control of quality of materials supplied for example in the manufacture of concrete, or quality of steel provided to site. The licensees will create arrangements for construction inspection and ONR inspections will sample arrangements to confirm they are adequate and are being complied with.

E. Authorisation of IOs

Third party organisations are used to support both the regulator and licensee's activities to secure licence compliance.

ONR does not license or authorise bodies other than operators of plant. Some third parties are accredited by the UK Accreditation Service (UKAS) e.g. individuals performing auditing or surveying functions. UKAS also provides an accreditation service to organisations such as laboratories or testing companies that perform tests in accordance with standard procedures. Other organisations may be accredited through different organisations, for example, Lloyd's Register Quality Assurance may provide accreditation against the quality assurance management system ISO 9001.

Third Party Organisations are selected by ONR via their procurement process which includes assessment of their capability in the particular topic that they are to service. The organisation must have suitably qualified and experienced personnel to provide the appropriate standard of support. Third Party Organisations are expected to have suitable Quality Management systems.

The regulator, licensee (or prospective licensee, purchaser) would normally place contracts with inspection organisations. Considerable care is invested on the part of ONR and the Third Party organisations to avoid conflict of interests, which may occur where the Third Party Organisation could be seen to be providing specialist assistance to ONR while being engaged by the licensee on a related activity.

F. Use of Management System Audits/Focused Audits

ONR engages with prospective nuclear site licensees as they develop management arrangements in the build up to nuclear site licence application. Further inspection of leadership and management for safety, including audit and inspection is undertaken once a site licence application is received and before a licence is granted. Management system and quality audits continue once an organisation receives a nuclear site licence.

Inspection of quality management arrangements of licensees and of the supply chain are undertaken by ONR, with supply chain auditing a key element of the licensees' arrangements for control and supervision of the supply chain.

International standards that define general requirements for conformity assessment bodies in the accreditation process

- EN ISO/IEC 17011, Conformity assessment. General requirements for accreditation bodies accrediting conformity assessment bodies
- EN ISO/IEC 17020, General criteria for the operation of various types of bodies performing inspection
- EN ISO/IEC 17021, Conformity assessment. Requirements for bodies providing audit and certification of management systems
- EN ISO/IEC 17024, Conformity assessment. General requirements for bodies operating certification of persons
- EN ISO/IEC 17025, General requirements for the competence of testing and calibration laboratories
- EN 45011, General requirements for bodies operating product certification systems

Western European
WENRA
Nuclear Regulators Association

Report

WENRA WGWD

Western European Nuclear Regulators Association

Report

Waste and Spent Fuel Storage Safety Reference Levels

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Report of Working Group on Waste and Decommissioning (WGWD)

Version 2.2, April 2014

WENRA WGWD

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Executive Summary

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The Western European Nuclear Regulators' Association (WENRA) is an international body made up of the Heads and senior staff members of nuclear regulatory authorities of European countries with nuclear power plants. The main objectives of WENRA is to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues.

To accomplish these tasks two working groups within the WENRA have been established - Reactor Harmonization Working Group (RHWG) and Working Group on Waste and Decommissioning (WGWD).

This document contains the results of the work of WGWD in the area of the safety for spent fuel and radioactive waste storage facilities. The objective of this report is to provide safety reference levels for these facilities, which are based on corresponding IAEA documents (requirements, guidances, etc). Although the IAEA safety standards establish an essential basis for safety of all nuclear installations covering also the spent fuel and radioactive waste stores, the WENRA safety reference levels incorporate more facility specific requirements.

The document was prepared by reviewing the Storage Report Version 1.0 by the working group based on support by the German task manager, Mr. Bernhard Gmal, and, for the final version, Mr. Sven Keßen. WGWD members during the review period are listed below:

Belgium	Joris CREEMERS
	Olivier SMIDTS
Bulgaria	Magda PERIKLIEVA
Czech Republic	Peter LIETAVA (former chairman of WGWD)
Finland	Jarkko KYLLÖNEN
France	Loïc TANGUY
Germany	Sven KESSEN
	Manuela M. RICHARTZ
Hungary	Gábor NAGY
Italy	Mario DIONISI
Lithuania	Algirdas VINSKAS
Netherlands	Thierry LOUIS
Romania	Daniela DOGARU
Slovakia	Alena ZAVAZANOVA
Slovenia	Polona TAVCAR
Spain	Gregorio OROZCO
	Jose Luis REVILLA
Sweden	Nicklas CARLVIK
Switzerland	Stefan THEIS (Chairman of WGWD)
United Kingdom	Joyce RUTHERFORD

WENRA Policy Statement

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We, the heads of the national nuclear safety authorities, members of WENRA, commit ourselves to a continuous improvement of nuclear safety in our respective countries.

Nuclear safety and radiation protection are based on the principle of the prime responsibility of the operators. Our role is to ensure that this responsibility is fully secured, in compliance with the regulatory requirements.

In order to work together, we created the Western European Nuclear Regulators' Association (WENRA) with the following main objectives to:

- build and maintain a network of chief nuclear safety regulators in Europe;
- promote exchange of experience and learning from each other´s best practices;
- develop a harmonized approach to selected nuclear safety and radiation protection issues and their regulation, in particular within the European Union;
- provide the European Union Institutions with an independent capability to examine nuclear safety and its regulation in applicant countries.

In order to develop a harmonized approach, we are making efforts to:

- share our experience feedback and our vision;
- exchange personnel, allowing an in-depth knowledge of working methods of each other;
- develop common safety reference levels in the fields of reactor safety, decommissioning safety, radioactive waste and spent fuel management facilities in order to benchmark our national practices.

We recognise the IAEA standards to form a good base for developing national regulations. The developed reference levels represent good practices in our counties and we are committed

- by the year of 2010 to adapt at a minimum our national legislation and implementation to the reference levels;
- to influence the revision of the IAEA standards when appropriate;
- to continuously revise the reference levels when new knowledge and experience are available.

We strive for openness and improvement of our work. For that purpose we are making efforts to

- keep the European nuclear safety and radiation protection bodies not belonging to WENRA and the EU Institutions informed of the progress made in our work;
- make the WENRA reports available on the Internet (www.wenra.org);
- invite stakeholders to make comments and suggestions on our reports and the proposed reference levels.

Signed in Stockholm December 2005

J-P. Samain, Belgium

S. Tzotchev, Bulgaria

D. Drabova, Czech Republic

J. Laaksonen, Finland

A-C. Lacoste, France

W. Renneberg, Germany

I. Lux, Hungary

S. Giulianelli, Italy

S. Kutas, Lithuania

P. Müskens, the Netherlands

V. Zsombori, Romania

M. Ziakova, Slovakia

A. Stritar, Slovenia

M. Teresa Estevan, Spain

J. Melin, Sweden

U. Schmocker, Switzerland

M. Weightman, United Kingdom

Glossary

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Acceptance criteria for storage
See: waste or spent fuel acceptance criteria

Ageing
General process in which characteristics of a structure, system or component gradually change with time or use.

> **Ageing degradation**
> Ageing effects that could impair the ability of a structure, system or component to function within its design limits.

> **Ageing management**
> Engineering, operations and maintenance actions to control within acceptable limits the ageing degradation of structures, systems and components.

Burnup credit
Credit in the safety assessment of a structure, component, system or facility that is given for the reduction in spent fuel nuclear reactivity as a result of fission

Conditioning
Those operations that produce a waste or spent fuel package suitable for handling, transport, storage and/or disposal. Conditioning may include the conversion of the waste to a solid waste form, enclosure of the waste in containers and, if necessary, providing an overpack.

Design basis accident
Accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

Discharge, authorized
Planned and controlled release of (usually gaseous or liquid) radioactive material into the environment in accordance with an authorization.

Licensee
The licensee is the legal person or organization having overall responsibility for a facility or activity

Remark: WGWD recognizes that this organization may change as the facility passes to the decommissioning phase according to national strategies.

Management system
A set of interrelated or interacting elements (system) for establishing policies and objectives and enabling the objectives to be achieved in an efficient and effective manner. The management system integrates all elements of an organization into one coherent system to enable all of the organization's objectives to be achieved. These elements include the organizational structure, resources and processes. Personnel, equipment and organizational culture as well as the documented policies and processes are parts of the management system. The organization's processes have to address the totality of the requirements on the organization as established in, for example, IAEA safety standards and other international codes and standards.

The term management system reflects and includes the evolution in the approach from the initial concept of 'quality control' (controlling the quality of products) through 'quality assurance' (the system to ensure the quality of products) to 'quality management' (the system to manage quality).

Monitoring
1. The measurement of dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of the results,
2. Continuous or periodic measurement of radiological or other parameters or determination of the status of a structure, system or component. Sampling may be involved as a preliminary step to measurement.

Nuclear facility
A facility and its associated land, buildings and equipment in which radioactive materials are produced, processed, used, handled, stored or disposed of on such a scale that consideration of safety is required.

Nuclear safety
See: Protection and Safety

Operation
All activities performed to achieve the purpose for which an authorized facility was constructed.

Operational limits and conditions
A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.

Owner
Owner means a body having legal title to waste or spent fuel including financial liabilities (it is usually the waste and spent fuel producer).

Passive Safety Feature
A safety feature which does not depend on an external input and/or continuous supply of media.

Protection and Safety
The protection of people against exposure to ionizing radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur.

Safety is primarily concerned with maintaining control over sources, whereas radiation protection is primarily concerned with controlling exposure to radiation and its effects. Clearly the two are closely connected: radiation protection is very much simpler if the source in question is under control, so safety necessarily contributes towards protection. Sources come in many different types, and hence safety may be termed nuclear safety, radiation safety, radioactive waste safety or transport safety, but protection (in this sense) is primarily concerned with protecting humans against exposure, whatever the source, and so is always radiation protection.

> **Radiation protection:**
> The protection of people from the effects of exposure to ionizing radiation, and the means for achieving this.
>
> **Nuclear safety:**
> The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

Quality management system
See: management system

Radiation protection
See: protection and safety

Regulatory body
An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

Safety analysis
Evaluation of the potential hazards associated with the conduct of an activity.

Safety assessment
1. Assessment of all aspects of facility practice which are relevant to protection and safety; for a nuclear facility this includes the site, the design and the operation of the facility.

2. The systematic process that is carried out throughout the design process to ensure that all the relevant safety requirements are met by the proposed (or actual) design. Safety assessment includes, but is not limited to, the formal safety analysis.

Safety case
A collection of arguments and evidence in support of the safety of a facility or activity. This will normally include the findings of a safety assessment.

Safety policy
A documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as visible part of an integrated organization policy.

Spent fuel

1. Nuclear fuel removed from a reactor following irradiation, that is no longer usable in its present form.[1]
2. Nuclear fuel that has been irradiated in and permanently removed from a reactor core.

Storage
The holding of spent fuel or of radioactive waste in a facility that provides for their/its containment, with the intention of retrieval.

Structures, systems and components (SSCs)
A general term encompassing all of the elements (items) of a facility or activity which contribute to protection and safety, except human factors.

- **Structures** are the passive elements: buildings, vessels, shielding, etc.
- A **system** comprises several **components**, assembled in such a way as to perform a specific (active) function.
- A **component** is a discrete element of a system.

Waste treatment
Operations intended to benefit safety and/or economy by changing the characteristics of the waste. Three basic treatment objectives are:

- volume reduction,
- removal of radionuclides from the waste, and
- change of composition.

[1] The adjective 'spent' suggests that *spent fuel* cannot be used as fuel in its present form (as, for example, in *spent source*). In practice, however (as in (2) above), *spent fuel* is commonly used to refer to fuel which has been used as fuel but will no longer be used, whether or not it could be (which might more accurately be termed 'disused fuel').

Treatment may result in an appropriate waste form.

Waste
Material for which no further use is foreseen.

Waste, radioactive
For legal and regulatory purposes, waste that contains or is contaminated with radionuclides at concentrations or activities greater than clearance levels as established by the regulatory body.

Waste or spent fuel acceptance criteria
Quantitative or qualitative criteria specified by the regulatory body, or specified by an operator and approved by the regulatory body, for radioactive waste or spent fuel to be accepted by the operator of a storage facility. Waste acceptance criteria might include, for example, restrictions on the activity concentration or the total activity of particular radionuclides (or types of radionuclides) in the waste or the spent fuel or criteria concerning the waste form or the packaging of the waste or the spent fuel.

Waste characterization
Determination of the physical, chemical and radiological properties of the waste to establish the need for further adjustment, treatment or conditioning, or its suitability for further handling, processing, storage or disposal.

Waste or spent fuel package
The product of conditioning that includes the waste or spent fuel form and any container(s) and internal barriers (e. g. absorbing materials and liner), as prepared in accordance with requirements for handling, transport, storage and/or disposal.

List of Abbreviations

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AMP	ageing management program
EIA	environmental impact assessment
EU	European Union
IAEA	International Atomic Energy Agency
NEA	Nuclear Energy Agency (OECD)
NPP	nuclear power plant
OEF	operational experience feedback
OLC	operational limits and conditions
PIE	postulated initiating event
PSR	periodic safety review
QM	quality management
R&D	research and development
RHWG	Reactor Harmonization Working Group
SSCs	structures, systems and components
SRL	safety reference level
V.1	Version 1 of the SRLs
V.2	Version 2 of the SRLs
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators' Association
WGWD	Working Group on Waste and Decommissioning

Part 1
Introduction and Methodology

-

1.1 Introduction

-

This report is the result of an effort by the Working Group on Waste and Decommissioning (WGWD) of WENRA, from 2002 to 2009. It presents the safety reference levels (SRLs) for radioactive waste and spent fuel management facilities and practices that are thought to be a good basis for future harmonization on a European level.

The SRLs cannot be considered as independent European safety requirements because current legislation in WENRA member states would not allow that due to fundamental differences reflecting the historical development in European countries. The SRLs are a set of requirements against which the situation of each country is assessed and it is each country's responsibility to implement actions to ensure that these levels are reached.

1.1.1 Background

WENRA, which has been established in February 1999, is the association of the Heads of nuclear regulatory authorities of European countries with at least one nuclear power plant in construction, operation or decommissioning phase. WENRA has been formally extended in 2003 to include future new European Union (EU) member states. Currently following countries are members of WENRA: Belgium, Bulgaria, the Czech Republic, Finland, France, Germany, Hungary, Italy, Lithuania, the Netherlands, Romania, Slovenia, Slovakia, Spain, Sweden, Switzerland and the United Kingdom. Recently various other states have been appointed to WENRA meetings with the status of "observers". However such states have not yet been participating in the work of WGWD and have not taken part in the preparation of this report.

The original objectives of the Association were:

- to provide the EU institutions with an independent capability to examine nuclear safety and its regulation in applicant countries,
- to provide the EU with an independent capability to examine nuclear safety and regulation in candidate countries,
- to evaluate and achieve a common approach to nuclear safety and regulatory issues which arise.

The second objective of WENRA has been fulfilled by the preparation of a report on nuclear safety in candidate countries having at least one nuclear power plant. After May 1st, 2004, when most of these candidate countries became regular EU member states, the new WENRA tasks, based on first and third original Association's objectives, became:

- provide the European Union institutions with an independent capability to examine nuclear safety and its regulation in applicant countries and
- to develop common approaches to nuclear safety and regulations and to encourage the harmonization of practices.

To perform these tasks two working groups within the WENRA have been established - Reactor Harmonization Working Group (RHWG) and Working Group on Waste and Decommissioning (WGWD). The work of WGWD has started in 2002.

1.1.2 Objective

The objective of this report is to provide SRLs for spent fuel and radioactive waste storage facilities. The design storage period involved will typically be several decades, depending on the national waste and spent fuel management strategy.

Although the SRLs in this report are oriented toward the licensees of the above-mentioned facilities, who are usually responsible for the safety of the facilities throughout their lifetime, they can also be used by the regulatory body for the review and evaluation of storage facilities' safety.

According to the WENRA policy statement the harmonization process of the national legal systems in member states should be finished by the year 2010. In 2009 WENRA decided to prolong the deadline in case of the storage SRLs until end of 2012.

1.1.3 Scope

The SRLs are primarily focussed on separate, purpose built or adapted storage facilities used to store spent fuel or radioactive waste in solid form. As this document is intended to cover a wide range of storage facilities, the reference levels will need to be implemented in different ways to be appropriate for the particular facility. The SRLs were also primarily developed for licensed nuclear facilities for storage, but can be used also for other facilities accommodating radioactive waste from industry, hospitals, research centres etc.

Under certain circumstances (steam generator exchange, decommissioning) large, bulky waste items are subject to storage. The SRLs of this document shall be applied as appropriate to such material as well.

These SRLs may also be applied to stores as integrated parts of other facilities, e.g. NPPs, facilities for waste conditioning or for disposal. In such cases it should be recognized that many of the SRLs of a general nature, e.g. on quality management and facility operation, may have to be applied together with SRLs developed for the other parts of the facility. A similar situation occurs if the storage facility is operated in combination with other facilities, or incorporates other nuclear activities than storage.

Spent fuel stores built for the operation of the reactors are not covered by this report. Because of the national policies on spent fuel, operators can consider the need to extend the use of the stores or adapting the existing ones, beyond the operational period of the reactor. Those facilities shall be covered by this report.

Because WGWD members do not all regulate the following matters, WGWD has concentrated on relevant nuclear and waste safety requirements and, in particular, it has not taken into account in detail other regulatory requirements such as Environmental Impact Assessment regulation (required by EU directives), discharge authorization, waste disposal, conventional occupational health and safety, physical protection including safeguards, and funding issues. In some countries, these matters are addressed by other national regulatory organizations.

1.1.4 Structure

The report consists of three main parts.

Following this introduction, Section 1.2 presents the general methodology that was followed to develop the SRLs and to analyse their application in participating countries.

Part 2 of the report presents the actual waste and spent fuel storage reference levels.

Part 3 of the report describes the results of the benchmarking process and the National Action Plans (NAP)

1.2 Methodology

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The working methodology of WGWD has gone through several steps and changes since 2002, when the working group was established. A list of topics to be covered by WGWD was defined taking into account the common field of responsibility of WENRA members. Generally for the development of storage SRLs relevant IAEA documents were consulted, the latest list of which is as follows:

- Fundamental Safety Principles, Safety Fundamentals SF-1, Vienna (2006)
- Storage of Spent Fuel, DS 371, Vienna (January 2010)
- Predisposal Management of Radioactive Waste, GSR Part 5, Vienna (2009),
- Periodic safety review of nuclear power plants, NS-G-2.10, Vienna (2003),
- A System for Feedback of Experience from Events in Nuclear Installations, NS-G-2.11, Vienna (2006),
- Storage of Radioactive Waste, WS-G-6.1, Vienna (2006),
- Safety of Nuclear Fuel Cycle Facilities, NS-R-5, Vienna (2009),
- Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety, Safety Requirements,
  GS-R-1, Vienna (2000).
- Management Systems for Facilities and Activities, Safety Requirements, GS-R-3, Vienna (2006).
- The Management System for the Processing, Handling and Storage of Radioactive Waste, GS-G-3.3, Vienna (2008)
- The Management System for Nuclear Installations GS-G-3.5, Vienna (2009)
- Preparedness and Response for a Nuclear or Radiological Emergency, GS-R-2, Vienna, 2002
- Periodic Safety Review of Nuclear Power Plants, DS 426, Vienna (2009)
- Safety Case and Safety Assessment for Predisposal Management of Radioactive Waste, DS 284, Vienna (August 2008)

A first set of SRLs was posted on the website of the WENRA organization at the beginning of 2006 and presented to stakeholders in order to receive their comments before June 1st, 2006. Most of the comments recommended to address more specifically the issues raised by the storage of spent fuel and radioactive waste in order to prevent the specific hazards they pose. WGWD reflected a considerable number of comments and established in December 2006 Version 1 (herein referred as V.1) of the waste and spent fuel storage report on which basis the following benchmarking exercise of the storage-SRLs in WENRA member countries was conducted.

An evaluation of the implementation of the SRLs in the regulations (national legal systems) and in the facilities has been performed till mid-2009 in each WENRA member state. In a benchmarking exercise the justification and evidence for implementation of each SRL was discussed country by country and agreed within WGWD in subgroups. After this evaluation, all member states developed national action plans in order to address identified discrepancies and to update their national regulations till the end of 2012. Progress of the national action plans is under continuous review of the working group.

Reflecting the results of the national assessments, the set of SRLs was subject to further improvement, which together with updated references of IAEA documents, led to this most recent Version 2 (herein referred as V.2) of the "Waste and Spent Fuel Storage Safety Reference Levels". For accomplishing this, two review readings of the SRLs were carried out in the plenary sessions of the 21st and 22nd meeting of the WGWD. Before the 23rd meeting an update of the references and quotations of relevant IAEA documents had been performed. After the 23rd meeting of WGWD with a final reading WENRA approved the report in spring 2010 for official release as draft on the WENRA homepage. Stakeholders have been invited to respond with comments until June 30th 2010. In subsequent WGWD meetings

- all comments received were evaluated
- SRL-texts were modified accordingly where agreed and
- any such decisions were discussed with representatives of relevant stakeholder organisations in a special working group session.

Finally the resulting storage report was approved by WENRA in autumn 2010 and published as Version 2.1 in February 2011.

Part 2
Waste and Spent Fuel Storage Safety Reference Levels

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These reference levels are intended for separate, purpose built or adapted storage facilities which should incorporate passive safety features as far as reasonably practical and which will be used to store spent fuel or waste in solid form. The design base storage period involved will typically be several decades, depending on the national waste and spent fuel management strategy. In the future WGWD may consider other aspects of radioactive waste and spent fuel management.

Some reference levels apply to the owner of the waste or spent fuel (S-04, S-05, S-06, S-07, S-18, S-51).

WGWD is conscious that some of the reference levels, in particular those related to the design of facilities, may not be fulfilled by existing facilities. Implementation of these levels for existing facilities will have to be examined within the national regulatory framework.

The term “nuclear safety” covers in this document also the measures for radiation protection.

The reference levels apply to a wide range of facilities for the storage of spent fuel and radioactive waste, for which the hazard potential may vary significantly. On the one hand, the reference levels apply to fuel stores which may require active protection systems of high reliability. On the other hand, the reference levels apply to the storage of wastes where the design of both the waste package and the store are based on the concept of passive safety.

Consideration therefore needs to be given as to whether individual reference levels are relevant in specific circumstances, and when they are relevant they need to be applied in a proportionate manner, taking account the magnitude of the hazard.

2.1
Safety area: Safety management

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2.1.1 Safety issue: Responsibility

S-01:
The licensee of the radioactive waste or spent fuel storage facility is responsible for the safety of all activities in the facility, and for the implementation of programs and procedures necessary to ensure safety, including the waste or spent fuel stored. In accordance with the graded approach, the programs and procedures necessary to ensure safety shall be commensurate with the scale of the facility and the type of the inventory.

*Related **IAEA** safety standards:*

The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks. (SF-1; principle 1)

The person or organization responsible for any facility or activity that gives rise to radiation risks or for carrying out a programme of actions to reduce radiation exposure has the prime responsibility for safety.(SF-1; para 3.3)

The operator is responsible for the safety of all activities in the storage of radioactive waste and for the implementation of the programmes and procedures necessary to ensure safety. In accordance with the graded approach, the programmes and procedures necessary to ensure safety will generally be less extensive for the operator of a small facility. (WS-G-6.1, para 3.11).

S-02:
To fulfil its prime responsibility for safety during the lifetime of the facility, the licensee shall establish and implement safety policies and ensure that safety issues are given the highest priority.

*Related **IAEA** safety standards:*

To fulfil its prime responsibility for safety throughout the lifetime of a fuel cycle facility, the operating organization shall establish, implement, assess and continually improve a management system that integrates safety, health, environmental, security, quality and economic elements to ensure that safety is properly taken into account in all the activities of an organization. (NS-R-5, para 4.1)

The operating organization shall establish and implement safety, health and environmental policies in accordance with national and international standards and shall ensure that these matters are given the highest priority (NS-R-5, para 4.2)

S-03:

The licensee shall commit itself to maintain the safety of the facility and, as far as reasonably practicable, improve it on the basis of operating experience.

*Related **IAEA** safety standards:*

Operators shall be responsible for the safety of predisposal radioactive waste management facilities or activities.4 The operator shall carry out safety assessments and shall develop a safety case, and shall ensure that the necessary activities for siting, design, construction, commissioning, operation, shut-down and decommissioning are carried out in compliance with legal and regulatory requirements. (GSR Part 5, Requirement 4)

S-04:

There shall be clear and unequivocal ownership of the waste and spent fuel stored in the facility.

*Related **IAEA** safety standards:*

There should be clear and unequivocal ownership of the spent fuel stored in the facility. […] (DS 371; para 3.29)

[…] The legal framework should include provisions to ensure a clear allocation of responsibility for safety throughout the entire process of predisposal management, in particular with respect to storage, and including any transfer between operators. The continuity of responsibility for safety should be ensured by means of authorization by the regulatory body. […] (WS-G-6.1, para 3.2).

S-05:

The waste or spent fuel owner shall be responsible for the overall strategy for the management of its waste and spent fuel, taking into account interdependencies between all stages of waste and spent fuel management and options available, from generation to disposal. The strategy shall be consistent with the overall national radioactive waste and spent fuel management strategy.

*Related **IAEA** safety standards:*

Interdependences among all steps in the predisposal management of radioactive waste, as well as the impact of the anticipated disposal option, shall be appropriately taken into account. (GSR Part 5, Requirement 6)

Owing to the interdependences among the various steps in the predisposal management of radioactive waste, all activities from the generation of radioactive waste up to its disposal, including its processing, are to be seen as parts of a larger entity, and the management elements of each step have to be selected so as to be compatible with those of the other steps. This has to be achieved principally through governmental and regulatory requirements and approaches. It is particularly important to consider the established acceptance criteria for disposal of the waste or the criteria that are anticipated for the most probable disposal option. (GSR Part 5, para 3.21)

S-06:

The interface between responsibilities of the licensee of the storage facility and the waste or spent fuel owner shall be clearly defined, agreed and documented.

*Related **IAEA** safety standards:*

The interface between the responsibilities of the operator and the spent fuel owner, if they differ, should be clearly defined, agreed and documented. (DS 371; para 3.29)

S-07:

Information about changes of waste and spent fuel ownership, or about changes to the relationship between owner and licensee, shall be provided to the regulatory authority.

*Related **IAEA** safety standards:*

Information about changes of ownership of waste or about changes in the relationship between owner and licensee has to be provided to the regulatory body. (GSR Part 5, para 3.18)

2.1.2 Safety issue: Organizational structure

S-08:

The licensee shall establish an organizational structure to enable its safety policy to be delivered with a clear definition of responsibilities and accountabilities, lines of authority and communication.

*Related **IAEA** safety standards:*

Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks. (SF-1; principle 3)

The operating organization shall establish an organizational structure to enable these policies to be carried out with a clear definition of responsibilities and accountabilities, lines of authority and communication. (NS-R-5; para 4.2).

S-09:

The licensee shall maintain the capability in terms of staffing, skills, experience and knowledge to enable it to competently undertake the activities during the lifetime of the facility from siting to decommissioning. Where the resources and skills necessary to deliver any part of theses undertakings are provided by an external organization, the licensee shall nevertheless retain within its organization the capability to assess the adequacy of the external organizations' capabilities of ensuring safety.

*Related **IAEA** safety standards:*

The operating organization shall maintain the capability in terms of staffing, skills, experience and knowledge to undertake competently all activities during the lifetime of the facility from siting to decommissioning. Where the resources and skills necessary to deliver any part of these undertakings are *provided by an external organization, the operating organization shall nevertheless retain within its organization the capability to assess the adequacy of the external organizations' capabilities for ensuring safety. (NS-R-5; para 4.9).*

S-10:

The licensee shall specify the necessary qualifications and experiences for all staff involved in activities that may affect safety and establish training programs for developing and maintaining the professional skills of the staff.

*Related **IAEA** safety standards:*

The operating organization shall specify the necessary qualifications and experience for all staff involved in activities that may affect safety. It shall also specify appropriate requirements on training and its assessment and approval. (NS-R-5; para 4.10).

2.1.3 Safety issue: Management system

S-11:

A management system shall be established, implemented, assessed and continually improved. It shall be aligned with the goals of the organization and shall contribute to their achievement. The main aim of the management system shall be to achieve and enhance safety by:

- bringing together in a coherent manner all the requirements for managing the organization
- describing the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied
- ensuring that health, environmental, security, quality and economic requirements are not considered separately from safety requirements, to help preclude their possible negative impact on safety.

*Related **IAEA** safety standards:*

A management system shall be established, implemented, assessed and continually improved. It shall be aligned with the goals of the organization and shall contribute to their achievement. The main aim of the management system shall be to achieve and enhance safety by:

- *Bringing together in a coherent manner all the requirements for managing the organization;*
- *Describing the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied;*
- *Ensuring that health, environmental, security, quality and economic requirements are not considered separately from safety requirements, to help preclude their possible negative impact on safety. (GS-R-3; para 2.1, also cited in GS-G-3.3, para 2.1)*

Leadership in safety matters has to be demonstrated at the highest levels in an organization. Safety has to be achieved and maintained by means of an effective management system. This system has to integrate all elements of management so that requirements for safety are established and applied coher-

ently with other requirements, including those for human performance, quality and security, and so that safety is not compromised by other requirements or demands. The management system also has to ensure the promotion of a safety culture, the regular assessment of safety performance and the application of lessons learned from experience. (SF-1, principle 3, para 3.12)

S-12:

The management system shall cover the full lifetime of a facility and the entire duration of activities in normal, transient and emergency situations. For a storage facility, these phases usually include planning, siting, design, construction, commissioning, operation and decommissioning.

*Related **IAEA** safety standards:*

This Safety Requirements publication is applicable throughout the lifetime of facilities and for the entire duration of activities in normal, transient and emergency situations. …. For a facility, these phases usually include siting, design, construction, commissioning, operation and decommissioning (or close-out or closure). (GS-R-3; para 1.11)

S-13:

The processes of the management system that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the organization shall be identified, and their development shall be planned, implemented, assessed and continually improved. The work performed in each process shall be carried out under controlled conditions, by using approved current procedures, instructions, drawings or other appropriate means that are periodically reviewed to ensure their adequacy and effectiveness.

*Related **IAEA** safety standards:*

The processes of the management system that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the organization shall be identified, and their development shall be planned, implemented, assessed and continually improved. (GS-R-3; para. 5.1)

The work performed in each process shall be carried out under controlled conditions, by using approved current procedures, instructions, drawings or other appropriate means that are periodically reviewed to ensure their adequacy and effectiveness. (GS-R-3; para. 5.9)

S-14:

The documentation of the management system shall include the following:

- the policy statements of the licensee;
- a description of the management system;
- a description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
- a description of the interactions with relevant external organizations;
- a description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved.

*Related **IAEA** safety standards:*

The documentation of the management system shall include the following:

- *The policy statements of the organization;*
- *A description of the management system;*
- *A description of the structure of the organization;*
- *A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;*
- *A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved. (GS-R-3; para. 2.8)*

2.1.4 Safety issue: Record keeping

S-15:

The licensee shall develop and maintain a record system on the location and characteristics of every waste and spent fuel package or unpackaged spent fuel element in storage, including information on its ownership and origin.

*Related **IAEA** safety standards:*

The operating organization should develop and maintain a records system on spent fuel data and on the storage system, which includes the radioactive inventory, location and characteristics of the spent fuel, information on ownership, origin and information about its characterization. […] (DS 371, para 3.27)

For the storage of radioactive waste, a variety of records should be compiled, managed and maintained in accordance with a management system. The scope and detail of the records will depend on the hazard associated with the facility and on the complexity of the operations and activities. (WS-G-6.1, para 4.3)

S-16:

The licensee shall ensure that each waste and spent fuel package or unpackaged spent fuel element can be uniquely identified with a marking system that will last for the storage period.

*Related **IAEA** safety standards:*

[…] There should be an unequivocal identification with a marking system that will last for the storage period. These records should be preserved and updated, to enable the implementation of the spent fuel management strategy whether disposal or reprocessing. (DS 371, para 3.27)

A tracking system for waste packages should be developed and implemented. The system should provide for the identification of waste packages and their locations and an inventory of waste stored. The sophistication of the waste tracking system required (e.g. including labelling and bar coding) will depend on the number of waste packages, the anticipated duration of storage of the waste and the hazard associated with it. (WS-G-6.1, para 4.11)

S-17:

The licensee shall implement an adequate system to provide up-to-date information on the radioactive inventory within the storage facility.

*Related **IAEA** safety standards:*

The operating organization should develop and maintain a records system […].These records should be preserved and updated, to enable the implementation of the spent fuel management strategy whether disposal or reprocessing. (DS 371, para 3.27)

The stored radioactive waste should be characterized (e.g. by radionuclide type, inventory, activity concentration, half-life and the physical, chemical and pathogenic properties of the waste) and the results should be documented in an inventory log. (WS-G-6.1, para 5.5)

S-18:

The owner and/or the licensee shall ensure that sufficient records are preserved and updated during the whole storage period (taking into account in particular the condition of waste and spent fuel package or unpackaged spent fuel element during storage), to enable implementation of its strategy for the management of waste or spent fuel, including disposal.

*Related **IAEA** safety standards:*

The operating organization of a spent fuel storage facility should receive detailed information concerning the characteristics of the spent fuel received for storage. This information should be supplied by the nuclear facility (i.e. power plant or research reactor) generating spent fuel (DS 371, para 6.123)

[…] The management system should be designed to ensure […] that the quality of the records and of subsidiary information such as the marking and labelling of waste packages is preserved. […] (WS-G-6.1, para 3.21)

2.2
Safety area: Design

-

The design of the storage facility should incorporate passive safety features as far as reasonably practicable, thereby minimising the reliance on active safety system, monitoring and human intervention to ensure safety. Where it is not reasonably practicable to incorporate passive safety features in the design, then the safety function will need to be fulfilled with active safety features. The SRLs in this subsection are connected with relevant design aspects.

2.2.1 Safety issue: Storage facility design requirements

S-19:
The storage facility shall be designed to fulfil the fundamental applicable safety functions:

- control of sub-criticality,
- removal of heat,
- radiation shielding,
- confinement of radioactive material,
- retrievability

during normal operation, anticipated operational occurrences and design basis accident conditions.

*Related **IAEA** safety standards:*

6.4. In general the storage facility should be designed to fulfil the main safety functions, i.e. maintaining subcriticality, removal of heat, containment of radioactive material and shielding from radiation, and in addition retrievability of the fuel […] […] (DS 371, para 6.4)

The following should be provided for in the design of storage facilities for radioactive waste for normal operations:

(a) Containment of the stored materials;
(b) Prevention of criticality (when storing fissile materials);
(c) Radiation protection (shielding and contamination control);
(d) Removal of heat (if applicable);
(e) Ventilation, as necessary;
(f) Inspection and/or monitoring of the waste packages, as necessary;
(g) Maintenance and repair of waste packages;
(h) Retrieval of the waste, whether for processing, repackaging or disposal;
(i) Inspection of waste packages and of the storage facility;
(j) Future expansion of the storage capacity, as appropriate;
(k) Transport of waste inside the storage facility to improve the flexibility of operations;

(l) Decommissioning. (WS-G-6.1, para 6.23)

The operating organization shall identify postulated initiating events that could lead to a release of radiation and/or significant amounts of radioactive material and associated chemical substances [...] (NS-R-5, para 6.8)

A design basis accident approach, or an equivalent methodology, shall be used to identify significant accident sequences. For each accident sequence identified, the safety functions, the corresponding SSCs important to safety and the administrative safety requirements that are used to implement the defence in depth concept shall be identified. (NS-R-5, para 6.9)

S-20:

The design of the storage facility shall take into account the expected operational lifetime of the facility to ensure that the safety conditions, the operational limits and conditions identified in the safety case will be met.

*Related **IAEA** safety standards:*

Predisposal radioactive waste management facilities shall be located and designed so as to ensure safety for the expected operating lifetime under both normal and possible accident conditions, and for their decommissioning. (GSR Part 5, Requirement 17)

S-21:

The design of the storage facility shall incorporate passive safety features as far as reasonably practicable.

*Related **IAEA** safety standards:*

[…]Due account shall be taken of the expected period of storage, and, to the extent possible, passive safety features shall be applied. For long term storage in particular, measures shall be taken to prevent degradation of the waste containment. (GSR Part 5, Requirement 11)

S-22:

The licensee shall demonstrate that design and construction of the facility are based on applicable standards and appropriate materials especially taking into account the expected lifetime of the facility.

*Related **IAEA** safety standards:*

The storage system, particularly the storage cask, should be constructed of suitable materials, using appropriate design codes and standards and construction methods, to maintain shielding and containment functions under the storage and loading/unloading conditions expected during its design lifetime unless adequate maintenance and/or replacement methods during operation can be demonstrated. […] (DS 371, Appendix I.54)

The need for and the extent of commissioning activities and tests will vary depending on the size, complexity and contents of the storage facility. Commissioning involves a logical progression of tasks and tests to demonstrate the correct functioning of specific equipment and features incorporated into the *design of the storage facility to provide for safe storage. The adequacy of the facility's design […] should be demonstrated and confirmed. (WS-G-6.1, para 4.17)*

S-23:

The radioactive waste and spent fuel storage facility shall be designed on the basis of assumed conditions for its normal operations and assumed incidents or accidents. The design basis shall be clearly and systematically defined and documented.

*Related **IAEA** safety standards:*

Predisposal radioactive waste management facilities shall be located and designed so as to ensure safety for the expected operating lifetime under both normal and possible accident conditions, and for their decommissioning. (GSR Part 5, Requirement 17)

S-24:

The licensee shall identify and classify structures, systems and components important to safety (SSCs), applying a graded approach.

*Related **IAEA** safety standards:*

The safety functions, and structures, systems and components important to safety (SSCs) shall be identified in the safety analysis report to the extent appropriate and in accordance with a graded approach. The SSCs provide barriers for the prevention of the occurrences of postulated initiating events (PIEs), the control and limitation of accident sequences and mitigation of the consequences (NS-R-5; para 2.12).

S-25:

The licensee shall address the ageing of SSCs and safety features of facilities for the storage of spent fuel and waste by establishing, if necessary, provisions for their maintenance, testing and inspection. Results derived from this program shall be used to review the adequacy of the design at appropriate intervals.[2]

*Related **IAEA** safety standards:*

In the design stage, design safety margins shall be adopted so as to accommodate the anticipated properties of materials at the end of their useful life. This is particularly important for fuel cycle facilities because of the range and characteristics of chemical and radiation conditions experienced in operational states and in accident conditions. Where details of the characteristics of materials are unavailable, a suitable material surveillance programme shall be implemented by the operating organization. Results derived from this programme shall be used to review the adequacy of the design at appropriate intervals. This may require provisions in the design for the monitoring of materials whose mechanical properties may change in service owing to factors such as fatigue (e.g. from cyclic mechanical or thermal loadings), stress corrosion, erosion, chemical corrosion or the induction of changes by irradiation. (NS-R-5, para 6.17)

[2] This may require design provisions to monitor materials whose mechanical properties may change in service owing to such factors as fatigue (cyclic mechanical or thermal loadings), stress corrosion, erosion, chemical corrosion or radiation induced changes.)

Before the start of operations, the operator should prepare a programme of periodic maintenance, testing and inspection of systems that are essential to safe operation. The need for maintenance, testing and inspection should be addressed from the design stage. […] Systems and components that should be considered for periodic maintenance, testing and inspection may include:

(a) Waste containment systems, including tanks and other containers;
(b) Waste handling systems, including pumps and valves;
(c) Heating and/or cooling systems;
(d) Radiation monitoring systems;
(e) Calibration of instruments;
(f) Ventilation systems;
(g) Normal and standby systems for electrical power supply;
(h) Utilities and auxiliary systems such as systems for water, gas and compressed air;
(i) The system for physical protection;
(j) Building structures and radiation shielding;
(k) Fire protection systems. (WS-G-6.1, para 6.79)

The operation of a spent fuel storage facility should include an appropriate programme of maintenance, inspection and testing of items important to safety, i.e. structures, systems and components. Safe access to all structures, systems, areas and components requiring periodic maintenance, inspection and testing should be provided. Such access should be sufficient for the safe operation of all required tools and equipment and for the installation of spares. (DS 371, para 6.108)

S-26:

The licensee shall establish operational limits and conditions (OLCs) in order to maintain the storage facility and waste and spent fuel packages or unpackaged spent fuel elements in a safe state during facility operation.

*Related **IAEA** safety standards:*

[…] All operations and activities important to safety have to be subject to documented limits, conditions and controls, and have to be carried out by trained, qualified and competent personnel.

(GSR Part 5, para 5.19)

The OLCs are the set of rules that establish parameter limits, the functional capability and the performance levels of equipment and personnel for the safe operation of a facility. (NS-R-5, para 2.13)

Operational limits and conditions shall be prepared before operation of the facility commences. (NS-R-5, para 9.21)

S-27:

The defined OLCs (see S-26) shall consider, in particular, and as appropriate:

- environmental conditions within the store (e. g. temperature, humidity, contaminants, …);
- the effects of heat generation from waste or spent fuel, covering both each individual waste and spent fuel packages or unpackaged spent fuel elements as well as the whole store;

- potential aspects of gas generation from waste or spent fuel, in particular the hazards of fire ignition, explosion, waste and spent fuel package or unpackaged spent fuel element deformations and radiation protection aspects;
- criticality prevention, covering both each individual waste and spent fuel packages or unpackaged spent fuel elements as well as the whole store (including operational occurrences and accidental conditions);
- suitability for handling and retrieval.

*Related **IAEA** safety standards:*

Operational limits and conditions for a spent fuel storage facility should be developed on the basis of the following:

(a) Design specifications and operating parameters and the results of commissioning tests;
(b) The sensitivity of items important to safety and the consequences of events following the failure of items, the occurrence of specific events or variations in operating parameters;
(c) The accuracy and calibration of instrumentation equipment for measuring safety related operating parameters; […] (DS 371, para 6.102)

Operational limits and conditions form an important part of the basis on which operation is authorized and as such should be incorporated into the technical and administrative arrangements that are binding on the operating organization and operating personnel. Operational limits and conditions for spent fuel storage facilities, which result from the need to meet legal and regulatory requirements, should be developed by the operating organization and subject to approval by the regulatory body as part of the licence conditions. [….] (DS 371, para 6.103)

While all operations can be directly or indirectly related to some aspect of safety, the aim of operational limits and conditions should be to manage and control the basic safety hazard in the facility and they should be directed toward:

(a) Preventing situations which might lead to unplanned exposure of people (workers and the public) to radiation; and
(b) Mitigating the consequences of such events should they occur. (DS 371, para 6.104)

Gas generation by radiolysis or chemical reaction may be associated with the storage of radioactive waste. The concentration of gases in air shall be kept below hazardous levels to avoid, for example, explosive gas/air mixtures. (WS-R-2 5.26)

If necessitated by the nature of the radioactive waste, dissipation of heat from the waste shall be ensured and criticality shall be prevented. (WS-R-2; para 5.28)

S-28:

The design of the facility shall take into account all relevant postulated initiating events (PIEs), depending on the storage characteristics. A list of potential PIE is provided in the appendix.

*Related **IAEA** safety standards:*

The operating organization shall identify postulated initiating events that could lead to a release of radiation and/or significant amounts of radioactive material and associated chemical substances. [...]

The set of postulated initiating events shall include both internally and externally initiated events (NS-R-5, para 6.8).

The postulated initiating events that may influence the design of the spent fuel storage facility and the integrity and safety of the spent fuel should be identified […]. (DS 371, para 5.19)

In addition to radiological hazards, external hazards (e.g. fire or explosion), which may contribute to radiologically significant consequences, should also be considered in the design of storage facilities for radioactive waste. (WS-G-6.1, para 6.25)

S-29:

The criticality safety shall be achieved by design as far as practicable. If burnup credit is adopted, compliance with the limiting burnup level shall be verified by administrative and operational controls.

*Related **IAEA** safety standards:*

As far as reasonably practicable, criticality hazard shall be controlled by means of design. (NS-R-5, para 6.43)

Approval to consider burnup credit in the safety assessment should be granted only if based on design engineered safety features and operational controls. Operational controls provide defence in depth and contribute to maintaining subcritical conditions. The minimum required burnup value should be verified by independent measurement. (DS 371, Appendix II, para II.8)

S-30:

The licensee shall make design arrangements for fire safety on the basis of a fire safety analysis and implementation of defence in depth (prevention, detection, control and mitigation of a fire).

*Related **IAEA** safety standards:*

The operating organization shall make design provisions for fire safety on the basis of a fire safety analysis and the implementation of the concept of defence in depth (i.e. for prevention, detection, control and mitigation). (NS-R-5, para 6.55).

2.2.2 Safety issue: Handling and retrieval requirements

S-31:

The handling equipment shall be designed particularly to take account of radiation protection aspects, ease of maintenance and minimization of the probability and consequences of associated incidents and accidents.

*Related **IAEA** safety standards:*

Handling equipment should be designed to minimize the probability and consequence of incidents and accidents, and to minimize the potential for damaging spent fuel, spent fuel assemblies, and storage or transport casks. […] (DS 371, para 6.49)

Waste handling equipment should be designed to include provision for the following:

(a) Safe operation under all anticipated conditions;
(b) Avoiding damage to the waste package;
(c) Safe handling of defective or damaged waste packages;
(d) Minimizing contamination of the equipment itself;
(e) Avoiding the spread of contamination. (WS-G-6.1, para 6.32)

S-32:

The storage facility shall be designed in such a way that any waste or spent fuel package or unpackaged spent fuel can be retrieved within an appropriate time, at the end of the facility operation or in order to intervene in the event of unexpected faults.

*Related **IAEA** safety standards:*

Waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management. […] (GSR Part 5, Requirement 11)

S-33:

The storage facility shall be designed so that waste and spent fuel packages or unpackaged spent fuel elements can be inspected to verify their continued integrity.

*Related **IAEA** safety standards:*

Waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management. […] (GSR Part 5, Requirement 11)

Provision has to be made for the regular monitoring, inspection and maintenance of the waste and of the storage facility to ensure their continued integrity. […] (GSR Part 5 para 4.22)

2.2.3 Safety issue: Storage capacity

S-34:

The licensee shall ensure that reserve storage capacity is included in the design or is otherwise available to allow for inspection, retrieval, maintenance or remedial work.

*Related **IAEA** safety standards:*

Design aspects associated with the layout of a spent fuel storage facility are set out in the following: [...] (g) Space should be provided to permit the inspection of spent fuel and inspection and maintenance of components, including spent fuel handling equipment; [...] (DS 371, para 6.47)

The facility should have a reserve storage capacity, which should be included in the design or should be otherwise available, e.g. to allow for reshuffling of spent fuel casks or unpackaged spent fuel elements for inspection, retrieval or maintenance work. The reserve capacity should be such that the largest type of storage cask can be unloaded or, in the case of a modular storage facility, that at least one module can be unloaded. (DS 371, para 6.15)

There should be reserve storage capacity available to accommodate waste arising in various situations. Such situations may include abnormal conditions (e.g. the need to empty a leaking tank) or periods when modifications or refurbishments are being undertaken. (WS-G-6.1, para 6.58)

2.3
Safety Area: Operation

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2.3.1 Safety issue: Conduct of Operation

S-35:
The storage facility shall be operated so that in accordance with the inspection program as defined in S-48 waste and spent fuel packages or unpackaged spent fuel elements can be inspected.

*Related **IAEA** safety standards:*

Waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management. […] (GSR Part 5, Requirement 11)

S-36:
The licensee shall ensure that the reserve storage capacity will stay available for retrieved waste and spent fuel packages or unpackaged spent fuel elements.

*Related **IAEA** safety standards:*

The facility should have a reserve storage capacity, which should be included in the design or should be otherwise available, e.g. to allow for reshuffling of spent fuel casks or unpackaged spent fuel elements for inspection, retrieval or maintenance work. The reserve capacity should be such that the largest type of storage cask can be unloaded or, in the case of a modular storage facility, that at least one module can be unloaded. (DS 371, para 6.15)

There should be reserve storage capacity available to accommodate waste arising in various situations. Such situations may include abnormal conditions (e.g. the need to empty a leaking tank) or periods when modifications or refurbishments are being undertaken. (WS-G-6.1, para 6.58)

2.3.2 Safety issue: Emergency Preparedness

If for the set of design basis accidents as consequence from the safety case events requiring protective measures cannot be excluded, planned emergency arrangements will be required. These emergency plans should be proportionate taking account of the magnitude of the accident consequence. For some facilities (such as with low radioactive inventory) an off-site emergency plan may not be required, which must be justified and the off-site aspects of this safety issue will not apply.

S-37:

Based upon an assessment of reasonably foreseeable events and situations that may require protective measures the licensee shall provide arrangements for responding effectively to events requiring protective measures at the scene for:

(a) regaining control of any emergency arising at the site, including events related to combinations of non-nuclear and nuclear hazards;
(b) preventing or mitigating the consequences at the scene of any such emergency and
(c) co-operating with external emergency response organizations in preventing adverse health effects in workers and the public.

*Related **IAEA** safety standards:*

Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents. (SF-1, Principle 9)

The primary goals of preparedness and response for a nuclear or radiation emergency are:

- *To ensure that arrangements are in place for an effective response at the scene and, as appropriate, at the local, regional, national and international levels, to a nuclear or radiation emergency;*
- *To ensure that, for reasonably foreseeable incidents, radiation risks would be minor;*
- *For any incidents that do occur, to take practical measures to mitigate any consequences for human life and health and the environment. (SF-1; para 3.34)*

Emergency preparedness and response arrangements commensurate with the threat category of the facility, […], should be developed and implemented. (WS-G-6.1, para 5.14)

The operator should draw up emergency plans based on the potential radiological impacts or accidents and be prepared to respond to accidents at all times as indicated in the emergency plans. (DS 371, para 3.28)

The potential radiological impacts of accidents should be assessed by the operating organization and reviewed by the regulatory body [21]. Provision should be made to ensure that there is an effective capability to respond to accidents. Considerations should include the development of scenarios of anticipated sequences of events (see Section 5) and the establishment of emergency procedures and emergency plan to deal with each of the scenarios, including checklists and lists of persons and organizations to be alerted. (DS 371, para 6.73)

S-38:

The licensee shall

- prepare an on-site emergency plan as basis for preparation and conduct of emergency measures (An example for the contents of such emergency plan is given in app. 2),
- establish the necessary organizational structure for clear allocation of responsibilities, authorities and arrangements for coordinating facility activities and cooperating with external response agencies throughout all phases of an emergency and

- ensure, that based on the on-site emergency plan trained and qualified personnel, facilities and equipment need to control an emergency are appropriate, reliable and available at the time.

*Related **IAEA** safety standards:*

The operating organization, taking into account the potential hazards of the facility, shall develop an emergency plan in coordination with other bodies having responsibilities in an emergency, including public authorities; shall establish the necessary organizational structure; and shall assign responsibilities for managing emergency response. (NS-R-5; para 9.62).

Emergency response procedures should be documented, made available to the personnel concerned and kept up to date. Exercises should be held periodically to test the emergency response plan and the degree of preparedness of the personnel. Inspections should be performed regularly to ascertain whether the equipment and other resources needed in the event of an emergency are available and in working order. (DS 371, para 6.74)

In addition to providing operating procedures and contingency procedures as described above, the operating organization should also develop an emergency plan […] (DS 371, para 6.99)

The appropriate responsible authorities shall ensure that:

(a) emergency plans [are] prepared and approved for any practice or source which could give rise to a need for emergency intervention;
(b) [response organizations are] involved in the preparation of emergency plans, as appropriate;
(c) the content, features and extent of emergency plans take into account the results of any [threat assessment] and any lessons learned from operating experience and from [emergencies] that have occurred with sources of a similar type […];
(d) emergency plans [are] periodically reviewed and updated." […] (GS-R-2, para 5.17)

Adequate tools, instruments, supplies, equipment, communication systems, facilities and documentation (such as procedures, checklists, telephone numbers and manuals) shall be provided for performing the functions specified in Section 478. These items and facilities shall be selected or designed to be operational under the postulated conditions (such as the radiological, working and environmental conditions) that may be encountered in the emergency response, and to be compatible with other procedures and equipment for the response (such as the communication frequencies of other response organizations), as appropriate. These support items shall be located or provided in a manner that allows their effective use under postulated emergency conditions (GS-R-2, para 5.25)

The operator and the response organizations shall identify the knowledge, skills and abilities necessary to be able to perform the functions specified […]. The operator and the response organizations shall make arrangements for the selection of personnel and for training to ensure that the personnel have the requisite knowledge, skills, abilities, equipment, and procedures and other arrangements to perform their assigned response functions. The arrangements shall include ongoing refresher training on an appropriate schedule and arrangements for ensuring that personnel assigned to positions with responsibilities for emergency response undergo the specified training. (GS-R-2, para 5.31)

S-39:

The on-site emergency plan shall be submitted to the regulatory body. At regular intervals there shall be emergency exercises, some of which shall be witnessed by the regulatory body.

Some of these exercises shall be integrated and shall include the participation of all organizations concerned. The plan shall be subject to review and updating in light of the experience gained.

*Related **IAEA** safety standards:*

In developing the emergency response arrangements, consideration has to be given to all reasonably foreseeable events. Emergency plans have to be exercised periodically to ensure the preparedness of the organizations having responsibilities in emergency response. (SF-1; para 3.37)

The emergency plan shall be approved by the regulatory body as appropriate and shall be tested in an exercise before radioactive material is introduced into the facility. There shall thereafter be exercises of the emergency plan at suitable intervals, some of which shall be observed by the regulatory body. Some of these exercises shall be integrated with local, regional and national response organizations, as appropriate, and shall involve the participation of as many as possible of the organizations concerned. The plans shall be subject to review and to updating in the light of the experience gained. (NS-R-5; para 9.66)

2.3.3 Safety issue: Operational Experience Feedback

S-40:

The licensee shall establish and conduct an Operating Experience Feedback (OEF) program to collect, screen, analyze and document safety relevant operating experience and events at the facility in a systematic way. Relevant operational experience and events reported by other facilities shall also be considered as appropriate.

*Related **IAEA** safety standards:*

Despite all measures taken, accidents may occur. The precursors to accidents have to be identified and analysed, and measures have to be taken to prevent the recurrence of accidents. The feedback of operating experience from facilities and activities - and, where relevant, from elsewhere - is a key means of enhancing safety. Processes must be put in place for the feedback and analysis of operating experience, including initiating events, accident precursors, near misses, accidents and unauthorized acts, so that lessons may be learned, shared and acted upon. (SF-1; para 3.17)

Adequate arrangements should be made for the review and approval of operating procedures, the systematic evaluation of operating experience, including that of other facilities, and the taking of corrective actions in a timely and appropriate manner to prevent and counteract developments adverse to safety. Provision should be made for controlling the distribution of operating procedures, in order to guarantee that operating personnel have access to only the latest approved edition. (DS 371, para 6.91)

In the generation and storage of waste, as well as subsequent management steps, a safety culture should be fostered and maintained to encourage a questioning and learning attitude to protection and safety and to discourage complacency. (WS-G-6.1, para 2.6)

S-41:

The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

*Related **IAEA** safety standards:*

All organizations involved in the process of operational experience feedback should screen information on events, taking into account their own needs. Operating organizations should have the objective of enhancing safety, plant availability and commercial performance by identifying the causes of events so as to be able to avoid their recurrence, and by evaluating the applicability of good practices used by others. [...] (NS-G-2.11, para 3.3)

Operating experience and events at the facility and reported by similar facilities should be collected, screened and analysed in a systematic way. Conclusions should be drawn and implemented by means of an appropriate feedback procedure [….]. (DS 371, para 6.100, see also para 6.91)

2.3.4 Safety issue: Operation facility modification

S-42:

Modifications of design, equipment, storage conditions, waste or spent fuel characteristics, control or management, especially changes of SSCs, OLCs or operational procedures in a spent fuel or radioactive storage shall be subject to planning, assessment, review and authorization processes commensurate to the importance to safety of the modification. These processes shall ensure that the modifications will not impact adversely the safety of the facility or associated facilities or the further management of spent fuel or waste.

*Related **IAEA** safety standards:*

The operating organization shall establish a process whereby its proposals for changes in design, equipment, feed material characteristics, control or management are subject to a degree of assessment and scrutiny appropriate to the safety significance of the change, so that the direct and wider consequences of the modification are adequately assessed. The process shall include a review of possible consequences to ensure that a foreseen modification or change in one facility will not adversely affect the operability or safety of associated or adjacent facilities (NS-R-5; para 9.35)

S-43:

Before introducing a modification according to S-42, personnel shall, as appropriate, have been trained according to the new operating procedures and all relevant documents necessary for facility operation shall have been updated.

*Related **IAEA** safety standards:*

[…] Provisions should be made for implementing a controlled distribution of operational procedures, in order to guarantee that operating personnel have only the last approved edition. (DS 371, para 6.91)

In accordance with the management system, arrangements should be in place for the review and approval of operating procedures and for the communication to operating personnel of any revisions. Periodic reviews should be undertaken to take account of operational experience. Any revisions should *be adopted only after they have been reviewed to ensure compliance with operational limits and conditions, approved by authorized persons and documented. (WS-G-6.1, Para 6.75)*

The operating organization should ensure that the appropriate revisions to plant procedures, personnel training and plant simulators necessitated by the modifications are implemented in a complete, correct and timely manner as part of the implementation process.(NS-G-2.3 para 3.9)

2.3.5 Safety issue: Maintenance, periodic testing and inspection

S-44:

A maintenance, periodic testing and inspection program shall be conducted according to written procedures in order to ensure that SSCs are able to function in accordance with the design intents and safety requirements.

*Related **IAEA** safety standards:*

Maintenance, calibration, periodic testing and inspection shall be performed to ensure that SSCs are able to function in accordance with the design intent and with safety requirements. In this context, the term maintenance includes both preventive and corrective actions. Maintenance, calibration and periodic testing shall also be carried out on the equipment necessary for implementation of the on-site emergency plan (NS-R-5; para 9.28).

S-45:

The extent of the program for maintenance, periodic testing or inspection of SSCs shall be in accordance with the facility safety case.

*Related **IAEA** safety standards:*

The frequency for maintenance, calibration, periodic testing and inspection of SSCs shall be in accordance with the facility licensing documentation. (NS-R-5; para 9.30).

S-46:

The result of maintenance, periodic testing and inspection shall be recorded and assessed.

*Related **IAEA** safety standards:*

The results of maintenance, testing and inspection shall be recorded and assessed (NS-R-5; para 9.32).

S-47:

The maintenance, periodic testing and inspection programs shall be reviewed at regular intervals to incorporate the lessons learned from experience.

*Related **IAEA** safety standards:*

The maintenance, calibration, periodic testing and inspection programmes shall be reviewed at regular intervals to incorporate the lessons learned from experience (NS-R-5; para 9.33).

S-48:

The licensee shall develop an inspection program for the verification of the continuing compliance of waste and spent fuel packages or unpackaged spent fuel stored with the limits specified in the safety case to ensure continued functionality of safety features on which safety case is based. This program shall address:

- the required environmental conditions within the storage facility,
- the state of waste and spent fuel packages or unpackaged spent fuel elements.

*Related **IAEA** safety standards:*

[…] Safety related operating instructions shall be prepared before operations commence. Operating instructions shall clearly describe the methods of operating, including all checks, tests, calibrations and inspections necessary to ensure compliance with the operational limits and conditions […]. (NS-R-5, para 9.22)

The integrity of stored spent fuel should be monitored in the operation of a spent fuel storage facility. [….] (DS 371, para 6.101)

2.3.6 Safety issue: Specific contingency plans

S-49:

The licensee's procedures for the receipt of waste and spent fuel packages or unpackaged spent fuel elements shall contain provisions to deal safely with those that fail to meet the acceptance criteria, e. g. returning to the owner, taking remedial actions.

*Related **IAEA** safety standards:*

Acceptance criteria should be developed for the spent fuel storage facility and the spent fuel, taking into account all relevant operational limits and conditions and the future reprocessing or disposal requirements, including retrieval. Before spent fuel is transferred to the storage facility, acceptance must be given by the operator and the respective legal authority. Contingency plans should be available on how to deal safely with spent fuel that does not comply with acceptance criteria. (DS 371, para 6.118)

The operators' procedures for the reception of waste have to contain provisions for safely managing waste that fails to meet the acceptance criteria; for example, by taking remedial actions or by returning the waste.(GSR Part 5, para 4.26)

S-50:

The licensee shall have plans and establish appropriate contingency arrangements for waste and spent fuel packages or unpackaged spent fuel elements that are not retrievable by normal means or show signs of degradation.

*Related **IAEA** safety standards:*

Spent fuel assemblies that have become damaged as a result of mechanical events, should be kept separate from intact fuel and provided with appropriate monitoring to detect any outer containment *failure. Consideration should be given to contingency arrangements on how to deal with spent fuel that is not retrievable by normal means or that cannot be transported easily. (DS 371, para 6.131)*

Procedures should be developed for the safe operation of a large waste storage facility. The extent and the degree of detail of specific procedures should be commensurate with the safety significance of the particular subject of the procedures and should cover, where applicable: [...]

(i) Contingency and emergency arrangements; [...] (WS-G-6.1, para 6.3)

2.3.7 Safety issue: Requirements for acceptance of waste and spent fuel packages and unpackaged spent fuel elements

S-51:

The owner and/or the licensee is responsible for ensuring that the waste and spent fuel packages and unpackaged spent fuel elements fulfil all relevant requirements such as:

- compatibility with handling, transport and storage requirements, including suitability for retrieval and transport after the anticipated storage period;
- known or likely requirements for subsequent disposal or other management aspects included in the owner's waste and spent fuel management strategy, such as the need for further treatment or conditioning of the waste or spent fuel.

*Related **IAEA** safety standards:*

[…]It is necessary that those persons responsible for a particular step in the predisposal management of radioactive waste, or for an operation in which waste is generated, adequately recognize these interactions and relationships so that the safety and the effectiveness of the predisposal management of radioactive waste may be considered in an integrated manner. This includes taking into account the identification of waste streams, the characterization of waste, and the implications of transporting and disposing of waste. There are two issues in particular to be addressed: compatibility (i.e. taking actions that facilitate other steps and avoiding taking decisions in one step that detrimentally affect the options available in another step) and optimization (i.e. assessing the overall options for waste management with all the interdependences taken into account). […] (GSR Part 5, para 3.22)

S-52:

The licensee shall establish acceptance criteria for its storage facility.

*Related **IAEA** safety standards:*

Waste packages and unpackaged waste that are accepted for processing, storage and/or disposal shall conform to criteria that are consistent with the safety case. (GSR Part 5, Requirement 12)

The responsibilities of the operator of a large storage facility for radioactive waste would typically include: [...]

(d) Developing and applying acceptance criteria for the storage of radioactive waste; [...] (WS-G-6.1, para 3.12)

The responsibilities of the operating organization of a spent fuel storage facility would typically include: [...]

(d) Developing and applying acceptance criteria for the storage of spent fuel as approved by the regulatory body; [...] (DS 371, para 3.17)

S-53:

These acceptance criteria shall take into account storage conditions and shall ensure compatibility with the safety case of the storage facility, and shall ensure suitability for handling and retrieval.

*Related **IAEA** safety standards:*

Waste acceptance criteria have to be developed that specify the radiological, mechanical, physical, chemical and biological characteristics of waste packages and unpackaged waste that are to be processed, stored or disposed of; for example, their radionuclide content or activity limits, their heat output and the properties of the waste form and packaging. (GSR Part 5, para 4.24)

Waste acceptance criteria should be developed for the storage facility, with account taken of all relevant operational limits and future requirements for disposal, if the latter are known. (WS-G-6.1, para 6.6)

Acceptance criteria should be developed for the spent fuel storage facility and the spent fuel, with account taken of all relevant operational limits and conditions and future demands for reprocessing or disposal, including retrieval of the spent fuel. (DS 371, para 6.1118)

S-54:

The licensee shall make sure that appropriate processes are set up and implemented, involving auditing, inspection and testing, to ensure that waste and spent fuel packages or unpackaged spent fuel elements meet the acceptance criteria for storage.

*Related **IAEA** safety standards:*

Upon receipt, waste packages should be checked for leakage and surface contamination and to ensure that they are consistent with the documentation. Waste characterization, process control and process monitoring should be applied within a formal management system. (WS-G-6.1 para 6.9)

Upon receipt, spent fuel casks should be checked for gamma and neutron radiation levels, leakage, surface contamination and to ensure that they are consistent with the documentation. Characterization of the spent fuel including process control and process monitoring, should be applied within a formal management system. (DS 371, para 6.120)

2.4 Safety area: Safety verification

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2.4.1 Safety issue: Contents and updating of the safety case

S-55:

The licensee shall provide a safety case and use it as a basis for continuous support of safe operation throughout the lifetime of a facility.

*Related **IAEA** safety standards:*

The operator shall prepare a safety case and a supporting safety assessment. In the case of a step by step development, or in the event of modification of the facility or activity, the safety case and its supporting safety assessment shall be reviewed and updated as necessary. (GSR Part 5, Requirement 13)

S-56:

The licensee shall use the safety case also as a basis for assessing the safety implications of changes to the facility or to operating practices.

*Related **IAEA** safety standards:*

[…] in the event of modification of the facility or activity, the safety case and its supporting safety assessment shall be reviewed and updated as necessary. (GSR Part 5, Requirement 13)

S-57:

The safety case shall cover both the facility itself and the waste and spent fuel packages or unpackaged spent fuel elements and their respective safety-relevant features. The safety case shall include a description of how all the safety aspects of the site, the design, construction and operation, as well as provisions for decommissioning of the facility, and the managerial controls satisfy the regulatory requirements (for a typical list of contents see Annex 3).

*Related **IAEA** safety standards:*

The safety case for a predisposal radioactive waste management facility shall include a description of how all the safety aspects of the site, the design, operation, shutdown and decommissioning of the facility and the managerial controls satisfy the regulatory requirements. The safety case and its supporting safety assessment shall demonstrate the level of protection provided and shall provide assurance to the regulatory body that safety requirements will be met. (GSR Part 5, Requirement 14)

S-58:

The licensee shall update the safety case to reflect

- modifications and new regulatory requirements and relevant standards;

- results of the periodic safety review;
- results from analysis of incidents

as soon as practicable and in accordance with safety relevance of the modification after the new information is available and applicable.

*Related **IAEA** safety standards:*

The operator shall carry out periodic safety reviews and shall implement any safety upgrades required by the regulatory body following this review. The results of the periodic safety review shall be reflected in the updated version of the safety case for the facility. (GSR part 5 Requirement 16, also Requirement 13, see S-55)

The licensing documentation shall be maintained and updated during the operational lifetime of the facility on the basis of the experience and knowledge gained and in accordance with the regulatory requirements, with account taken of modifications to the facility (NS-R-5; para 2.15).

The safety case and supporting safety assessments including their implementing management systems should be periodically reviewed in accordance with regulatory requirements. The review of management systems should include aspects of safety culture. In addition, they should be reviewed and updated:

(a) When there is any significant change to the installation or radionuclide inventory that affects safety;
(b) When changes occur in the site characteristics that may impact on the storage facility, e.g. industrial development, nearby population;
(c) When significant changes in knowledge and understanding occur (such as from research data or operational experience feedback);
(d) When there is an emerging safety issue due to a regulatory concern or an incident; and
(e) Periodically at predefined periods as specified by the regulatory body. Some Member States specify not less than once in ten years.

Safety should be reassessed in the case of significant, unexpected deviations in the storage conditions, e.g. if safety relevant spent fuel properties change and begin to deviate from those taken as a basis in the safety assessment. (DS 371, para 5.27)

2.4.2 Safety issue: Periodic safety review

S-59:
The licensee shall carry out at regular intervals a review of the safety of the facility (PSR). The review shall be made periodically, at a frequency which shall be established by the national regulatory framework (e. g. every ten years).

*Related **IAEA** safety standards:*

The process of safety assessment for facilities and activities is repeated in whole or in part as necessary later in the conduct of operations in order to take into account changed circumstances (such as the application of new standards or scientific and technological developments), the feedback of operating experience, modifications and the effects of ageing. For operations that continue over long periods of time, assessments are reviewed and repeated as necessary. Continuation of such operations is subject *to these reassessments demonstrating to the satisfaction of the regulatory body that the safety measures remain adequate. (SF-1; para 3.16)*

The safety assessment and the management systems within which it is conducted have to be periodically reviewed at predefined intervals in accordance with regulatory requirements. […] (GSR Part 5 para 5.12)

S-60:

The scope and methodology of the PSR shall be clearly defined and justified. The PSR shall confirm the compliance with the licensing requirements. It shall also identify and evaluate the safety significance of differences from applicable current safety standards and good practices and take into account the cumulative effects of changes to procedures, modifications to the facility and the operating organization, technical developments, operational experience accumulated and ageing of SSCs. It shall include consideration of the acceptance criteria for waste and spent fuel packages and unpackaged spent fuel elements and any deviation from these criteria during storage.

*Related **IAEA** safety standards:*

See also S-59

In accordance with the national regulatory requirements, the operating organization shall carry out periodic safety reviews to confirm that the licensing documentation remains valid and that modifications made to the facility, as well as changes in its operating arrangements or utilization have been accurately reflected in the licensing documentation. In conducting these reviews, the operating organization shall expressly consider the cumulative effects of changes to procedures, modifications to the facility and the operating organization, technical developments, operating experience and ageing. (NS-R-5; para 4.26)

S-61:

The results of the PSR shall be documented. All reasonably practicable improvement measures shall be subject to an action plan.

*Related **IAEA** safety standards:*

Protection must be optimized to provide the highest level of safety that can reasonably be achieved. (SF-1; Principle 5)

Central to the management and verification of safety is the ability of an organization to establish effective review and improvement as an ongoing process. To establish this process, the operating organization shall periodically conduct a review of the facility's operational and safety performance to identify, investigate and correct adverse trends that may have an impact on safety. Such a process shall also cover safety culture, and the improvement of attitudes and the operating environment for safe operation. (NS-R-5 para 9.70)

The results of the reviews and the PSR reports should be recorded in a systematic and auditable manner. (DS 426, para 8.10)

Appendix 1
Postulated Initiating Events

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External postulated events

Natural phenomena

- Extreme weather conditions (precipitation: rain, snow, ice, hail, wind, lightning, high or low temperature, humidity);
- flooding
- earthquake
- natural fires
- effect of terrestrial and aquatic flora and fauna (blockage of inlet and outlets, damages on structure)

Human induced phenomena

- fire, explosion or release of corrosive/hazardous substance (from surrounding industrial and military installations or transport infrastructure);
- aircraft crash (accidents);
- missiles due to structural/mechanical failure in surrounding installations;
- flooding (failure of a dam, blockage of a river);
- power supply and potential loss of power;
- civil strife (infrastructure failure, strikes and blockages);

Internal postulated events

- loss of energy and fluids: electrical power supplies, air and pressurised air, vacuum, super heated water and steam, coolant, chemical reagents and ventilation;
- improper use of electricity and chemicals;
- mechanical failure including drop loads, rupture (pressure retaining vessels or pipes), leaks (corrosion), plugging;
- instrumentation and control, human failures;
- internal fires and explosions (gas generation, process hazards);
- flooding, vessel overflows;

*Related **IAEA** safety standards:*

External postulated initiating events

<u>Natural phenomena</u>

- *Extreme weather conditions:*
  Precipitation including rain, hail snow, ice, frazil ice, wind including tornadoes, hurricanes, cyclones, dust or sand storms, lightning, extreme high or low temperature, extreme humidity;
- *Flooding,*
- *Earthquakes and eruption of volcanoes*
- *Natural fires*
- *Effects of terrestrial and aquatic flora and fauna (leading to blockages of inlets and outlets, and damage to structures)*

<u>Human induced phenomena</u>

- *Fires, explosions or releases of corrosive/hazardous substances*
- *(from surrounding industrial and military installations or transport infrastructure)*
- *Aircraft crashes*
- *Missile strikes(arising from structural/mechanical failure in surrounding installations);*
- *Flooding (e. g. failure of a dam, blockage of a river);*
- *Loss of power supply*
- *Civil strife (leading to infrastructure failure, strikes and blockages).*

Internal postulated events

- *Loss of energy and fluids (loss of electrical power supplies, air and compressed air, vacuum, super heated water and steam, coolant, chemical reagents, and ventilation;*
- *Failures in use of electricity or chemicals;*
- *Mechanical failure including drop loads, rupture (pressure retaining vessels or pipes), leaks (due to corrosion), plugging;*
- *Failure of and human error with instrumentation and control systems;*
- *Internal fires and explosions (due to gas generation and, process hazards);*
- *Flooding (e. g. vessel overflows).*

(selected from NS-R-5, Annex 1)

Appendix 2
Contents of the On-Site Emergency Plan

-

The emergency plan of the licensee shall provide for arrangements to address the following:

Emergency preparedness

(1) The requirements for personnel training;
(2) the list of potential accidents, including combinations of nuclear and non-nuclear hazards as necessary. If relevant, the description of possible severe accidents and their consequences;
(3) the conditions and criteria under which an emergency shall be declared, and a description of suitable means for alerting response personnel and the public authorities;
(4) an inventory of the emergency equipment to be kept in readiness at specified locations;

Personal and organizational responsibilities and provisions

(1) The designation of persons who will be responsible for directing on-site activities and for ensuring liaison with off-site organizations;
(2) a list of job titles and/or functions of persons empowered to declare it;
(3) the chain of command and communication, including a description of related facilities and procedures; there shall be a means of informing all persons on the site of the actions to be taken in the event of an emergency;
(4) the actions to be taken by persons and organizations involved in the implementation of the plan;
(5) provisions for declaring the termination of an emergency.;

Assessment of impacts of incidents

(1) The arrangements for assessment of the radiological conditions on and off the site (water, vegetation, soil, air sampling);
(2) assessment of the state of the facility;

Mitigation of adverse consequences

(1) Provisions for minimizing the exposure of persons to ionising radiation and for ensuring medical treatment of casualties;
(2) the actions to be taken on the site to limit the extent of radioactive release and spread of contamination;

*Related **IAEA** safety standards:*

The emergency plan of the operating organization shall include:

(a) The designation of persons who will be responsible for directing on-site activities and for ensuring liaison with off-site organizations;
(b) The requirements for personnel training;
(c) A listing of possible accidents and, if relevant, descriptions of the accidents and their foreseeable consequences;
(d) The conditions under which, and criteria according to which an emergency shall be declared, a list of job titles and/or functions of the persons empowered to declare an emergency, and a description of suitable means for alerting response personnel and public authorities;
(e) The arrangements for assessment of radiological conditions on and off the site (for water, vegetation and soil and by air sampling);
(f) Provisions for minimizing the exposure of persons to radiation and for ensuring the medical treatment of casualties;
(g) Assessment of the state of the facility and the actions to be taken on the site to limit the extent of radioactive releases and the spread of contamination;
(h) The chain of command and communication, including a description of related facilities and procedures;
(i) An inventory of the emergency equipment to be kept in readiness at specified locations;
(j) The actions to be taken by persons and organizations involved in the implementation of the emergency plan;
(k) Provisions for declaring the termination of an emergency. (NS-R-5; para 9.63).

Appendix 3

Typical Contents of a Safety Case

-

The preparation of a safety case including the supporting safety assessment is a step by step development. The safety case is progressively developed and refined as the storage facility project proceeds. The proposed content of the safety case takes into account the scope of this document (see chapter 01.3) and therefore does not specifically address items such as EIA, physical protection including safeguards, etc.

The detailed structure and format of the safety case depends on the requirements of national regulatory systems and may be different country by country.

The safety case shall as appropriate among others:

– describe the site characteristics, storage facility layout, design basis and safety functions of the facility and contain a list of safety relevant SSCs to demonstrate how safety is achieved throughout the anticipated storage period;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:

(a) A description of the site and facility (including the maximum expected inventory of spent fuel and its acceptance criteria, the storage facility and its characteristics, structures, systems and components, including the characteristics of items important to the safety of the spent fuel storage facility, in accordance with the requirements of its licence) and a specification of applicable regulations and guidance;
… (DS 371, para. 5.22)

The content of a safety case for a facility may vary between Member States but the components of the safety case for a predisposal waste management facility or activity should include:
...

Descriptions of the facilities and the site. These descriptions should be based on traceable information and should identify the features of the facilities and the site. They should be at a level of detail that is sufficient to inform an assessment of the processes and events that might affect the performance of the facilities.

... (DS 284, para. 4.4)

– describe handling and storage activities and any other type of operations to be performed in the storage facility;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:
...

(b) A description of spent fuel handling and storage activities and any other operations at the facility;

(DS 371, para. 5.22)

– describe the expected amount and characteristics of waste or spent fuel packages or unpackaged spent fuel elements to be stored;

*Related **IAEA** safety standards:*

(DS 371, para. 5.22)

The content of a safety case for a facility may vary between Member States but the components of the safety case for a predisposal waste management facility or activity should include:

- *A description of the waste, a discussion of possible the options for management of the waste, and the rationale for the chosen / proposed waste management options. (DS 284, para. 4.4)*

– contain information on and justify the predicted lifetime of the storage facility;

*Related **IAEA** safety standards:*

... Due account shall be taken of the expected period of storage, ... (GSR Part 5, Requirement 11)

The safety case will have to justify the expected lifetime of the facility. The expected lifetime of the facility needs to be sufficient for the activity being undertaken. (DS 284, para. 6.43)

– include assessment of the safety of normal operation and during possible accident conditions in response to postulated initiating events and provide clear evidence of compliance with safety criteria and radiological limits (safety assessment);

*Related **IAEA** safety standards:*

facility specific safety case and supporting assessment should generally include aspects such as:
....

(g) Documentation of safety analyses and the safety assessment for inclusion in the documentation supporting the licensing of the facility;

....
The expected values for subcriticality, heat removal capacity and calculated radiation doses inside and at the boundary of the spent fuel storage facility;(DS 371, para. 5.22)

The content of a safety case for a facility may vary between Member States but the components of the safety case for a predisposal waste management facility or activity should include:
...
A safety assessment that provides assurance to the regulatory body and other interested parties

that operations will be conducted safely and that safety requirements will be met. (DS 284, para. 4.4)

– describe the management system;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:

….
The management system;
…. (DS 371, para. 5.22)

To ensure the safety of predisposal radioactive waste management facilities and the fulfilment of waste acceptance criteria, management systems are to be applied to the siting, design, construction, operation, maintenance, shutdown and decommissioning of such facilities and to all aspects of processing, handling and storage of waste. Features that are important to safe operation, and that are considered in the management system, are to be identified on the basis of the safety case and the assessment of environmental impacts [2, 8, 14]. These activities are required to be supported by means of an effective management system that establishes and maintains a strong safety culture [8, 14]. (GSR Part 5 para. 3.24)

The content of a safety case for a facility may vary between Member States but the components of the safety case for a predisposal waste management facility or activity should include:

…
Descriptions of the managerial ... controls over the facilities.
... (DS 284, para. 4.4)

– describe the provisions for the management and minimization of waste produced during operation of the facility;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:
….

(u) Provisions for the management of radioactive waste and for decommissioning.

. (DS 371, para. 5.22)

– contain descriptions of commissioning programme and assessment of its results including justification of any non-compliances;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:
….

(g) The commissioning programme;

…. (DS 371, para. 5.22)

- define an appropriate program for demonstrating the continuing long term compliance of waste and spent fuel packages or unpackaged spent fuel stored within the acceptance criteria including the environmental conditions within the storage facility;

  *Related **IAEA** safety standards:*

  ... For long term storage in particular, measures shall be taken to prevent the degradation of the waste containment. (GSR Part 5, Requirement 11)

  A facility specific safety case and supporting assessment should generally include aspects such as:
  ....

  (k) Monitoring programmes, including a programme for shielding verification, a programme for surveillance of the condition of stored spent fuel and a programme for surveillance of stored spent fuel assemblies, if appropriate;

  .... (DS 371, para. 5.22)

  Because long-term storage is an interim measure, the safety case should describe the provisions for the regular monitoring ... of the waste and the storage facility to ensure their continued integrity over the anticipated lifetime of the facility. (DS 284, para. 6.56)

- contain operational documentation such as:

  - operational limits and conditions for safe operation of the storage facility and their technical bases, and waste and spent fuel packages or unpackaged spent fuel acceptance criteria;

    *Related **IAEA** safety standards:*

    A facility specific safety case and supporting assessment should generally include aspects such as:
    ....

    (f) Establishment of operational limits, conditions and administrative controls based on the safety assessment. If necessary, the design of the spent fuel storage facility should be modified and the safety assessment should be updated. Such controls should include acceptance criteria for spent fuel casks, including canisters containing failed fuel;

    ..... (DS 371, para. 5.22)

    Waste packages and unpackaged waste that are accepted for processing, storage and/or disposal shall conform to criteria that are consistent with the safety case. (GSR Part 5, Requirement 12).

    Predisposal radioactive waste management facilities shall be operated in accordance with ... the conditions imposed by the regulatory body. (GSR Part 5, Requirement 19).

- procedures and operational manuals for activities with significant safety implications

  *Related **IAEA** safety standards:*

  A facility specific safety case and supporting assessment should generally include aspects such as:
  …

  (j) Procedures and operational manuals for activities with significant safety implications;

  ….. (DS 371, para. 5.22)

  Operations shall be based on documented procedures. ...(GSR Part 5, Requirement 19).

- the operational inspection, maintenance and testing provisions,

  *Related **IAEA** safety standards:*

  A facility specific safety case and supporting assessment should generally include aspects such as:
  ….

  (g) Organizational control of operations;

  (k) A programme for periodic maintenance, inspection and testing; (DS 371, para. 5.22)

  Waste shall be stored in such a manner that it can be inspected ... (GSR Part 5, Requirement 11)

  ... Due consideration shall be given to the maintenance of the facility to ensure its safe performance. ...
  (GSR Part 5, Requirement 19).

  Because long-term storage is an interim measure, the safety case should describe the provisions for the regular ... inspection and maintenance of the waste and the storage facility to ensure their continued integrity over the anticipated lifetime of the facility. (DS 284, para. 6.56)

- the operational experience feedback programme,

  *Related **IAEA** safety standards*

  A facility specific safety case and supporting assessment should generally include aspects such as:
  ….

  (n) A programme for feedback of operational experience;

  …. (DS 371, para. 5.22)

- the programme for management of ageing;

  *Related **IAEA** safety standards*

  For storage beyond the original design lifetime, a re-evaluation of the initial design (and of the current design if it is significantly different), operations, maintenance, ageing management, safety assessment and any other aspect of the spent fuel storage facility relating to safety should be performed. […]
  (DS 371, para.5.29)

– describe the arrangements for qualification and training of personnel;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:

….

(o) The training programme for staff;

…. (DS 371, para. 5.22)

– describe the emergency preparedness arrangements at least at the level of on-site emergency plan;

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:

- *The emergency preparedness and response plan;*
  …. (DS 371, para. 5.22)
- *Emergency preparedness and response plans, if developed by the operator, are subject to the approval of the regulatory body (GSR Part 5, Requirement 19)*

– include the site strategy for decommissioning and the (initial) decommissioning plan[3];

*Related **IAEA** safety standards:*

A facility specific safety case and supporting assessment should generally include aspects such as:
….

(u) Provisions for the management of radioactive waste and for decommissioning.

(DS 371, para. 5.22)

The operator shall develop, in the design stage, an initial plan for the shutdown and decommissioning of predisposal radioactive waste management facilities and shall periodically update it throughout the operational period. (GSR Part 5, Requirement 20).

---

[3] Further details on the structure and content of decommissioning plan are covered by WGWD document "Decommissioning Safety Reference Levels Report"

The content of a safety case for a facility may vary between Member States but the components of the safety case for a predisposal waste management facility or activity should include:

- *...*
- *Plans regarding the ... decommissioning of the facilities. ...*
- *... (DS 284, para. 4.4)*

Part 3
Benchmarking, SRL-update and action Plans

-

Part 3 of the Storage Report provides information on

- the process of benchmarking, i.e. the verification of the application of storage SRLs in WENRA member countries using a systematic appraisal procedure in the working group,
- updating some of the SRLs in light of experience from the benchmarking procedure
- the national action plans (NAPs), working documents to support carrying out corrective actions whenever any deficiencies had been identified in the previous benchmarking process.
- The WENRA approval procedure for such corrective actions.

It has to be highlighted that a first set of 77 storage SRLs (the so called version 1, V.1 SRLs) had been drafted, which was never approved by WENRA-directors. However this set was the basis for the initial benchmarking procedure as described in the following chapters 3.1 and 3.2. Parallel to working on the NAPs and on the basis of experience from the benchmarking procedure WGWD redrafted the storage report. The resulting 61 storage SRLs (referred to as V.2 SRLs) as described in this report have been approved by WENRA directors and are referenced throughout this report except for the previously mentioned chapters 3.1 and 3. The relations between V.1 and V.2 storage SRLs are explained in some detail in chapter3.3.

3.1
Benchmarking of original storage SRLs (V1)

-

The Benchmarking process compassed two main steps of evaluation. In the first step all participating countries performed a self-assessment of their national regulatory system with regard to the WENRA safety reference levels. In accordance with WENRA's Reactor Harmonization Working Group (RHWG), a code of three degrees for evaluation has been applied:

A – The requirement is covered explicitly by national regulatory system: no action required
B – A difference exists, but can be justified from the safety point of view: no action required
C – A difference exists and should be addressed for harmonization in the national action plan.

For the self-assessment, each country had to perform the rating level by level and to justify the proposed rating by quoting the relevant text sections from the corresponding national regulation in an evaluation table.

In the second step of the benchmarking, the results of the self-assessment were reviewed by other countries. Four sub-groups have been created from the seventeen participating member countries in order to review the rating and justifications within the groups. Each country had to justify its self-assessment to the members of the review group. In the sub-group sessions, the self-assessment of the group members were reviewed in detail and up- or downgraded if appropriate. The group sessions took place during the WGWD meetings, starting at the 18th meeting in Budapest end of May 2007 and formally ending at the 22nd meeting in Brussels in April 2009. So the legal benchmark results reflect the regulatory state of the participating countries at the year 2007.

Due to partially different levels of requirements for spent fuel and low or medium level waste, a separate benchmarking was performed for each of these two categories of storage facilities. The evaluation process outlined here above in shortness is referred to as **legal benchmarking**.

In accordance with the RHWG approach a further step of benchmarking has been added, addressing the implementation of the safety reference levels in existing facilities. This part is referred to as **implementation benchmarking**. For the implementation benchmarking, selected existing facilities underwent the same benchmarking procedure as described above, consisting of self-assessment and review through peer review groups. The objective was to evaluate the degree of compliance with the WENRA SRLs in selected operating storage facilities.

The countries were asked to propose, if possible, facilities which are representative with regard to capacity, safety level and operation time.

3.2
Benchmarking Results

-

The summary of results presented in the following tables and figures is based on the summary tables, which were prepared by the secretaries of the sub-groups. The tables 1a and 2a give an overview of the legal benchmarking results by country and SRL for spent fuel and radioactive waste respectively. The results of the implementation benchmarking are presented in table 1b and 2b for each country. If several facilities have been subjected to the implementation benchmarking, the results are presented in separate columns for each facility. For the implementation benchmark 20 facilities for spent fuel storage and 24 facilities for radioactive waste storage have been evaluated in total. The rating is represented by the colors green for A, blue for B and red for C. For the implementation benchmark additionally the 'NA' (not applicable) was accepted in cases, where a requirement was not adequate or obsolete, as for example criticality safety measures in a storage facility for low level waste excluding fissile material.

Some SRLs which are subject to revision are marked by brown color.

For the legal benchmarking the 'NA' rating was excluded and in addition the evaluating sub groups were instructed not to make extensive use of the B rating. Only in cases where neither C nor A seemed adequate, a B rating was recommended. In total 8 B ratings were assigned for spent fuel storage and 15 B ratings were assigned for radioactive waste storage. Some examples of B ratings are given in table 3 below.

The total number of ratings is 77, in compliance with the number of SRLs. The number of participating countries is 17, while for the implementation benchmarking 20 facilities for spent fuel storage and 24 facilities for radioactive waste storage have been evaluated.

Finally the figures 1 and 2 of this section provide compilations of the legal benchmarking results with regard to the C ratings. Each column corresponds to one safety issue, which comprises several SRLs. The height of the column represents the number of countries, which have at least one (or more) C ratings for the respective safety issue. As the number of participating countries is 17, one can see from the figures that for the safety issues "Storage facility design" (S-19 to S-34) and "Maintenance, in-service inspection and functional testing" (S-50 to S-58) all countries received at least one C rating for spent fuel storage and radioactive waste storage as well. It emphasized again, that the benchmarking results presented here are reflecting the legal and licensing status as of the year 2007.

Legal Benchmark		V1.1	Countries sorted by V1.1 report order																
Spent Fuel Storage		SRL	C1	C2	C3	C4	C5	C6	C7	C8	C9	C10	C11	C12	C13	C14	C15	C16	C17
Safety Management	Respon-sibility	1																	
		2																	
		3																	
		4																	
		5																	
		6																	
		7																	
	Organi-sational Structure	8																	
		9																	
		10																	
	Quality Manage-ment	11																	
		12																	
		13																	
		14																	
	Record keeping	15																	
		16																	
		17																	
		18																	
Design	Storage Facility Design	19																	
		20																	
		21																	
		22																	
		23																	
		24																	
		25																	
		26																	
		27																	
		28																	
		29																	
		30																	
		31																	
		32																	
		33																	
		34																	
	Handling and Retrieval	35																	
		36																	
		37																	
		38																	
	Storage Cap.	39																	
Operation	Operation	40																	
		41																	
	Emergency Prepared-ness	42																	
		43																	
		44																	
	Operational Expe-rience Feedback	45																	
		46																	
	Operation Facility Modification	47																	
		48																	
		49																	
	Maintenance, In-Service-Inspection and Functional Testing	50																	
		51																	
		52																	
		53																	
		54																	
		55																	
		56																	
		57																	
		58																	
	Specific Contingency Plans	59																	
		60																	
		61																	
	Waste/Spent Fuel Acceptance	62																	
		63																	
		64																	
		65																	
Safety Verification	Contents and Updating of the Safety Case	66																	
		67																	
		68																	
		69																	
		70																	
	Periodic Safety Review	71																	
		72																	
		73																	
		74																	
		75																	
		76																	
		77																	
Sum		A	38	49	58	54	70	62	49	50	61	14	48	48	24	19	51	34	18
		B	1	4	2	0	0	0	0	0	1	0	0	0	0	0	0	0	0
		C	38	24	17	23	7	15	28	27	15	63	29	29	53	58	26	43	59

Table 1a: Legal Benchmark Results for Spent Fuel Storage by Countries

WENRA WGWD

Implem. Benchmark		V1.1	Storage facilities sorted by V1.1 report order																			
Spent Fuel Storage		SRL	I1	I2	I3	I4	I5	I6	I7	I8	I9	I10	I11	I12	I13	I14	I15	I16	I17	I18	I19	I20
Safety Management	Responsibility	1																				
		2																				
		3																				
		4																				
		5													NA	NA						
		6																				
		7																				
	Organisational Structure	8																				
		9																				
		10																				
	Quality Management	11																				
		12																				
		13																				
		14																				
	Record keeping	15																				
		16																				
		17																				
		18																				
Design	Storage Facility Design	19																				
		20																				
		21																				
		22																				
		23																				
		24																				
		25																				
		26																				
		27																				
		28												NA								
		29																				
		30												NA								
		31												NA								
		32																				
		33																				
		34																				
	Handling and Retrieval	35																				
		36																				
		37																				
		38												NA								
	Storage Cap.	39																				
Operation	Operation	40																				
		41																				
	Emergency Preparedness	42																				
		43																				
		44																				
	Operational Experience Feedback	45																				
		46																				
	Operation Facility Modification	47																				
		48																				
		49																				
	Maintenance, In-Service-Inspection and Functional Testing	50																				
		51																				
		52																				
		53																				
		54																				
		55																				
		56												NA	NA	NA						
		57																				
		58																				
	Specific Contingency Plans	59																				
		60																				
		61																				
	Waste/Spent Fuel Acceptance	62																				
		63																				
		64																				
		65																				
Safety Verification	Contents and Updating of the Safety Case	66																				
		67																				
		68																				
		69																				
		70																				
	Periodic Safety Review	71																				
		72																				
		73																				
		74																				
		75																				
		76																				
		77																				
Sum		A	39	71	68	67	68	64	66	69	69	67	66	57	65	64	70	73	68	64	62	49
		B+NA	7	4	4	10	9	13	7	4	2	3	4	7	4	2	0	0	0	0	1	0
		C	31	2	5	0	0	0	4	4	6	7	7	13	8	11	7	4	9	13	14	28

Table 1b: Implementation Benchmark Results for Spent Fuel Storage by Facilities

Legal Benchmark		V1.1	Countries sorted by V1.1 report order																
Waste		SRL	C1	C2	C3	C4	C5	C6	C7	C8	C9	C10	C11	C12	C13	C14	C15	C16	C17
Safety Management	Respon-sibility	1																	
		2																	
		3																	
		4																	
		5																	
		6																	
		7																	
	Organi-sational Structure	8																	
		9																	
		10																	
	Quality Manage-ment	11																	
		12																	
		13																	
		14																	
	Record keeping	15																	
		16																	
		17																	
		18																	
Design	Storage Facility Design	19																	
		20																	
		21																	
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		30																	
		31																	
		32																	
		33																	
		34																	
	Handling and Retrieval	35																	
		36																	
		37																	
		38																	
	Storage Capacity	39																	
Operation	Operation	40																	
		41																	
	Emergency Preparedness	42																	
		43																	
		44																	
	Operational Expe-rience Feedback	45																	
		46																	
	Operation Facility Modification	47																	
		48																	
		49																	
	Maintenance, In-Service-Inspection and Functional Testing	50																	
		51																	
		52																	
		53																	
		54																	
		55																	
		56																	
		57																	
		58																	
	Specific Contingency Plans	59																	
		60																	
		61																	
	Waste Acceptance	62																	
		63																	
		64																	
		65																	
Safety Verification	Contents and Updating of the Safety Case	66																	
		67																	
		68																	
		69																	
		70																	
	Periodic Safety Review	71																	
		72																	
		73																	
		74																	
		75																	
		76																	
		77																	
Sum		A	37	43	53	48	70	63	46	50	65	10	47	48	25	19	16	34	11
		B	1	5	2	1	0	0	0	0	1	0	0	0	1	1	2	1	0
		C	39	29	22	28	7	14	31	27	11	67	30	29	51	57	59	42	66

Table 2a: Legal Benchmark Results for Radioactive Waste Storage by Countries

Implem. Benchmark		V1.1	Storage facilities sorted by V1.1 report order																							
Waste		SRL	I1	I2	I3	I4	I5	I6	I7	I8	I9	I10	I11	I12	I13	I14	I15	I16	I17	I18	I19	I20	I21	I22	I23	I24
Safety Management	Respon-sibility	1																								
		2																								
		3																								
		4																								
		5												NA						NA	NA					
		6																								
		7																								
	Organi-sational Structure	8																								
		9																								
		10																								
	Quality Manage-ment	11																								
		12																								
		13																								
		14																								
	Record keeping	15																								
		16																								
		17																								
		18																								
Design	Storage Facility Design	19																								
		20																								
		21																								
		22																	NA							
		23																	NA							
		24																	NA							
		25																								
		26																								
		27																								
		28																		NA						
		29																								
		30												NA	NA	NA	NA	NA	NA		NA			NA		NA
		31												NA	NA	NA	NA	NA	NA		NA	NA		NA		NA
		32																								
		33												NA	NA	NA	NA	NA	NA	NA	NA	NA		NA		NA
		34																						NA		
	Handling and Retrieval	35																						NA		
		36																						NA		
		37																								
		38																						NA		
	Storage Capacity	39																						NA		
Operation	Operation	40																								
		41																								
	Emergency Preparedness	42																								
		43												NA	NA	NA	NA	NA								
		44												NA	NA	NA	NA	NA								
	Operational Expe-rience Feedback	45																								
		46																								
	Operation Facility Modification	47																								
		48																								
		49																								
	Maintenance, In-Service-Inspection and Functional Testing	50																								
		51																								
		52																						NA		
		53																								
		54																								
		55																								
		56															NA		NA	NA	NA					
		57																						NA		
		58															NA		NA					NA		
	Specific Contingency Plans	59																						NA		
		60																						NA		
		61															NA							NA		
	Waste Acceptance	62																								
		63																								
		64																								
		65																						NA		
Safety Verification	Contents and Updating of the Safety Case	66																								
		67																						NA		
		68																								
		69																								
		70																						NA		
	Periodic Safety Review	71															NA							NA		
		72															NA									
		73															NA									
		74															NA									
		75															NA									
		76															NA									
		77															NA									
Sum		A	68	71	63	69	72	63	62	71	73	60	60	69	56	58	61	55	45	63	50	50	71	54	62	38
		B+NA	3	2	12	4	4	9	10	5	3	7	10	8	21	7	16	22	10	7	5	3	1	18	1	3
		C	6	4	2	4	1	5	5	1	1	10	7	0	0	12	0	0	22	7	22	24	5	5	14	36

Table 2b: Implementation Benchmark Results for Radioactive Waste Storage by Facilities

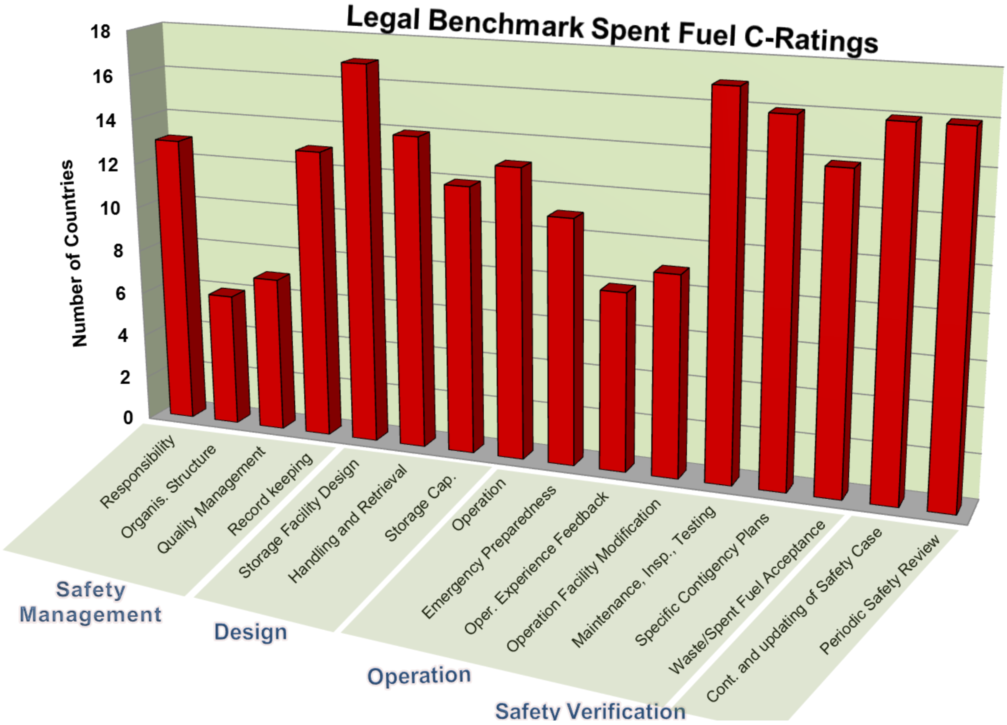

Figure 1: Number of countries with C-ratings sorted by safety issues for spent fuel storage

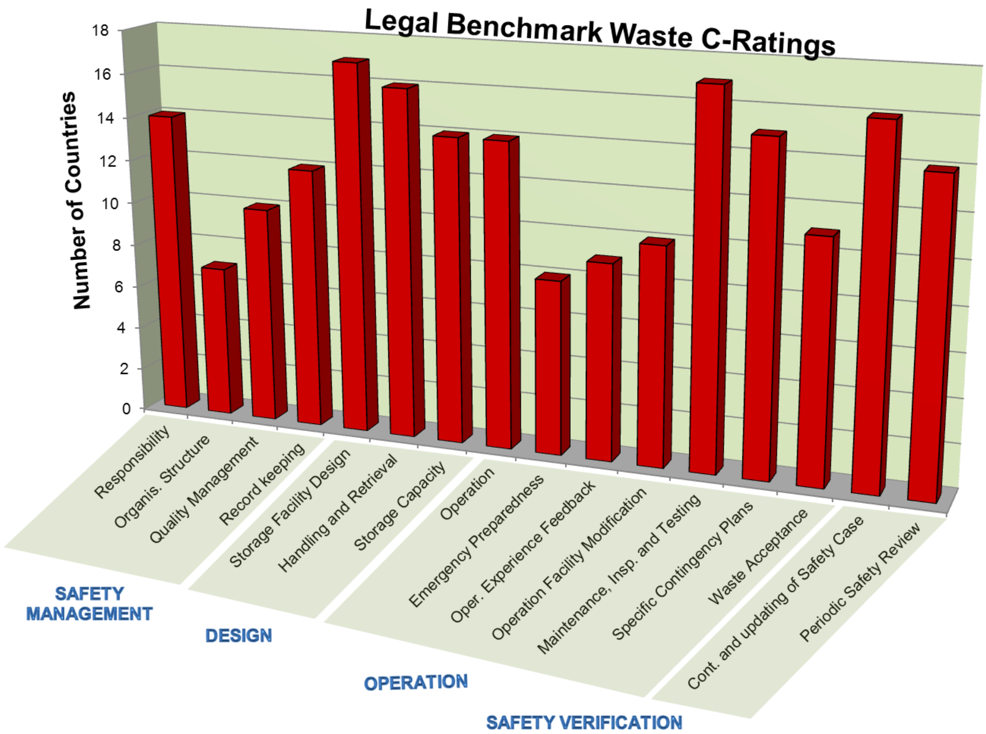

Figure 2: Number of countries with C-ratings sorted by safety issues for radioactive waste storage

3.3 Preparation of National Action Plans, SRL update

-

After final conclusion of the regulatory benchmarking procedure in 2009, the WGWD members were requested in accordance with the approach of the RHWG to develop and present national actions plans (NAPs) of their countries, in order to demonstrate the planned activities and efforts for harmonizing their national regulations with the WENRA safety reference levels (SRLs). The need for harmonization was derived from the results of the legal benchmarking for each country, where existing differences of the national regulations with respect to the WENRA SRLs have been identified.

The NAP initially had to provide information on planned modification and amendments of relevant national regulations. It had to be treated as a 'living document' and be improved and completed stepwise in line with ongoing process for harmonization of the national regulations. Finally, it provides a document that demonstrates the respective national regulation being in line with the WENRA SRLs. In accordance with the regulatory benchmarking, the NAP had to cover two areas of radioactive material storage: spent fuel and low or medium level waste. This activity was initiated by the WGWD chairman at the 22nd meeting in Brussels in April 2009 and had to be performed in parallel to other tasks of the WGWD. At the following meetings, the country representatives regularly gave short oral reports on the status and progress of their NAPs.

The deadline for implementation of NAP-actions had originally been set to end of 2012 but was in later decision of WENRA directors prolonged until end of 2013. This prolongation was deemed necessary because the requirements in the original draft set of 77 storage V.1-SRLs had been reworded and rearranged resulting in a finally approved set of only 61 storage V.2-SRLs. It is to be emphasized that in doing so no requirement of the original V.1-SRLs has been lost. In some cases, however, the degree of detail was adjusted to the general and binding character of WENRA-SRLs. Furthermore the new V.2-SRLs took into account the most recent developments in IAEA publications especially the modified approach to quality ("management system" approach replaced "quality management" / "quality control" / "quality assurance").

Before taking any action, obviously the results of the benchmarking exercise, which referred to the V.1-SRLs, had first to be related to the updated V.2-set of SRLs. To support member countries in this translation procedure WGWD prepared the following cross reference table indicating the relation between old and new SRLs and providing information on changes of the addressed requirements.

Cross Reference Table for WGWD-reports 1.0 and 2.0, based on short descriptions

SRL V1.0	Requirement (short description)	SRL V2.0	SRL V1.0	Requirement (short description)	SRL V2.0
	* Relevant requirement changes in V2.0			* Relevant requirement changes in V2.0	
S-01	Responsibilities	S-01	S-40	Ability to inspect packages (operation)	S-35
S-02	Prime responsibility, Safety policy	S-02	S-41	Reserve storage capacity	S-36
S-03	Maintaining and improving safety	S-03	S-42	Emergency plan, organisation	S-37, S-38
S-04	Ownership	S-04	S-43	Emergency plan content [11 items]	S-38
S-05	Safety responsibilities of waste owner	S-05	S-44	Emergency plan: Review and training	S-39
S-06	Interface between licensee and owner	S-06	S-45	Operation Experience Feedback (OEF)	S-40
S-07	Information to regulatory authority	S-07	S-46	Improvement measures from OEF	S-41
S-08	Organisational structure	S-08	S-47	Process evaluating safety impact of changes	S-42
S-09	Licensee's capabilities	S-09	S-48	Modification of storage conditions	S-42
S-10	Defining qualification and experience	S-10	S-49	Training and documentation for modifications	S-43
S-11	Quality management system	S-11	S-50	Maintenance, testing and inspection (MTI)	S-44
S-12	QM in the design phase	S-12	S-51	MTI program and procedures	S-45
S-13	Quality of safety related work	S-13	S-52	MTI frequency	-
S-14	Procurement and quality	-	S-53	MTI equipment and items	-
-	Documentation of the management system	S-14	S-54	MTI recording and assessment	S-46
S-15	Waste record system	S-15	S-55	MTI review	S-47
S-16	Identification of packages	S-16	S-56	MTI cross-effects awareness	-
S-17	Inventory information system	S-17	S-57	Verification of compliance with safety case	S-48
S-18	Record update and availability	S-18	S-58	Program for inspection and maintenance	S-44
S-19	Design and safety functions	S-19	S-59	Receipt procedure for failed packages	S-49
S-20	Design for the lifetime of the facility	S-20	S-60	Plan for loss of integrity or degradation	-
S-21	Designing for passive safety features	S-21	S-61	Contingency arrangements	S-50
S-22	Construction standards	S-22	S-62	Package design requirements	S-51
S-23	Design basis	S-23	S-63	Acceptance criteria	S-52
S-24	SSC identification	S-24	S-64	Compatibility with conditions and safety case	S-53
S-25	Ageing of SSCs and safety features	S-25	S-65	Auditing, inspection and testing on reception	S-54
S-26	Establishing operational limits and conditions	S-26	S-66	Safety case and application	S-55
S-27	OLC conditions (5 items)	S-27	S-67	Safety case content [17 items]	-
S-28	Storage limits	-	-	Use of the safety case for assessing changes	S-56
S-29	List of Probable Initiating Events (PIE)	S-28	S-68	Safety case for facility and packages	S-57
S-30	Prevention of criticality accidents	S-29	S-69	Update of the safety case	S-58
S-31	Criticality prevention by design	S-29	S-70	Conditions to revise the safety case	S-58
S-32	Release prevention	-	S-71	Periodic Safety Review (PSR)	S-59
S-33	Design for fire safety and DiD	S-30	S-72	PSR and changes or modifications	S-60
S-34	Design for handling equipment	S-31	S-73	PSR and improvement measures	S-61
S-35	Design for package retrieval	S-32	S-74	PSR and safety case update	S-58
S-36	Retrievability of packages	-	S-75	PSR frequency	S-59
S-37	Ability to inspect packages (Design)	S-33	S-76	PSR scope and methodology	S-60
S-38	Equipment for handling degraded packages	-	S-77	PSR and deviations/interdependancies	S-60
S-39	Reserve capacity	S-34			

Table 3: Cross Reference Table for WGWD Reports 1.0 and 2.0 based on short descriptions

3.4
Benchmarking of the National Action Plans

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As previously explained it was understood that all agreed C-ratings in the individual national regulatory systems would require actions in order to reach full compatibility with the set of WENRA-SRLs. The whole procedure included the following successive steps:

1. Preparation of comprehensive list of C-ratings
2. For each C-rated SRL of V.1:
   a. Find corresponding new SRL of V.2
   b. Use the new V.2 text of this SRL for updating national regulation
3. Follow step 2b also for any SRLs which had been identified as “unclear” and any SRL with relevant requirement changes in the transformation procedure from V.1 to V.2.
4. Supply reference for actions as carried out and report to WGWD

The final objective of the NAPs was to provide the necessary arguments to WGWD that missing requirements had been fully included in each country’s national regulatory system. For the final approval a second benchmarking exercise was carried out, specifically concentrating on those NAPs which were claimed to be finally concluded. For this review process WGWD used the same techniques as for the original legal benchmarking, sometimes working in the plenary and sometimes in up to four sub-groups, as appropriate. An improved template for the NAP has been developed to facilitate handling of the documents for the group benchmarking. After introducing the new template at the 27th meeting in Ljubljana in October 2011, the first group benchmarking of NAPs took place at the 29th meeting in Stockholm on 25 – 27 September 2012. In total nine NAPs fully or partially ready for benchmarking had been submitted in advance of this meeting. After plenary discussion and agreement on the rules for evaluation, the benchmarking was performed in two subgroups. Further evaluations of NAPs have been done in the following WGWD meetings until the 32nd in Rome in February 2014. Some countries were not able to fulfill their NAP within this time frame, in particular because they choose time-consuming parliamentary procedures, which could not yet be completed.

3.5 Country Implementation Reports

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In this section the results of the NAP benchmarking are presented for each country in two parts. The first part (text) consists of a short description on the measures taken for fulfillment of the NAP, provided by each country. The second part is a table, which lists in the first column the SRLs for which differences had been identified initially, whereas the second and third column show the status of harmonization for spent fuel and waste. An A-Rating in the second column indicates that the required harmonization has been implemented in national regulations and was agreed by the WGWD. For countries, whose NAP benchmarking procedure could not yet be concluded by the WGWD at least at their meeting in Feb. 2014 information as provided by the respective country representative is presented.

BELGIUM

Regulatory changes taken for the National Action Plan

In Belgium, most of the WENRA Waste and Spent Fuel Storage Safety Reference Levels are covered by the generic chapter 2 of the Royal Decree "Safety requirements for nuclear installations", published on November 30th, 2011. This chapter 2 includes the WENRA Reactor Safety Reference Levels that Belgium considered to be applicable to all its major nuclear installations (class I installations), which includes Waste and Spent Fuel Storage installations.

To comply with the remaining Waste and Spent Fuel Storage Safety Reference Levels, a new chapter of this Royal Decree was drafted. Its publication is expected before the end of 2014.

At the 29 WGDW meeting in Stockholm in September 2012, Belgium reported its (planned) regulatory implementations for benchmarking. All (proposed) changes were endorsed by the WENRA WGWD, so once the Royal Decree is published, Belgian regulations will be in full agreement with the requirements mandated by the WGWD SRLs.

Results of the NAP Benchmarking (Belgium)

Spent Fuel Storage and Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-03	A	Royal Decree "Safety requirements for nuclear installations", Ch. 2, Art. 4.2
S-07	A	(not relevant in Belgium)
S-09	A	Royal Decree, Ch. 2, Art. 4.3
S-10	A	Royal Decree, Ch. 2, Art. 4.3
S-11	A	Royal Decree, Ch. 2, Art. 5.1
S-12	A	Royal Decree, Ch. 2, Art. 5.1
S-13	A	Royal Decree, Ch. 2, Art. 5.5
S-14	A	Royal Decree, Ch. 2, Art. 5.2
S-16	C	Royal Decree, Ch. 5, Art. 53 (not yet published)
S-19	C	Royal Decree, Ch. 5, Art. 50 (not yet published)
S-20	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-21	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-22	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-24	A	Royal Decree, Ch. 2, Art. 8
S-25	A	Royal Decree, Ch. 2, Art. 10
S-26	A	Royal Decree, Ch. 2, Art. 9.1
S-27	C	Royal Decree, Ch. 5, Art. 52 (not yet published)
S-31	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-32	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-33	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-34	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-35	C	Royal Decree, Ch. 5, Art. 53 (not yet published)
S-36	C	Royal Decree, Ch. 5, Art. 51 (not yet published)
S-37	A	Royal Decree, Ch. 2, Art. 16
S-40	A	Royal Decree, Ch. 2, Art. 11
S-41	A	Royal Decree, Ch. 2, Art. 11
S-42	A	Royal Decree, Ch. 2, Art. 15
S-43	A	Royal Decree, Ch. 2, Art. 15
S-44	A	Royal Decree, Ch. 2, Art. 12
S-45	A	Royal Decree, Ch. 2, Art. 12
S-46	A	Royal Decree, Ch. 2, Art. 12
S-47	A	Royal Decree, Ch. 2, Art. 11
S-48	C	Royal Decree, Ch. 5, Art. 55 (not yet published)
S-49	C	Royal Decree, Ch. 5, Art. 56 (not yet published)
S-50	C	Royal Decree, Ch. 5, Art. 56 (not yet published)
S-51	C	Royal Decree, Ch. 5, Art. 54 (not yet published)
S-52	C	Royal Decree, Ch. 5, Art. 54 (not yet published)
S-53	C	Royal Decree, Ch. 5, Art. 54 (not yet published)
S-54	C	Royal Decree, Ch. 5, Art. 54 (not yet published)
S-55	A	Royal Decree, Ch. 2, Art. 13
S-56	A	Royal Decree, Ch. 2, Art. 13
S-57	C	Royal Decree, Ch. 2, Art. 13; Royal Decree, Ch. 5, Art. 54 (not yet published)
S-58	A	Royal Decree, Ch. 2, Art. 13
S-59	C	Royal Decree, Ch. 2, Art. 14; Royal Decree, Ch. 5, Art. 58 (not yet published)
S-60	C	Royal Decree, Ch. 2, Art. 14; Royal Decree, Ch. 5, Art. 58 (not yet published)
S-61	A	Royal Decree, Ch. 2, Art. 14; Royal Decree, Ch. 5, Art. 58 (not yet published)

BULGARIA

Regulatory changes taken for the National Action Plan

In accordance with the National Action Plan, approved by an order of the chairman of the BNRA, revision and analysis of the compliance with the requirements of the Act on the Safe Use of Nuclear Energy, new IAEA documents and the WGWD Safety Reference Levels (SRLs) have been carried out.

During the amendment of the legislative documents, the taking into account of the developed by the WGWD SRLs and the harmonization of the Bulgarian legislation with that of the European countries were of primary importance, with the objective of achieving a common approach in the management of RAW and SF.

At the time of the 29th WGWD meeting in Stockholm and at the 30th WGWD meeting in Prague, Bulgaria reported the incorporation of the SRLs (rated C in the NAP) in the draft of the Regulation on Safety of RAW Management.

The amendments of the regulations were related with the establishment of a management system, qualification of the personnel, ensuring reserve storage capacity, etc.

Before the adoption of the regulations, in a systematic way the amendments in the documents were discussed and agreed on with the representatives of the stakeholders and other competent authorities.

In the course of amending the legislation all of the WGWD SRLs have been incorporated in the Regulation on Safety of RAW Management and the Regulation for Safety of Spent Fuel Management which have entered into force in 2013.

Results of the NAP Benchmarking (Bulgaria)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-05	A	Reg. on SF management, Art. 2a and 117
S-10	A	Reg. on SF management, Art. 117 and 73
S-11	A	Reg. on SF management, Art. 118
S-12	A	Reg. on SF management, Art. 116
S-13	A	Reg. on SF management, Art. 118
S-14	A	Reg. on SF management, Art. 120, 116, 118
S-16	A	Reg. on SF management, Art. 119
S-18	A	Reg. on SF management, Art. 119
S-36	A	Reg. on SF management, Art. 82
S-37	A	Reg. on SF management, Art. 2b and 109
S-43	A	Reg. on SF management, Art. 90 and 119
S-49	A	Reg. on SF management, Art. 78
S-50	A	Reg. on SF management, Art. 78
S-51	A	Reg. on SF management, Art. 82
S-57	A	Reg. on SF management, Art. 110
S-58	A	Reg. on SF management, Art. 110

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-05	A	Regulation on RAW management, Art. 3 and 5
S-10	A	Regulation on RAW management, Art. 40 and 48
S-11	A	Regulation on RAW management, Art. 49, 47, 50
S-12	A	Regulation on RAW management, Art. 47
S-13	A	Regulation on RAW management, Art. 49
S-14	A	Regulation on RAW management, Art. 47, 48, 51, 53
S-16	A	Regulation on RAW management, Art. 50 and 6
S-21	A	Regulation on RAW management, Art. 35
S-30	A	Regulation on RAW management, Art. 22 and 31
S-34	A	Regulation on RAW management, Art. 35
S-36	A	Regulation on RAW management, Art. 41
S-40	A	Regulation on RAW management, Art. 42
S-41	A	Regulation on RAW management, Art. 42
S-43	A	Regulation on RAW management, Art. 50
S-46	A	Regulation on RAW management, Art. 49
S-47	A	Regulation on RAW management, Art. 49
S-49	A	Regulation on RAW management, Art. 5 and 10
S-57	A	Regulation on RAW management, Art. 54
S-58	A	Regulation on RAW management, Art. 56

CZECH REPUBLIC

Regulatory changes taken for the National Action Plan

The WGWD Safety Reference Levels for waste and spent fuel storage are considered in the Czech Republic in the process of update of national legal framework. This process has already been launched in 2009 but is not finished yet. It is expected that the new Atomic Act and related decrees will enter into force in mid-2015.

Current national legal framework, especially the Act No. 18/1997 Coll. (Atomic Act) and Decree No. 307/2002 Coll. (on radiation protection) already comply with most of WGWD reference levels. However there are some non-compliances, especially related to the safety issues responsibility, management system, operation, OEF, maintenance, specific contingency plan, acceptance criteria and contents and updating of the safety case.

The new national legal framework, especially the Decree on safe radioactive waste management, which is in well advanced stage of preparation, will fully comply with the WGWD Safety Reference Levels.

Results of the NAP Benchmarking (Czech Republic)
Spent Fuel Storage SRLs:

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	C	In the draft version of new Atomic Act (V. 2)
S-05	B	In the draft versions of new Atomic Act (V. 2) and new Decree on the safe radioactive waste management (V. 3)
S-10	C	In the draft version of new Atomic Act (V. 2)
S-11	C	In the draft version of new Decree on management system
S-12	C	In the draft version of new Decree on management system
S-13	C	In the draft version of new Decree on management system
S-14	C	In the draft version of new Decree on management system
S-16	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-21	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-23	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-28	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-34	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-36	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-37	C	In the draft version of new Decree on emergency preparedness
S-40	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback
S-41	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback.
S-47	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback
S-49	C	In the draft versions of new Atomic Act and new Decree on the safe radioactive waste management
S-50	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-52	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-53	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-54	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-57	C	In current Atomic Act (18/1997 Coll.) and in the draft version of new Atomic Act (V. 2)
S-58	C	In the draft versions of new Atomic Act (V. 2) and new Decree on safety documentation.
S-60	B	In the draft versions of new Atomic Act (V. 2) and new Decree on safety documentation
S-61	B	In the draft version of new Atomic Act (V. 2)

Results of the NAP Benchmarking (Czech Republic, cont.)

Waste Storage SRLs:

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	C	In the draft version of new Atomic Act (V. 2)
S-05	B	In the draft versions of new Atomic Act (V. 2) and new Decree on the safe radioactive waste management (V. 3)
S-10	C	In the draft version of new Atomic Act (V. 2)
S-11	C	In the draft version of new Decree on management system
S-12	C	In the draft version of new Decree on management system
S-13	C	In the draft version of new Decree on management system
S-14	C	In the draft version of new Decree on management system
S-16	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-19	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-21	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-23	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-28	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-33	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-34	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-35	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-36	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-37	C	In the draft version of new Decree on emergency preparedness
S-40	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback
S-41	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback
S-47	C	In the draft versions of new Atomic Act (V. 2) and new Decree on operating experience feedback
S-48	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-49	C	In the draft versions of new Atomic Act and new Decree on the safe radioactive waste management
S-50	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-52	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-53	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-54	C	In the draft version of new Decree on the safe radioactive waste management (V. 3)
S-57	C	In current Atomic Act (18/1997 Coll.) and in the draft version of new Atomic Act (V. 2)
S-58	C	In the draft versions of new Atomic Act (V. 2) and new Decree on safety documentation
S-60	B	In the draft versions of new Atomic Act (V. 2) and new Decree on safety documentation
S-61	B	In the draft version of new Atomic Act (V. 2)

FINLAND

Regulatory changes taken for the National Action Plan

The Finnish Radiation and Nuclear Safety Authority STUK regulates use of nuclear energy in Finland and gives detailed guidance in the form of guides called YVL Guides. When the WGWD Safety Reference Levels for waste and spent fuel storage were published in February 2011, STUK had already begun a full revision of the regulatory guides. When performing the revision, the WENRA storage reference level requirements were implemented into the new Finnish regulations. The revision of the Finnish Guides was finalized in 2013 and they came into force at December 1st 2013. Storage of waste and spent fuel are discussed in distinct Guides.

Finland reported its regulatory implementations of the benchmarking based on drafts of the Guides at the 30th and 31st WGDW meetings in Prague and Trnava in 2013. The requirements presented in the new YVL Guides were approved and the Finnish regulations were found to be in full agreement with the requirements mandated by the WGWD SRLs. Only minor editorial changes were made to the Guides after the benchmarking had been presented at the WGWD meetings.

Legislation:
- NEA = Nuclear Energy Act, 11.12.1987/990
- NED = Nuclear Energy Decree, 12.2.1988/161
- GD 733 = Government Decree on the Safety of NPPs 733/2008
- GD 735 = Government Decree on Emergency Response Arrangements at NPPs 735/2008

Guidance
- YVL A.3 = Guide YVL A.3, Management systems for nuclear facilities
- YVL A.4 = Guide YVL A.4, Organisation and personnel of a nuclear facility (only in Finnish)
- YVL A.10 = Guide YVL D.3, Operating experience feedback at nuclear facilities
- YVL C.5 = Guide YVL C.5, Emergency preparedness arrangements of a NPP
- YVL D.3 = Guide YVL D.3, Handling and storage of nuclear fuel
- YVL D.4 = Guide YVL D.4, Handling of low- and intermediate-level nuclear waste and decommissioning of a nucl. facility

Results of the NAP Benchmarking (Finland)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	GD 733, 29 §, Guide YVL A.3
S-08	A	GD 733, 30 §, Guide YVL A.3
S-09	A	NEA 7i §, GD 733, 30 §, 29 §, Guide YVL A.3
S-10	A	GD 733, 30 §
S-11	A	GD 733, 29 §, 28 §, Guide YVL A.3
S-12	A	GD 733 29 §, 30 §, Guide YVL A.3
S-13	A	GD 733, 29 §, Guide YVL A.3
S-14	A	Guide YVL A.3
S-16	A	Guide YVL D.3
S-18	A	Guide YVL D.3
S-37	A	NEA 7 and 9 §, GD 735 6 and 12 §, Guide YVL C.5
S-43	A	GD 733, 30 §, Guide A.4
S-48	A	Guide YVL D.3
S-49	A	Guide YVL D.3
S-52	A	Guide YVL D.3
S-53	A	Guide YVL D.3
S-54	A	Guide YVL D.3
S-57	A	Guide YVL D.3, D.4
S-58	A	Guide YVL D.3

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	GD 733, 29 §, Guide YVL A.3
S-08	A	GD 733, 30 §, Guide YVL A.3
S-09	A	NEA 7i §, GD 733, 30 §, 29 §, Guide YVL A.3
S-10	A	GD 733, 30 §
S-11	A	GD 733, 29 §, 28 §, Guide YVL A.3
S-12	A	GD 733 29 §, 30 §, Guide YVL A.3
S-13	A	GD 733, 29 §, Guide YVL A.3
S-14	A	Guide YVL A.3
S-34	A	NEA 7h §, Guide YVL D.4
S-36	A	Guide YVL D.4
S-37	A	NEA 7 and 9 §, GD 735 6 and 12 §, Guide YVL C.5
S-43	A	GD 733, 30 §, Guide A.4
S-46	A	Guide YVL D.4, Guide YVL A.10
S-48	A	Guide YVL D.4
S-49	A	Guide YVL D.4
S-50	A	Guide YVL D.4
S-52	A	Guide YVL D.4
S-53	A	Guide YVL D.4
S-54	A	Guide YVL D.4
S-57	A	Guide YVL D.4
S-58	A	Guide YVL D.4,

FRANCE

Regulatory changes taken for the National Action Plan

Since the publication of the WGWD Safety Reference Levels for waste and spent fuel storage in February 2011, France has continued to fulfill its obligations to implement necessary changes into its national regulations. In 2012, the ministerial order of 7th February setting general rules relative to basic nuclear installations was published and entered into effect on 1st July 2013. This order – which follows the TSN act of 2006 – enabled an important update of the French regulatory framework that used to rely mainly on the quality order of 1984 and on the environment order of 1999. This order also permits to transpose directly a number of important safety reference levels identified by WENRA, such as those concerning the safety policy, the integrated management system or the safety verification. Additionally, this ministerial order contained a dedicated title on waste management and specific requirements for storage and disposal facilities. However, this ministerial order sets generic requirements that have to be further developed in decisions to be issued by ASN and then approved by the Minister for nuclear safety, to give them a regulatory status. Thus, several decisions are under writing by ASN and among them decisions on waste management, decommissioning, storage facilities, periodic safety review, integrated management system, etc. The validation process includes different steps of consultation of stakeholders. Some of these decisions have already been published (e.g. ASN Resolution of 16th July 2013 relative to control of nuisance effects and the impact of basic nuclear installations on health and the environment) but others won't be fully approved before the end of year 2014 or 2015.

At the 30 WGWD meeting in Prague in February 2013, France reported its regulatory implementations for benchmarking, relying on dispositions of the Ministerial order of 7th February and on early drafts of the decisions under validation. This benchmarking enabled France to check that its obligations will be fulfilled once these decisions are finally approved.

Results of the NAP Benchmarking (France)

Spent Fuel and Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	Ministerial order of 7th Feb. 12 - article 2.3,1
S-03	A	MO of 7/2/12 - article 2.3,1 and chapter 2.7 "continual improvement" : articles 2.7.1 to 2.7.3
S-05	A	Waste act - MO of 7/2/12 - article 7.2 + ASN decision on "waste management in nuclear facilities" under finalization
S-06	A	MO of 7/2/12 - article 8.4.3
S-07	A	L,594 of the environnement code
S-10	A	MO of 7/2/12 - article 2.5.5
S-11	A	MO of 7/2/12 - article 2.4.1 and 2.4.2
S-12	A	MO of 7/2/12 - article 2.4.1
S-13	A	MO of 7/2/12 - article 2.4.1 and 2.4.2
S-14	C	MO of 7/2/12 - article 2.3.2 + ASN decision on safety management under discussion
S-15	A	MO of 7/2/12 - articles 6.5 and 8.4.2
S-16	C	ASN decision (under development) on storage facilities - article 5.3.2,2
S-17	A	MO of 7/2/12 - articles 6.5
S-18	A	MO of 7/2/12 - article 6.5 + ASN decision on storage (under development) article 5.3.3.4
S-19	A	MO of 7/2/12 - article 3.4 and 3,2 + ASN decision on storage (under development) article 5.3.3.4
S-20	C	MO of 7/2/12
S-21	C	ASN decision on storage (under development)
S-22	C	ASN decision on storage (under development)
S-29	C	ASN decision on storage (under development) + another decision about criticality under development
S-31	C	ASN decision on storage (under development)
S-32	A	MO of 7/2/12 - article 8.4.2 + ASN decision on storage (under development)
S-33	A	MO of 7/2/12 - article 8.4.2 + ASN decision on storage (under development)
S-34	C	ASN decision on storage (under development)
S-35	C	MO of 7/2/12 - article 8.4.2 + ASN decision on storage (under development)
S-36	C	ASN decision on storage (under development)
S-37	A	Public Health Code + decree 2/11/2007+ MO of 7/2/12
S-40	A	MO of 7/2/12
S-43	C	MO of 7/2/12
S-46	C	MO of 7/2/12
S-47	C	MO of 7/2/12 - title 2 chapter VII
S-48	C	MO of 7/2/12 - article 8.4.2 + ASN decision on storage (under development)
S-49	C	ASN decision on storage (under development)
S-51	A	MO of 7/2/12 + ASN decision on conditioning (under development)
S-52	A	MO of 7/2/12
S-53	C	ASN decision on storage (under development)
S-57	C	ASN decision on storage (under development)
S-58	A	Ministerial decree of 2 Nov. 2007 + decision on waste storage (under development)

GERMANY

Regulatory changes taken for the National Action Plan

In parallel to the publication of the WGWD Safety Reference Levels (SRLs) for waste and spent fuel storage, version 2.1 in February 2011, Germany continued to fulfill its commitments within the WENRA WGWD to implement necessary modifications and amendments into its regulatory framework in order to harmonize the national regulations with the agreed code of SRLs. Most of the required revisions to the German regulations were related to two safety issues, namely the proper establishment of

- a safety management system and
- periodic safety reviews (PSR) of spent fuel storage facilities, including systematic ageing management.

In the German regulatory framework general issues of storage are covered in two guidelines, formulating specific recommendations and requirements for storage facilities dealing with spent fuel and radioactive wastes, respectively /ESK 13a, 13b/. Those guidelines have been drafted and finalised by the Nuclear Waste Management Commission (ESK), an independent expert committee advising the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB). After publication and formal enactment through BMUB, the guidelines have mandatory character for licensees and operators of spent fuel and waste storage facilities. The new guidelines on waste and spent fuel storage are the result of a complete review of the former guidelines (issued in 2001 and 2003, respectively) considering the results of the first regulatory benchmarking. The newly revised and upgraded versions were then published in June 2013 and include requirements to implement a safety management system according to the new formulation of the related SRLs.

In the case of PSRs, the ESK developed a set of recommendations which were brought into effect in November 2010 as a separate set of guidelines /ESK 10/. After receiving application feedback from selected facilities, an updated version of the PSR Guidelines is expected for publication by the end of 2014.

The German approach to implement the necessary changes into the national regulatory framework was presented and discussed at the 29th WGWD meeting in Stockholm in September 2012. All changes were approved by the WGWD.

/ESK 10/ ESK recommendations for guides to the performance of periodic safety reviews for storage facilities for spent fuel and heat-generating radioactive waste (PSÜ-ZL), 14.11.2010

/ESK 13a/ Guidelines for dry cask storage of spent fuel and heat-generating waste; recommendations of the Nuclear Waste Management Commission (ESK), revised version of 10.06.2013

/ESK 13b/ Guidelines for the storage of radioactive waste with negligible heat generation; recommendations of the Nuclear Waste Management Commission (ESK), revised version of 10.06.2013

Results of the NAP Benchmarking (Germany)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	Update of guidelines for SF storage (ESK 13a)
S-11	A	Update of guidelines for SF storage (ESK 13a)
S-12	A	Update of guidelines for SF storage (ESK 13a)
S-13	A	Update of guidelines for SF storage (ESK 13a)
S-14	A	Update of guidelines for SF storage (ESK 13a)
S-35	A	Update of guidelines for SF storage (ESK 13a)
S-37	A	Update of guidelines for SF storage (ESK 13a)
S-39	A	Update of guidelines for SF storage (ESK 13a)
S-47	A	Update of guidelines for SF storage (ESK 13a)
S-49	A	Update of guidelines for SF storage (ESK 13a)
S-57	A	Update of guidelines for SF storage (ESK 13a)
S-59	A	Guidelines for PSR (ESK 10)
S-60	A	Guidelines for PSR (ESK 10)
S-61	A	Guidelines for PSR (ESK 10)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	Update of guidelines for waste storage (ESK 13b)
S-11	A	Update of guidelines for waste storage (ESK 13b)
S-12	A	Update of guidelines for waste storage (ESK 13b)
S-13	A	Update of guidelines for waste storage (ESK 13b)
S-14	A	Update of guidelines for waste storage (ESK 13b)
S-21	A	Update of guidelines for waste storage (ESK 13b)
S-37	A	Update of guidelines for waste storage (ESK 13b)
S-39	A	Update of guidelines for waste storage (ESK 13b)
S-47	A	Update of guidelines for waste storage (ESK 13b)
S-49	A	Update of guidelines for waste storage (ESK 13b)
S-57	A	Update of guidelines for waste storage (ESK 13b)

HUNGARY

Regulatory changes taken for the National Action Plan

The spent fuel management is regulated by the Hungarian Atomic Energy Authority (HAEA) in Hungary. There is only one facility on the list of spent fuel management facilities, the Interim Spent Fuel Storage Facility next to Paks NPP.

After benchmarking the Hungarian legal system to the WGWD Safety Reference Levels for spent fuel storage, Hungary has four SRLs in "C" ratings (S-50, S-53, S-54, S-57) which will be handled by the modification of the Volume 6 of the Nuclear Safety Codes. The planned implementation deadline is September 1^{st} 2014. The Hungarian Atomic Law will be handling the following reference level requirements: S-05, S-06, S-07. The S-04 is not fully covered and S-51 is not published yet, so they are classified as "C". Once the proposed changes to Hungary's regulatory framework will be implemented, these regulations will be in full accordance with the safety criteria required by the WGWD

A summary can be found in the attached tables.

According to the Amendment of the Hungarian Atomic Law as of July 6^{th} 2013, the HAEA will take over the supervision of all Hungarian radwaste disposal facilities as the "competent authority". According to the plan the takeover will take place on the 1^{st} of July 2014. In order to prepare this authority take over, a Decree, similar to the current "Nuclear Safety Codes" is under elaboration which has to go into effect on the date of the takeover. The new legalization shall take care all of the safety reference levels of waste. During the last half year we have been working to create and implement a new criteria system for it. Now we are working on the cross-checking process with other concerned authorities and the higher levels of the legalization.

Results of the NAP Benchmarking (Hungary)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-04	C	Necessary legal instruments not yet determined
S-05	A	Implementation of the A 2011/70/EURATOM Directive in §5 of the “Atomic Law”
S-06	C	1. § (1) 5/A. § (1) 40. § (1) of Hungarian Atomic Law
S-07	C	1. § (1) 5/A. § (1) 40. § (1) of Hungarian Atomic Law
S-10	A	New requirement in Nuclear Safety Code, Vol.1, 1.8.1.100.
S-11	A	New requirement in Nuclear Safety Code, Vol.2, 2.2.1.0100.
S-12	A	New requirement in Nuclear Safety Code, Vol.2, 2.2.1.0200.
S-13	A	New requirement in Nuclear Safety Code, Vol. 2., para 2.5.1.0200
S-14	A	New requirement in Nuclear Safety Code, Vol. 2., para 2.2.4.0100
S-15	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.3.19.0100
S-16	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.3.19.0200
S-18	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.3.19.0400
S-33	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.2.7.0900
S-34	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.3.4.0500
S-36	A	New requirement in Nuclear Safety Code, Vol. 6., para 6.3.4.0500
S-37	A	New legal requirements: i) IAEA SF-1, Principle 9 has been adopted to the Govt. Decree 118/2011 (VII.11.) ii) Chapter 6.3.21. Emergency preparedness in Nuclear Safety Code, Vol. 6. has been amended.
S-40	A	New legal requirement in Nuclear Safety Code, Vol. 6., para 6.3.17.0100
S-42	A	A totally new licensing process for modification was developed in Ch. 6.3.9. of Nuclear Safety Codes, Vol.6.
S-44	A	New requirements in Nuclear Safety Codes, Vol.6. Ch. 6.3.10. "Maintenance'
S-48	A	New requirements in Nuclear Safety Codes, Vol.6. Ch. 6.3.14. 'Assessment of safety operation'
S-49	A	New requirements for the acceptance criteria in the Nuclear Safety Codes, Vol.6. Ch. 6.3.4. 'Handling spent fuel assemblies'
S-50	C	New requirements for the retrievability of SFs in the Nuclear Safety Codes, Vol.6. Ch. 6.2.1. 'Safety functions'
S-51	C	Is covered by amendment to the regulation, not yet published
S-52	A	New requirements for the acceptance criteria in the Nuclear Safety Codes, Vol.6. Ch. 6.3.4. 'Handling spent fuel assemblies'
S-53	C	New requirements for the acceptance criteria in the Nuclear Safety Codes, Vol.6. Ch. 6.3.4. 'Handling spent fuel assemblies'
S-54	C	New requirements for the acceptance criteria in the Nuclear Safety Codes, Vol.6. Ch. 6.3.4. 'Handling spent fuel assemblies'
S-57	C	Detailed content of the SC will be developed in a Safety Guide

Results of the NAP Benchmarking (Hungary, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	C	Amendment to be published in 2014
S-04	C	Amendment to be published in 2014
S-05	C	Amendment to be published in 2014
S-06	C	Amendment to be published in 2014
S-07	C	Amendment to be published in 2014
S-08	C	Amendment to be published in 2014
S-09	C	Amendment to be published in 2014
S-10	C	Amendment to be published in 2014
S-11	C	Amendment to be published in 2014
S-12	C	Amendment to be published in 2014
S-13	C	Amendment to be published in 2014
S-14	C	Amendment to be published in 2014
S-15	C	Amendment to be published in 2014
S-16	C	Amendment to be published in 2014
S-18	C	Amendment to be published in 2014
S-19	C	Amendment to be published in 2014
S-20	C	Amendment to be published in 2014
S-21	C	Amendment to be published in 2014
S-22	C	Amendment to be published in 2014
S-23	C	Amendment to be published in 2014
S-24	C	Amendment to be published in 2014
S-25	C	Amendment to be published in 2014
S-26	C	Amendment to be published in 2014
S-27	C	Amendment to be published in 2014
S-28	C	Amendment to be published in 2014
S-29	C	Amendment to be published in 2014
S-31	C	Amendment to be published in 2014
S-32	C	Amendment to be published in 2014
S-33	C	Amendment to be published in 2014
S-34	C	Amendment to be published in 2014
S-35	C	Amendment to be published in 2014
S-36	C	Amendment to be published in 2014
S-37	C	Amendment to be published in 2014
S-40	C	Amendment to be published in 2014
S-41	C	Amendment to be published in 2014
S-42	C	Amendment to be published in 2014
S-43	C	Amendment to be published in 2014
S-44	C	Amendment to be published in 2014
S-45	C	Amendment to be published in 2014
S-46	C	Amendment to be published in 2014
S-47	C	Amendment to be published in 2014
S-48	C	Amendment to be published in 2014
S-49	C	Amendment to be published in 2014
S-50	C	Amendment to be published in 2014
S-51	C	Amendment to be published in 2014
S-54	C	Amendment to be published in 2014
S-57	C	Amendment to be published in 2014
S-58	C	Amendment to be published in 2014

ITALY

Regulatory changes taken for the National Action Plan

The WGWD Safety Reference Levels for waste and spent fuel storage are considered in ITALY in the process of update of national legal framework.

A Regulatory Guide has been developed on “Safety Criteria for Radioactive Waste Storage”. The Guide is in an advanced stage of development. A consultation process with interested entities (Operators, others Ministers, etc.) will start by April 2014. Publication is expected by summer 2014. SRLs 11 and 13 addressing management systems are not completely responding to the WENRA SRLs for the reason that a general regulatory guide on Management Systems for Radioactive Waste Management and Decommissioning is foreseen by next year.

As far as the SRLs on spent fuel storage is concerned, ITALY is of the opinion that it is not necessary to develop specify regulatory guide for spent fuel storage for the reason that the existing spent fuel still present in Italy is in the process of being transferred to France for reprocessing. This process will be completed by 2015. The only spent fuel Italy will have to manage is only 1.6 tHM of U-Th that will be stored in a dual purpose metal cask.

Results of the NAP Benchmarking (Italy)
Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-03	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-05	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-08	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-09	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-10	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-11	C	Regulatory guide on management systems (in development)
S-12	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-13	C	Regulatory guide on management systems (in development)
S-14	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-17	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-18	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-19	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-20	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-21	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-22	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-24	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-25	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-26	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-27	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-28	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-29	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-31	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-32	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-33	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-34	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-35	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-36	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-37	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-39	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-40	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-41	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-42	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-43	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-44	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-45	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-46	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-47	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-48	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-49	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-50	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-51	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-53	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-54	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-55	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-56	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-57	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-58	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-59	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-60	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”
S-61	A	Reg. Guide on “Safety Criteria for Radioactive Waste Storage”

LITHUANIA

Regulatory changes taken for the National Action Plan

After benchmarking the Lithuanian legal system to the WGWD Safety Reference Levels for waste and spent fuel storage, Lithuania had about 30% of C ratings. This means Lithuania had to improve its legal system in order to reach “A” ratings for all safety reference levels. Main deficiencies found were related to periodic safety reviews of safety case and operational limits and conditions issues also to some more specific requirements defined in safety reference levels.

In 2010, two legal documents (BSR–3.1.2–2010: Regulation on the Pre-disposal Management of Radioactive Waste at the Nuclear Facilities, and BSR–3.1.1–2010: General Requirements for Spent Nuclear Fuel Storage Facility of the Dry Type) were revised. During this revision, requirements were supplemented. During the 30th WGWD meeting in Prague in February 2013, Lithuania provided all the changes of legal system to the members of the group and had no objection to state that Lithuanian national action plan for storage document is implemented.

Results of the NAP Benchmarking (Lithuania)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	Updated BSR-3.1.1-2010
S-06	A	Updated BSR-3.1.1-2010
S-07	A	Updated BSR-3.1.1-2010
S-10	A	Updated BSR-3.1.1-2010
S-11	A	Updated BSR-3.1.1-2010
S-12	A	Updated BSR-3.1.1-2010
S-13	A	Updated BSR-3.1.1-2010
S-14	A	Updated BSR-3.1.1-2010
S-16	A	Updated BSR-3.1.1-2010
S-17	A	Updated BSR-3.1.1-2010
S-22	A	Updated BSR-3.1.1-2010
S-27	A	Updated BSR-3.1.1-2010
S-29	A	Updated BSR-3.1.1-2010
S-31	A	Updated BSR-3.1.1-2010
S-37	A	Updated BSR-3.1.1-2010
S-38	A	Updated BSR-3.1.1-2010
S-39	A	Updated BSR-3.1.1-2010
S-42	A	Updated BSR-3.1.1-2010
S-44	A	Updated BSR-3.1.1-2010
S-48	A	Updated BSR-3.1.1-2010
S-50	A	Updated BSR-3.1.1-2010
S-51	A	Updated BSR-3.1.1-2010
S-54	A	Updated BSR-3.1.1-2010
S-56	A	Updated BSR-3.1.1-2010
S-57	A	Updated BSR-3.1.1-2010
S-58	A	Updated BSR-3.1.1-2010
S-59	A	Updated BSR-3.1.1-2010
S-60	A	Updated BSR-3.1.1-2010
S-61	A	Updated BSR-3.1.1-2010

Results of the NAP Benchmarking (Lithuania, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	**A**	Updated BSR-3.1.2-2010
S-06	**A**	Updated BSR-3.1.2-2010
S-07	**A**	Updated BSR-3.1.2-2010
S-10	**A**	Updated BSR-3.1.2-2010
S-11	**A**	Updated BSR-3.1.2-2010
S-12	**A**	Updated BSR-3.1.2-2010
S-13	**A**	Updated BSR-3.1.2-2010
S-14	**A**	Updated BSR-3.1.2-2010
S-17	**A**	Updated BSR-3.1.2-2010
S-19	**A**	Updated BSR-3.1.2-2010
S-22	**A**	Updated BSR-3.1.2-2010
S-27	**A**	Updated BSR-3.1.2-2010
S-29	**A**	Updated BSR-3.1.2-2010
S-34	**A**	Updated BSR-3.1.2-2010
S-36	**A**	Updated BSR-3.1.2-2010
S-37	**A**	Updated BSR-3.1.2-2010
S-38	**A**	Updated BSR-3.1.2-2010
S-39	**A**	Updated BSR-3.1.2-2010
S-42	**A**	Updated BSR-3.1.2-2010
S-44	**A**	Updated BSR-3.1.2-2010
S-50	**A**	Updated BSR-3.1.2-2010
S-56	**A**	Updated BSR-3.1.2-2010
S-57	**A**	Updated BSR-3.1.2-2010
S-58	**A**	Updated BSR-3.1.2-2010
S-59	**A**	Updated BSR-3.1.2-2010
S-60	**A**	Updated BSR-3.1.2-2010
S-61	**A**	Updated BSR-3.1.2-2010

THE NETHERLANDS

Regulatory changes taken for the National Action Plan

The Netherlands committed itself in 2011 to implement the WGWD Safety Reference Levels (SRLs) on radioactive waste management in its legal system.

The most relevant elements of the Dutch legal system are given by the Nuclear Energy Act, together with the Radiation Protection Decree and the Nuclear Installations, Fissionable Materials and Ores Decree. This legislation provides for a system of mainly general goal oriented rules and regulations. It also establishes a licensing system.

The Netherlands has a small nuclear program with one national radioactive waste management organization, i.e. the Central Organisation for Radioactive Waste (COVRA), located at one site. The single and unique role of COVRA in the Netherlands has been established in legislation. Due to this fact, The Netherlands has decided in the past to regulate waste and spent fuel storage mainly by means of the COVRA license conditions rather than by means of generic guidelines in legislation.

The implementation of the SRLs into the Dutch legal system was benchmarked for the first time at the 21st WGWD meeting in Sofia in November 2008. The Netherlands reported detailed references to the COVRA-license and plans for updating the COVRA-license.

At the 29th WGWD meeting in Stockholm in September 2012, the Netherlands reported its progress in the legal implementations for re-benchmarking.

A majority of SRLs are implemented in the COVRA license and rated B (justified difference) by WGWD, based on the single case in the Netherlands described above. The remaining eight C-ratings deal with requirements on aspects of the management system, periodic testing and inspection and contingency plan and arrangements. These SRLs will be implemented by means of a revision of the COVRA license in 2014 and by means of a new ordinance on Management and Organisation in 2015.

Results of the NAP Benchmarking (Netherlands)
Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	Ordinance on the implementation of the Nuclear Safety Directive, article 5, effective since June 30, 2013
S-03	A	Ordinance on the implementation of the Nuclear Safety Directive, article 2 and 3, effective since June 30, 2013
S-05	C	Introduce requirement in next amendment of COVRA-license due in 2014
S-06	B	No action is foreseen to address A-practice in legal system
S-09	A	Ordinance on implementation Nuclear Safety Directive, effective since June 30, 2013
S-11	C	Ordinance on Management and Organisation, due in 2015
S-12	C	Ordinance on Management and Organisation, due in 2015
S-13	C	Ordinance on Management and Organisation, due in 2015
S-14	C	Ordinance on Management and Organisation, due in 2015
S-15	B	COVRA license
S-16	B	COVRA license
S-18	B	COVRA license
S-20	B	COVRA license
S-21	B	COVRA license
S-25	B	COVRA license
S-27	B	COVRA license
S-28	B	COVRA license
S-32	B	COVRA license
S-33	B	COVRA license
S-34	B	COVRA license
S-35	B	COVRA license
S-36	B	COVRA license
S-39	B	Nuclear Energy Act, Article 40, COVRA license
S-40	B	COVRA license
S-47	C	Introduce requirement in next amendment of COVRA-license, due in 2014
S-48	C	Introduce requirement in next amendment of COVRA-license, due in 2014
S-49	B	COVRA license
S-50	C	Introduce requirement in next amendment of COVRA-license, due in 2014
S-51	B	COVRA license
S-52	B	COVRA license
S-53	B	COVRA license
S-54	B	COVRA license
S-57	B	COVRA license
S-58	B	COVRA license
S-59	A	Ordinance on the implementation of the Nuclear Safety Directive, article 5, effective since June 30, 2013
S-60	B	COVRA license
S-61	A	Ordinance on the implementation of the Nuclear Safety Directive, article 5, effective since June 30, 2013

ROMANIA

Regulatory changes taken for the National Action Plan

The WGWD Safety Reference Levels for waste and spent fuel storage are considered in national regulatory framework which is under revision now.

The revision of the regulatory framework was required since the transposition of the provisions of Council Directive 2011/70/EURATOM establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste and will continue with the transposition of the Council Directive 2013/59/EURATOM of 5th December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation.

Current national legal framework, especially the Law 111/1996 on the safe deployment, regulation, licensing and control of nuclear activities, Order 14/2000 approving the Fundamental Regulation on the radiological safety as well as Order 56/2005 approving the Fundamental regulation on the safe management of radioactive waste and spent nuclear fuel, already comply with some of WGWD reference levels.

The non-compliances with WGWD reference levels will be treated in the new revised regulatory framework especially in the Order approving the Regulation on the safety requirements for predisposal activities and facilities and safety requirements for storage of spent nuclear fuel which is in very advanced stage. The Regulation on the safety requirements for predisposal activities and facilities and safety requirements for storage of spent nuclear fuel will comply with WGWD Safety Reference Levels and it is estimate to be in force at the end of 2014.

Results of the NAP Benchmarking (Romania)
Spent Fuel and Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	CNCAN order, to be enacted in 2014
S-03	A	CNCAN order, to be enacted in 2014
S-06	A	CNCAN order, to be enacted in 2014
S-10	A	CNCAN order, to be enacted in 2014
S-11	A	CNCAN order, to be enacted in 2014
S-12	A	CNCAN order, to be enacted in 2014
S-13	A	CNCAN order, to be enacted in 2014
S-14	A	CNCAN order, to be enacted in 2014
S-16	A	CNCAN order, to be enacted in 2014
S-17	A	CNCAN order, to be enacted in 2014
S-18	A	CNCAN order, to be enacted in 2014
S-19	A	CNCAN order, to be enacted in 2014
S-20	A	CNCAN order, to be enacted in 2014
S-21	A	CNCAN order, to be enacted in 2014
S-22	A	CNCAN order, to be enacted in 2014
S-23	A	CNCAN order, to be enacted in 2014
S-24	A	CNCAN order, to be enacted in 2014
S-25	A	CNCAN order, to be enacted in 2014
S-26	A	CNCAN order, to be enacted in 2014
S-27	A	CNCAN order, to be enacted in 2014
S-28	A	CNCAN order, to be enacted in 2014
S-29	A	CNCAN order, to be enacted in 2014
S-30	A	CNCAN order, to be enacted in 2014
S-31	A	CNCAN order, to be enacted in 2014
S-32	A	CNCAN order, to be enacted in 2014
S-33	A	CNCAN order, to be enacted in 2014
S-34	A	CNCAN order, to be enacted in 2014
S-35	A	CNCAN order, to be enacted in 2014
S-36	A	CNCAN order, to be enacted in 2014
S-37	A	CNCAN order, to be enacted in 2014
S-39	A	CNCAN order, to be enacted in 2014
S-40	A	CNCAN order, to be enacted in 2014
S-41	A	CNCAN order, to be enacted in 2014
S-42	A	CNCAN order, to be enacted in 2014
S-43	A	CNCAN order, to be enacted in 2014
S-44	A	CNCAN order, to be enacted in 2014
S-45	A	CNCAN order, to be enacted in 2014
S-46	A	CNCAN order, to be enacted in 2014
S-47	A	CNCAN order, to be enacted in 2014
S-48	A	CNCAN order, to be enacted in 2014
S-49	A	CNCAN order, to be enacted in 2014
S-50	A	CNCAN order, to be enacted in 2014
S-51	A	CNCAN order, to be enacted in 2014
S-52	A	CNCAN order, to be enacted in 2014
S-53	A	CNCAN order, to be enacted in 2014
S-54	A	CNCAN order, to be enacted in 2014
S-55	A	CNCAN order, to be enacted in 2014
S-56	A	CNCAN order, to be enacted in 2014
S-57	A	CNCAN order, to be enacted in 2014
S-58	A	CNCAN order, to be enacted in 2014
S-60	A	CNCAN order, to be enacted in 2014
S-61	A	CNCAN order, to be enacted in 2014

SLOVAKIA

Regulatory changes taken for the National Action Plan

With the publication of the WGWD Safety Reference Levels for waste and spent fuel storage in February 2011, Slovakia continued to fulfil its obligations to implement necessary changes into its national regulations.

ÚJD SR as a central governmental body, within its competency, prepares legislation and establishes binding nuclear safety criteria for nuclear installations.

Necessary changes identified during the legal benchmarking of Safety Reference Levels for waste and spent fuel storage were implemented mainly by the update of the Act No. 541/2004 Coll. l. on peaceful use of nuclear energy (the Atomic Act) or by the update of its respective regulations.

The Atomic Act came into effect on Dec. 1st, 2004 and repealed the original Act No. 130/1998 Coll. l., as well as all its implementing regulations. The Act has been amended several times: 125/2006, 238/2006, 21/2007, 94/2007, 335/2007, 408/2008, 120/2010, 137/2010, 145/2010, 350/2011 and the last amendment was No. 143/2013. By before mentioned updates of the Atomic Act were amongst others aspects addressed also areas identified during the legal benchmarking of Safety Reference Levels for waste and spent fuel storage like e.g. graded approach and periodic safety review.

Details of safe management of spent fuel and radioactive waste were both elaborated by the Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel (valid from 2012/03/01), which replaced previously binding Regulation of ÚJD SR No. 53/2006 Coll. l., setting the details of requirements for handling nuclear materials, radioactive waste and spent nuclear fuel. Main updates here were related to appropriate contingency arrangements during storage of radioactive waste and spent fuel, requirements to develop and maintain a record system on the location and characteristics of radioactive waste, retrieval of radioactive waste and spent fuel within an appropriate time, at the end of the facility operation or in order to intervene in the event of unexpected faults, etc.

Missing aspects related to area of management system were addressed by the Regulation No. 431/2011 Coll. on a quality management system (valid from 2012/1/1).

Missing aspects related to the demonstration of construction standards and material used, with respect to the length of the storage period of radioactive waste and spent fuel, were addressed by the Regulation No.430/2011 Coll. on details on nuclear safety requirements for nuclear facilities (valid from 2012/1/1).

At the 29th WGWD meeting in Stockholm in September 2012, Slovakia reported its regulatory implementations for benchmarking, all changes were approved and the Slovak regulations were found to be in full agreement with the requirements mandated by the WGWD SRLs.

Results of the NAP Benchmarking (Slovakia)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	**A**	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act); Regulation No. 431/2011 Coll. on a quality management system
S-10	**A**	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-11	**A**	Regulation No. 431/2011 Coll. on a quality management system; Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-12	**A**	Regulation No. 431/2011 Coll. on a quality management system
S-13	**A**	Regulation No. 431/2011 Coll. on a quality management system; Act No. 350/2011 amending and supplementing Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-14	**A**	Regulation No. 431/2011 Coll. on a quality management system
S-22	**A**	Regulation No. 430/2011 Coll. on details on nuclear safety requirements for nuclear facilities; Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-31	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel; Regulation No. 431/2011 Coll. on a quality management system
S-34	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-37	**A**	Act No. 350/2011 amending and supplementing Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act); Regulation No. 35/2012 Coll., changing and amending Decree No. 55/2006 Coll., on details of emergency planning in case of a nuclear incident or accident
S-45	**A**	Regulation No. 430/2011 Coll. on details on nuclear safety requirements for nuclear facilities
S-48	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-49	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-50	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-54	**A**	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-58	**A**	Act No. 350/2011 amending and supplementing Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act) (→ Regulation No. 33/2012 Coll.)
S-60	**A**	Regulation No. 33/2012 Coll. on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear equipment: Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel

Results of the NAP Benchmarking (Slovakia, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act); Regulation No. 431/2011 Coll. on a quality management system
S-10	A	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-11	A	Regulation No. 431/2011 Coll. on a quality management system; Act No. 350/2011 amending and supplementing Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-12	A	Regulation No. 431/2011 Coll. on a quality management system
S-13	A	Regulation No. 431/2011 Coll. on a quality management system; Act No. 350/2011 amending and supplementing Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act)
S-14	A	Regulation No. 431/2011 Coll. on a quality management system
S-15	A	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-22	A	Regulation No. 430/2011 Coll. on details on nuclear safety requirements for nuclear facilities; Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-34	A	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-37	A	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act); Regulation No. 35/2012 Coll., Decree No. 55/2006 Coll., on details of emergency planning in case of a nuclear incident or accident
S-45	A	Regulation No. 430/2011 Coll. on details on nuclear safety requirements for nuclear facilities
S-48	A	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-49	A	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-50	A	Regulation No. 30/2012 Coll., laying down details of requirements for the handling of nuclear materials, nuclear waste and spent nuclear fuel
S-58	A	Act No. 350/2011 to Act No. 541/2004 Coll. on the Peaceful Use of Nuclear Energy (Atomic Act) (→ Regulation No. 33/2012 Coll.)
S-60	A	Regulation No. 33/2012 Coll. on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear equipment

SLOVENIA

Regulatory changes taken for the National Action Plan

Slovenian Nuclear Safety Administration (SNSA) as the competent authority in the field of radioactive waste and spent fuel storage continuously takes all necessary actions for implementation of changes in obligations into the national regulatory requirements. Slovenian regulatory framework in the pertinent field consists mainly of the Ionizing Radiation Protection and Nuclear Safety Act, Resolution on the 2006-2015 National Program for Managing Radioactive Waste and Spent Nuclear Fuel and a list of rules which regulate specific areas of waste and spent fuel management in detail. Slovenia made the main step forward to the full consistency of its regulatory framework with the new international standards and recommendations when, in 2009, two new regulations were published, namely Rules on radiation and nuclear safety factors (JV5) and the Rules on operational safety of radiation and nuclear facilities (JV9). The rules set detailed requirements for design bases, contents of applications and main safety documentation, management system, modification management, periodic safety reviews and others.

At the 30th WGWD meeting in Prague, Slovenia reported on the implementation of storage SRLs and its action plan. The majority of the SRLs were implemented through new rules JV5 and JV9. Therefore all changes were approved except one SRL where a better reference was required. The SRL refers to the reserve storage capacity to stay available for retrieved waste and spent fuel packages. Based on this requirement the SNSA made a proposal for additional amendment of the JV5 Rules. It is expected that this amendment will be published by the end of 2014. Beside the identified deficiency the Slovenian regulations were found to be in full agreement with the requirements mandated by the WGWD SRLs. There is one SRL rated B but it is not required to make any changes in national legislation. It refers to an option on having adopted the burnup credit.

Results of the NAP Benchmarking (Slovenia)
Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	JV5 (Rules on radiation and nuclear safety factors)
S-08	A	JV5
S-09	A	JV5 and ZVISJV (Ionizing Radiation Protection and Nuclear Safety Act)
S-11	A	JV5 and ZVISJV
S-12	A	JV5
S-13	A	JV5
S-14	A	JV5
S-19	A	JV5
S-20	A	JV5
S-21	A	JV5
S-22	A	JV5
S-23	A	JV5
S-24	A	JV5
S-25	A	JV5 and JV9 (Rules on operational safety of radiation or nuclear facilities)
S-26	A	JV5
S-27	A	JV5
S-28	A	JV5
S-29	B	JV5
S-30	A	JV5
S-31	A	JV5
S-32	A	JV5
S-33	A	JV5
S-34	A	JV5
S-35	A	JV5 and JV7 (Regulation on radioactive waste and spent fuel management)
S-36	C	Amendments of JV5 proposed, not yet published
S-37	A	JV9
S-38	A	JV9 and ZVISJV
S-39	A	JV5, JV9 and National Emergency Response Plan for Nuclear and Radiological Accidents
S-40	A	ZVISJV and JV9
S-41	A	JV9
S-42	A	ZVISJV and JV9
S-43	A	JV9
S-44	A	JV9
S-45	A	JV5
S-46	A	JV9
S-47	A	JV9
S-48	A	JV9
S-49	A	JV5
S-50	A	JV5
S-59	A	ZVISJV and JV9
S-60	A	JV9
S-61	A	JV9

Results of the NAP Benchmarking (Slovenia, cont.)
Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-02	A	JV5 (Rules on radiation and nuclear safety factors)
S-08	A	JV5
S-11	A	JV5 and ZVISJV (Ionizing Radiation Protection and Nuclear Safety Act)
S-14	A	JV5
S-19	A	JV5
S-20	A	JV5
S-21	A	JV5
S-22	A	JV5
S-23	A	JV5
S-24	A	JV5
S-25	A	JV5 and JV9 (Rules on operational safety of radiation or nuclear facilities)
S-26	A	JV5
S-27	A	JV5
S-28	A	JV5
S-29	B	JV5
S-30	A	JV5
S-31	A	JV5
S-32	A	JV5
S-33	A	JV5
S-34	A	JV5
S-35	A	JV5 and JV7 (Regulation on radioactive waste and spent fuel management)
S-36	C	Amendments of JV5 proposed, not yet published
S-37	A	JV9
S-38	A	JV9 and ZVISJV
S-39	A	JV5, JV9 and National Emergency Response Plan for Nuclear and Radiological Accidents
S-40	A	ZVISJV and JV9
S-41	A	JV9
S-42	A	ZVISJV and JV9
S-43	A	JV9
S-44	A	JV9
S-45	A	JV5
S-46	A	JV9
S-47	A	JV9
S-48	A	JV9
S-49	A	JV5
S-50	A	JV5
S-59	A	ZVISJV and JV9
S-60	A	JV9
S-61	A	JV9

SPAIN

Regulatory changes taken for the National Action Plan

As result of the first benchmarking, the Spanish NAP Table had a total of 39 SRLs evaluated as ´C´, for both spent fuel and waste categories. This led to the elaboration of several CSN Safety Standards, in particular the IS-29, "Safety Criteria at Spent Fuel and High Level Waste Storage Facilities (Official Gazette of 2-11-2010) which addresses not only Spent Fuel and High Level Waste but also `Special Waste´, therein defined as:

> "fuel accessories […], reactor internals […] and that other waste which is not susceptible, given its radiological characteristics, of being managed at the L-I LW surface disposal facility (El Cabril)".

During the 2nd benchmarking, in the 30th Meeting some difficulties arose about the applicability of this IS-29 to graphite in Vandellós 1 NPP (currently undergoing a differed decommissioning process). The position shown by Spanish representatives in the 32nd Meeting is that the IS-29 applies also to graphite waste, given that the scope of this standard encompasses all waste not accepted in `El Cabril´ disposal facility. This position was accepted by the Group.

In addition, a new text for the SRL-47 was provided, which was also accepted. Therefore, after the 32nd meeting, the Spanish NAP table has been evaluated with all the 61 SRLs as A in both categories Spent Fuel and Waste, i.e. in full agreement with the Storage Report V-2.1 .

Results of the NAP Benchmarking (Spain)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	Royal Decree 1836/1999: Regulation on Nuclear and Radioactive Facilities, modified by Royal Decree 35/2008
S-11	A	IS-26: "Basic Nuclear Safety Requirements applicable to Nuclear Installations (June 2010)"
S-12	A	IS-26
S-13	A	IS-26
S-14	A	IS-26
S-16	A	IS-29 "CSN Safety Standard Safety Criteria for Spent Fuel and High Level Waste (Jul 2010)"
S-17	A	IS-29
S-18	A	IS-29
S-19	A	IS-29
S-20	A	IS-29
S-21	A	IS-29
S-22	A	IS-29
S-24	A	IS-29
S-25	A	IS-29
S-27	A	IS-29
S-28	A	IS-29
S-29	A	IS-29
S-30	A	IS-29
S-31	A	IS-29
S-32	A	IS-29
S-33	A	IS-29
S-34	A	IS-29
S-35	A	IS-20 "CSN Safety Standard Design Criteria for Spent Fuel Storage Casks (Feb 2003)"; IS-29
S-36	A	IS-29
S-37	A	Royal Decr. 1836/1999; IS-29
S-40	A	IS-29
S-41	A	IS-29
S-44	A	IS-29
S-45	A	IS-20
S-46	A	IS-20
S-47	A	IS-20
S-48	A	IS-20
S-49	A	IS-20
S-50	A	IS-20
S-51	A	IS-20
S-52	A	IS-20
S-53	A	IS-20
S-54	A	IS-20
S-55	A	IS-20
S-56	A	IS-20
S-57	A	IS-20
S-58	A	IS-29
S-59	A	IS-29
S-60	A	IS-26; IS-29
S-61	A	IS-26

Results of the NAP Benchmarking (Spain, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	Royal Decree 1836/1999: Regulation on Nuclear and Radioactive Facilities, modified by Royal Decree 35/2008
S-11	A	IS-26: "Basic Nuclear Safety Requirements applicable to Nuclear Installations"
S-12	A	IS-26
S-13	A	IS-26
S-14	A	IS-26
S-16	A	IS-29 "Safety Criteria for Spent Fuel and High Level Waste "
S-17	A	IS-29
S-18	A	IS-29
S-19	A	IS-29
S-20	A	IS-29
S-21	A	IS-29
S-22	A	IS-29
S-24	A	IS-29
S-25	A	IS-29
S-27	A	IS-29
S-28	A	IS-29
S-29	A	IS-29
S-30	A	IS-29
S-31	A	IS-29
S-32	A	IS-29
S-33	A	IS-29
S-34	A	IS-29
S-35	A	IS-29
S-36	A	IS-29
S-37	A	Royal Decree 1836/1999, IS-29
S-39	A	Royal Decree 1836/1999, IS-29
S-40	A	IS-29
S-41	A	IS-29
S-44	A	IS-29
S-45	A	IS-20 "Design Criteria for Spent Fuel Storage Casks"
S-46	A	IS-20
S-47	A	IS-29
S-48	A	IS-20
S-49	A	IS-29
S-50	A	IS-29
S-51	A	IS-29
S-52	A	IS-29
S-53	A	IS-29
S-54	A	IS-29
S-55	A	IS-29
S-56	A	IS-29
S-58	A	IS-29
S-59	A	IS-29
S-60	A	IS-26
S-61	A	IS-26

SWEDEN

Regulatory changes taken for the National Action Plan

The Swedish regulations on waste and spent fuel were updated and elaborated and taken in force on the first of November 2012. A lot of work was done to make sure that the WGWD Safety Reference Levels for waste and spent fuel storage was implemented in Swedish regulations.

The changes necessary to the Swedish regulations were mostly in the area of designing storage facilities in a manner that would facilitate retrievability, inspection, maintenance of the stored material. Also acceptance criteria needed to be implemented.

At the 29 WGDW meeting in Stockholm in September 2012, Sweden reported its regulatory implementations for benchmarking. All changes were approved with the sole exception of one reference to both waste and spent fuel storage concerning the SRL-50 regarding contingency arrangements for material that are not retrievable by normal means or show signs of degradation. This reference was prior to the 31 WGWD in Rome evaluated as ‘Earmarked’. That SRL was addressed on the 31 WGDW in Rome, where sufficient information was provided and SRL-50 was benchmarked as an ‘A’. Swedish regulations were now found to be in full agreement with the requirements mandated by the WGWD SRLs.

Results of the NAP Benchmarking (Sweden)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	SSMFS 2008:1 (rev. 2011:3) CH2 §9, SSMFS 2008:32 §§ 10-13
S-11	A	SSMFS 2008:1 (rev. 2011:3) CH2 §8
S-12	A	SSMFS 2008:1 (rev. 2011:3) CH1 §1, CH2 §8 CH5 §2
S-13	A	SSMFS 2008:1 (rev. 2011:3) CH2 §8
S-14	A	SSMFS 2008:1 (rev. 2011:3) CH2 §§8-9
S-15	A	SSMFS 2008:1 (rev. 2011:3) CH6 §10 AR, CH8, SSMFS 2008:3 and SSMFS 2008:38
S-17	A	SSMFS 2008:1 (rev. 2011:3) CH6 §10 AR, CH8, SSMFS 2008:3 and SSMFS 2008:38
S-18	A	SSMFS 2008:3 and COUNCIL REGULATION (EC) No 1334/2000
S-20	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-21	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-27	A	SSMFS 2008:1 (rev. 2011:3) AR APP3
S-32	A	SSMFS 2008:1 (rev. 2011:3) CH6 §§1-2
S-33	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-34	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-35	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-36	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-37	A	SSMFS 2008:1 (rev. 2011:3) CH2 §12
S-39	A	SSMFS 2008:1 (rev. 2011:3) CH2 §13
S-48	A	SSMFS 2008:1 (rev. 2011:3) CH5 §3, AR CH6 §2
S-49	A	SSMFS 2008:1 (rev. 2011:3) CH6 §12
S-50	A	SSMFS 2008:1 (rev. 2011:3) AR CH6 §2
S-52	A	SSMFS 2008:1 (rev. 2011:3) CH6 §11
S-53	A	SSMFS 2008:1 (rev. 2011:3) CH6 §11
S-54	A	SSMFS 2008:1 (rev. 2011:3) CH6 §12
S-57	A	SSMFS 2008:1 (rev. 2011:3) CH4 §2
S-58	A	SSMFS 2008:1 (rev. 2011:3) CH4 §§2,4,5
S-60	A	The Act on Nuclear Activities §10 a

Results of the NAP Benchmarking (Sweden, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	SSMFS 2008:1 (rev. 2011:3) CH2 §9, SSMFS 2008:32 §§ 10-13
S-11	A	SSMFS 2008:1 (rev. 2011:3) CH2 §8
S-12	A	SSMFS 2008:1 (rev. 2011:3) CH1 §1, CH2 §8 CH5 §2
S-13	A	SSMFS 2008:1 (rev. 2011:3) CH2 §8
S-14	A	SSMFS 2008:1 (rev. 2011:3) CH2 §9
S-18	A	SSMFS 2008:1 (rev. 2011:3) CH6 §10 and SSMFS 2008:38
S-20	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-21	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-22	A	SSMFS 2008:1 (rev. 2011:3) CH3, CH6 §2
S-27	A	SSMFS 2008:1 (rev. 2011:3) AR APP3
S-32	A	SSMFS 2008:1 (rev. 2011:3) CH6 §§1-2
S-33	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-34	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-35	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-36	A	SSMFS 2008:1 (rev. 2011:3) CH6 §2
S-37	A	SSMFS 2008:1 (rev. 2011:3) CH2 §12
S-39	A	SSMFS 2008:1 (rev. 2011:3) CH2 §13
S-48	A	SSMFS 2008:1 (rev. 2011:3) CH5 §3, AR CH6 §2
S-49	A	SSMFS 2008:1 (rev. 2011:3) CH6 §12
S-50	A	SSMFS 2008:1 (rev. 2011:3) CH6 §4
S-52	A	SSMFS 2008:1 (rev. 2011:3) CH6 §11
S-53	A	SSMFS 2008:1 (rev. 2011:3) CH6 §11
S-54	A	SSMFS 2008:1 (rev. 2011:3) CH6 §12
S-57	A	SSMFS 2008:1 (rev. 2011:3) CH4 §2
S-58	A	SSMFS 2008:1 (rev. 2011:3) CH4 §§2,4,5
S-60	A	The Act on Nuclear Activities §10 a

SWITZERLAND

Regulatory changes taken for the National Action Plan

Before benchmarking WENRA, safety reference levels requirements on interim storage facilities as formulated in the nuclear energy act and the nuclear energy ordinance have not been detailed in a specific regulatory guide with the exception of a guide addressing specifically spent fuel dry storage facilities (HSK-R-52). This is why most general SRLs could be rated A at first hand whereas most of the C-ratings refer to more specific reference levels which, according to Swiss regulatory principles, should be reserved to be detailed in regulatory guides.

As the old guide on dry storage of spent fuel had to be updated anyway, this procedure has been used to widen the scope of this guide in order to address also wet storage of spent fuel as well as storage of any other radioactive waste. Most C-ratings could be addressed in this new regulatory guide ENSI-G04: "*Design and operation of storage facilities for radioactive waste and spent fuel*" which has been published March 1st, 2012.

Some of the C-ratings on emergency preparedness are covered by the regulatory guide ENSI-B12: *"Emergency preparedness in nuclear facilities*". Although this guide had already been published before the first benchmarking exercise it had not been considered to the full extend applicable.

Those C-ratings which referred to human and organizational factors have been considered in the update of the already existing regulatory guide ENSI-G07 "*Organization of nuclear facilities*" which has been published June 28th, 2013. However even the previous version of this guide already covered the SRL requirements as this only was relevant for the management system related SRLs, which were marked as "unclear" in the translation procedure from SRLs in Version 1 of the report to SRLs in the up-to-date Version 2.

Results of the NAP Benchmarking (Switzerland)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-06	A	Reg Guide G04
S-10	A	No changes necessary
S-11	A	Reg. Guide G04 and G07
S-12	A	Reg. Guide G07
S-13	A	Reg. Guide G07
S-14	A	Reg. Guide G07
S-16	A	Reg. Guide G04
S-27	A	Reg. Guide G04
S-33	A	Reg. Guide G04
S-34	A	Reg. Guide G04
S-35	A	Reg. Guide G04
S-37	A	Ordinance SR 732.112.2
S-38	A	Reg. Guide B12
S-39	A	Reg. Guide B11
S-43	A	Reg. Guide G04; Reg. Guide A04
S-44	A	Reg. Guide G04
S-47	A	Reg. Guide G04
S-48	A	Reg. Guide G04
S-49	A	Reg. Guide G04 and G07
S-50	A	Reg. Guide G04
S-51	A	Reg. Guide G04
S-54	A	Reg. Guide G04
S-57	A	NEO
S-58	A	Reg. Guide G04
S-59	A	Reg. Guide G04
S-60	A	Reg. Guide G04
S-61	A	Reg. Guide G04

Results of the NAP Benchmarking (Switzerland, cont.)

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-06	A	Reg. Guide G04
S-10	A	No changes necessary
S-11	A	Reg. Guide G04 and G07
S-12	A	Reg. Guide G07
S-13	A	Reg. Guide G07
S-14	A	Reg. Guide B05
S-22	A	Reg. Guide G04, ch. 6.1.1
S-27	A	Reg. Guide G04; Reg. Guide B05
S-29	A	Reg. Guide G04
S-33	A	Reg. Guide G04
S-34	A	Reg. Guide G04
S-35	A	Reg. Guide G04
S-36	A	Reg. Guide G04
S-37	A	Ordinance SR 732.112.2
S-38	A	Reg. Guide B12
S-39	A	Reg. Guide B11
S-43	A	Reg. Guide G04; Reg. Guide A04
S-44	A	Reg. Guide G04
S-47	A	Reg. Guide G04
S-48	A	Reg. Guide G04
S-49	A	Reg. Guide G04 and G07
S-50	A	Reg. Guide G04
S-54	A	Reg. Guide G04
S-57	A	NEO
S-58	A	Reg. Guide G04
S-59	A	Reg. Guide G04
S-60	A	Reg. Guide G04
S-61	A	Reg. Guide G04

UNITED KINGDOM

Regulatory changes taken for the National Action Plan

The initial benching for the storage SRLs showed that the UK regulatory system was largely compliant with the SRLs, as there were only 6 SRLs in category C out of a total of 77. Since the original benchmarking the Office for Nuclear Regulation (ONR) has produced a new suite of guidance for radioactive waste, in conjunction with the UK's environmental regulators (i.e. the Joint Guidance on the management of higher activity radioactive waste on nuclear licensed sites, from the Health and Safety Executive, the Environmental Agency and Scottish Environmental Protection Agency to nuclear licensees). ONR has also reviewed and updated its Technical Assessment Guides (TAGs) and Technical Inspection Guides (TIGs). A new Technical Assessment Guide on the management of spent fuel has also been produced. The ONR Safety Assessment Principles are currently under review.

The table below shows how the storage SRLs originally marked as category C have been addressed in the UK regulatory system. The table also addresses those SRLs which have been re-benchmarked because the SRLs were significantly re-worded after the original benchmarking. The evidence to support the categorisation has been peer reviewed by the WGWD. The UK's regulatory system is therefore fully compliant with the storage SRLs.

The above statement and the table below apply to the legal benchmarking of the storage SRLs with respect to both radioactive waste and spent fuel.

Results of the NAP Benchmarking (United Kingdom)

Spent Fuel Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	New evidence from Licence Conditions and Safety Assessment Principles
S-11	A	New evidence from Licence Conditions and Safety Assessment Principles
S-12	A	New evidence from Licence Conditions and Safety Assessment Principles
S-13	A	Updated regulatory guidance on management systems
S-14	A	Updated regulatory guidance on management systems
S-31	A	New regulatory guidance (Joint Guidance on Radioactive Waste)
S-37	A	New evidence from Licence Conditions and Safety Assessment Principles and regulations
S-43	A	Updated guidance on licence conditions

Waste Storage

# SRL (new or changes req.)	Current status	Actions taken / relevant regulations
S-10	A	New evidence from Licence Conditions and Safety Assessment Principles
S-11	A	New evidence from Licence Conditions and Safety Assessment Principles
S-12	A	New evidence from Licence Conditions and Safety Assessment Principles
S-13	A	Updated regulatory guidance on management systems
S-14	A	Updated regulatory guidance on management systems
S-31	A	New regulatory guidance (Joint Guidance on Radioactive Waste)
S-37	A	New evidence from Licence Conditions and Safety Assessment Principles and regulations
S-43	A	Updated guidance on licence conditions

WENRA
WESTERN EUROPEAN NUCLEAR
REGULATORS ASSOCIATION

RHWG
REACTOR HARMONISATION
WORKING GROUP

WGWD
WORKING GROUP ON WASTE
AND DECOMISSIONING

Western European
WENRA
Nuclear Regulators Association

Report

WENRA
WORKING GROUP ON WASTE AND DECOMMISSIONING (WGWD)

DECOMMISSIONING SAFETY REFERENCE LEVELS REPORT

version 2.0

November 2011

Contents

Executive Summary

The Western European Nuclear Regulator's Association (WENRA) is an international body made up of the Heads and senior staff members of Nuclear Regulatory Authorities of European countries with nuclear power plants. The main objectives of WENRA is to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues.
To accomplish these tasks two working groups within the WENRA have been established - Reactor Harmonisation Working Group (RHWG) and Working Group on Waste and Decommissioning (WGWD).
This document contains the results of the work of WGWD in the area of the decommissioning of nuclear installations performed from 2009 to 2011 to improve the Decommissioning Safety Reference Level Report, version 1.0 of March 2007. The objective of this report is to provide safety reference levels for decommissioning activities. The version 1.0 was based on corresponding IAEA documents (requirements, guidances, etc). Although the IAEA safety standards establish an essential basis for safety of all nuclear installations covering also their decommissioning, the WENRA safety reference levels incorporate more this activity specific requirements. In version 2.0 lessons learned from the national benchmarking processes for version 1.0, especially on the implementation of the safety reference levels in the national legal and regulatory framework, were incorporated.
The document was prepared by the WENRA WGWD, based on previous version 1.0 and on the input provided by the members of WGWD listed below.

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TERMS OF REFERENCE OF THE WESTERN EUROPEAN NUCLEAR REGULATORS' ASSOCIATION (WENRA)

26 March 2010

1. We the Heads of Nuclear Regulatory Authorities (signatories) of European countries with nuclear power plants:

   - drawing from the experience already gained with WENRA and noting its achievements,
   - recognizing that the current regulatory challenges in Europe lead to envisage the activities of WENRA in a broader perspective,
   - re-affirming the need for increased co-operation between us, and
   - maintaining our independence,

   have again revised the previous Terms of Reference of the Western European Nuclear Regulators' Association (WENRA), which were signed on 4 February 1999 and revised on 14 March 2003.

2. With the general aim of improving nuclear safety, WENRA has the following objectives:

   - to build and maintain a network of chief nuclear safety regulators in Europe,
   - to promote exchange of experience and learning from each others best practices,
   - to develop a harmonized approach to nuclear safety and regulation, in particular within the European Union,
   - to discuss and, where appropriate, express its opinion on significant safety and regulatory issues.

3. Decisions in the name of WENRA are taken by consensus.

4. WENRA will keep the European Union Institutions informed about its activities, and is prepared to consider requests from these institutions for advice on nuclear safety and regulatory matters.

5. Heads of the regulatory authorities (or corresponding) in other European countries, which have expressed an interest, are invited as observers to WENRA. Observers have the right to express their opinion at the WENRA meetings but can not participate in the decision making. Observers may send suitably qualified participants to the working groups.

6. WENRA will develop and maintain, when appropriate, suitable relations with regulatory authorities from other countries as well as with international organisations.

7. WENRA will ensure appropriate opportunities for Stakeholders to comment on its work.

Belgium Willy DE ROOVERE	Bulgaria Sergey TZOTCHEV
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Glossary

Ageing

General process in which characteristics of a structure, system or component gradually change with time or use.

Management of ageing

Engineering, operations and maintenance actions to control within acceptable limits the ageing degradation of structures, systems or components.

Clearance

Removal of radioactive materials or radioactive objects within authorized practices from any further control by the regulatory body. (Removal from control in this context refers to control applied for radiation protection purposes.)

Decommissioning

Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are 'closed' and not 'decommissioned'). For a repository, the corresponding term is closure.

Decommissioning plan

An initial or final document – depending on the operational phase of the facility - with detailed information about the concept and schedule for the decommissioning and dismantling of the nuclear facility.

Initial decommissioning plan based on the decommissioning strategy includes the feasibility of decommissioning, main steps of the decommissioning/dismantling and the end state of the facility and is the basis for the estimation of decommissioning costs. This document is of general nature during the design and operational phase and will be updated during the operational phase to the level as appropriate.

Final decommissioning plan as the basis to start major decommissioning activities shall be prepared before the beginning of the decommissioning phase together with the safety case. This detailed document will be updated as required during the decommissioning stages.

Decommissioning strategies

Immediate dismantling is the strategy in which the equipment, structures and parts of a nuclear facility containing radioactive contaminants are removed or decontaminated to a level that permits the facility to be released for unrestricted use, or with restrictions imposed by the regulatory body. In this case decommissioning implementation activities begin shortly after permanent cessation of operations. It implies prompt and complete decommissioning and involves the removal and processing of all radioactive material from the facility to another new or existing licensed nuclear facility for either long-term storage or disposal.

Deferred dismantling (sometimes called safe storage, safe store or safe enclosure) is the strategy in which parts of a nuclear facility containing radioactive contaminants are either processed or placed in such a condition that they can be safely stored and maintained until they can subsequently be decontaminated and/or dismantled to levels that permit the facility to be released for other uses. The period in which those parts are safely stored and maintained is the "period of deferment".

Entombment is the strategy in which radioactive contaminants are encased in a structurally long-lived material until radioactivity decays to a level permitting unrestricted release of the nuclear facility, or release with restrictions imposed by the regulatory body. Because radioactive material will remain on the site, this essentially means that the facility will eventually become designated as a near surface waste disposal facility as long as it can meet the requirements for a near surface disposal facility.

Decontamination

The complete or partial removal of contamination by a deliberate physical, chemical or biological process.

Discharge, authorized

Planned and controlled release of (usually gaseous or liquid) radioactive material into the environment in accordance with an authorization.

Emergency

A non-routine situation that necessitates prompt action, primarily to mitigate a hazard or adverse consequences for human health and safety, quality of life, property or the environment. This includes nuclear and radiological emergencies and conventional emergencies such as fires, release of hazardous chemicals, storms or earthquakes. It includes situations for which prompt action is warranted to mitigate the effects of a perceived hazard.

Nuclear or radiological emergency. An emergency in which there is, or is perceived to be, a hazard due to:

(a) The energy resulting from a nuclear chain reaction or from the decay of the products of a chain reaction; or

(b) Radiation exposure.

Points (a) and (b) approximately represent nuclear and radiological emergencies, respectively. However, this is not an exact distinction.

Emergency Preparedness

The capability to take actions that will effectively mitigate the consequences of an emergency for human health and safety, quality of life, property and the environment.

End state

A predetermined criterion defining the point at which the specific task or process is to be considered completed. The licensee can apply for termination of the license when the proposed end-state of decommissioning activities has been reached.

Licensee

The licensee is the organization having overall responsibility for a facility or activity (the responsible organization)

> **Remark:** WGWD recognizes that this organisation may change as the facility passes to the decommissioning phase according to national strategies

Management system

A set of interrelated or interacting elements (system) for establishing policies and objectives and enabling the objectives to be achieved in an efficient and effective manner.
The management system integrates all elements of an organization into one coherent system to enable all of the organization's objectives to be achieved. These elements include the organizational structure, resources and processes. Personnel, equipment and organizational culture as well as the documented policies and processes are parts of the management system.
The organization's processes have to address the totality of the requirements on the organization as established in, for example, IAEA safety standards and other international codes and standards.

Monitoring

1. The measurement of dose or contamination for reasons related to the assessment or control of exposure
2. Continuous or periodic measurement of radiological or other parameters or determination of the status of a system, structure or component. Sampling may be involved as a preliminary step to measurement.

Nuclear facility

A facility and its associated land, buildings and equipment in which nuclear materials are produced, processed, used, handled, stored or disposed of on such a scale that consideration of safety is required.

Nuclear safety

See 'Protection and Safety'

Operation

All activities performed to achieve the purpose for which an authorized facility was constructed.

Protection and safety

The protection of people against exposure to ionizing radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur.
Safety is primarily concerned with maintaining control over sources, whereas radiation protection is primarily concerned with controlling exposure to radiation and its effects. Clearly the two are closely connected: radiation protection is very much simpler if the source in question is under control, so safety necessarily contributes towards protection. Sources come in many different types, and hence safety may be termed nuclear safety, radiation safety, radioactive waste safety or transport safety, but protection (in this sense) is primarily concerned with protecting humans against exposure, whatever the source, and so is always radiation protection.

> **Radiation protection**: The protection of people from the effects of exposure to ionizing radiation, and the means for achieving this.
>
> **Nuclear safety**: The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

Radiation protection

See ‘protection and safety’

Regulatory body

An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

Safety assessment

Assessment of all aspects of the site, design, operation and decommissioning of an authorized facility that are relevant to protection and safety.

Note: assessment should be distinguished from analysis. Assessment is aimed at providing information that forms the basis of a decision on whether or not something is satisfactory. Various kinds of analysis may be used as tools in doing this. Hence an assessment may include a number of analyses.

Safety case

A collection of arguments and evidence in support of the safety of a facility or activity. This will normally include the findings of a safety assessment and a statement of confidence in these findings.

Safety policy

A documented commitment by the licensee to a high nuclear safety performance supported by clear safety objectives and targets and a commitment of necessary resources to achieve these targets. The safety policy is issued as separate safety management document or as visible part of an integrated organisation policy.

Structures, systems and components (SSCs)

A general term encompassing all of the elements (items) of a facility or activity which contribute to protection and safety, except human factors.

- **Structures** are the passive elements: buildings, vessels, shielding, etc.
- A **system** comprises several **components**, assembled in such a way as to perform a specific (active) function.
- A **component** is a discrete element of a system.

Use

Authorized use: Use of radioactive materials or radioactive objects from an authorized practice in accordance with an authorization.

Restricted use: The use of an area or of materials, subject to restrictions imposed for reasons of radiation protection and safety. Restrictions would typically be expressed in the form of prohibition of particular activities (e.g. materials may only be recycled or reused within a facility).

Unrestricted use: The use of an area or of materials without any radiologically based restrictions.

List of Abbreviations

EU	European Union
IAEA	International Atomic Energy Agency
OLC	operational limits and conditions
RHWG	(WENRA) Reactor Harmonisation Working Group
SSCs	structures, systems and components
SRLs	safety reference level
WENRA	Western European Nuclear Regulators Association
WGWD	(WENRA) Working Group on Waste and Decommissioning

Part I.

Introduction and Used Methodology

1. Introduction

This version 2.0 of the Decommissioning Safety Reference Level Report is the result of an effort by the Working Group on Waste and Decommissioning of WENRA, from 2009 to 2011 to improve the version 1.0 of March 2007. The improvement is based on lessons learned from the benchmarking processes for version 1.0, especially on the implementation of the reference levels in the national legal and regulatory framework. Version 2.0 presents the safety reference levels (SRLs) for decommissioned facilities that are thought to be a good basis for future harmonisation on a European level.

The SRLs can not be considered as independent European safety requirements because current legislation in WENRA member states would not allow that due to fundamental differences reflecting the historical development in European countries. The SRLs are a set of requirements against which the situation of each country is assessed and it is each country's responsibility to implement actions to ensure that these levels are reached.

1.1. Background

WENRA, which has been established in February 1999, is the association of the Heads of nuclear regulatory authorities of European countries with at least one nuclear power plant in construction, operation or decommissioning phase. WENRA has been formally extended in 2003 to include future new European Union (EU) Member States. Currently following countries are members of WENRA: Belgium, Bulgaria, the Czech Republic, Finland, France, Germany, Hungary, Italy, Lithuania, the Netherlands, Romania, Slovenia, Slovakia, Spain, Sweden, Switzerland and the United Kingdom.

The original objectives of the Association were:

- to develop a common approach to nuclear safety and regulation, in particular within the EU,
- to provide the EU with an independent capability to examine nuclear safety and regulation in candidate countries,
- to evaluate and achieve a common approach to nuclear safety and regulatory issues which arise.

The second objective of WENRA has been fulfilled by the preparation of a report on nuclear safety in candidate countries having at least one nuclear power plant. After 1 May 2004, when most of these candidate countries became a regular EU Member States, the new WENRA tasks, based on first and third original Association's objectives, became:

- to develop an independent nuclear safety assessment capability, based on in-depth knowledge of nuclear installations, and
- to develop common approaches to nuclear safety and regulations and to encourage the harmonisation of practices.

To perform these tasks two working groups within the WENRA have been established - Reactor Harmonisation Working Group (RHWG) and Working Group on Waste and Decommissioning (WGWD). The work of WGWD has started in 2002.

1.2. Objective

The term “decommissioning” refers to administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a nuclear facility other than a repository. These actions involve decontamination, dismantling and removal of radioactive materials, waste, components and structures. They are carried out to achieve a progressive and systematic reduction in radiological hazards.

This report provides harmonised safety reference levels applicable during design, construction, operation and decommissioning of a nuclear facility to ensure a safe decommissioning process.

Although the SRLs in this report are oriented toward the licensees they can also be used by the regulatory body for the review and assessment of decommissioning activities safety.

These SRL constitute the basis for a common approach to nuclear safety during decommissioning in the WENRA Member States and, based on national action plans, should be implemented in the legal and regulatory framework system of each Member State by end 2013.

1.3. Scope

The decommissioning SRLs apply for nuclear reactors (of any power), fuel reprocessing facilities, fuel manufacturing facilities, uranium concentration and conversion facilities, uranium enrichment facilities, research facilities involving nuclear material. They may also be applied for waste storage facilities and other waste management facilities. These reference levels are not intended to be applicable to uranium mining and milling, and for isotope production facilities other than reactors.

The point at which decommissioning starts will vary from country to country depending on national arrangements, ranging from the decision on shutdown the facility up to the begin of dismantling activities.

For the purposes of this document, is assumed that the normal operational phase includes the removal of the bulk of fuel and radioactive materials from the facility in accordance with the safety case for normal operations. In certain cases part of the nuclear inventory of a facility is only removed after the start of decommissioning activities. In such case appropriate SRLs (e.g. for criticality control) for the operational phase of the facility remain applicable. The decommissioning phase is assumed to start technically once further operations cannot be carried out using normal operational methods or within the bounds of the safety case for normal operation. The decommissioning phase is usually governed by a specific decommissioning license.

The decommissioning SRLs address mainly the radiological hazards resulting from the activities associated with the decommissioning of facilities, primarily with decommissioning after a planned shutdown. Non-radiological hazards can also arise during decommissioning activities. These hazards should be given due consideration during the planning process and in the risk analyses as far as they may influence the radiological hazards or risks.

Regulatory requirements for Environmental Impact Assessment (required by EU directives), waste disposal, conventional occupational health and safety, physical protection and decommissioning funding, are important for decommissioning. Aspects on waste disposal are addressed in a new Safety Reference Levels Report of the WGWD. The other matters are not always regulated by the WENRA members, but are addressed by other national regulatory organisations. As a result, WGWD did not take into account in detail these matters and has therefore concentrated on the nuclear safety requirements.

As this document is intended to cover a wide range of sites and facilities (from small isolated nuclear facility to large complex reprocessing or reactor sites), the reference levels will need to be implemented in different ways to be appropriate for the particular facility, taking into account the magnitude of the hazard in a graded approach. In accordance with that graded approach, the decommissioning strategies and plans necessary to ensure safety need to be commensurate with the type and status of the facility and the hazards associated with the decommissioning of the facility.

It should be noted, that some SRLs from other WENRA reports need to be considered during decommissioning, if spent fuel is still in the nuclear facility during decommissioning or storages for spent fuel or radioactive waste are part of the decommissioning project. Vice versa, some of the decommissioning related SRL shall be considered during construction and operation of nuclear power plants and storage facilities and as such complement the related reports on safety reference levels.

1.4. Structure

The report consists of two main parts.

Following this Introduction in Part I of the report, Section I.-2 presents the general methodology that was followed to develop the version 2.0 of the SRLs.

Part II of the report presents the actual decommissioning safety reference levels.

2. Methodology

The objective of this report is to provide safety reference levels for decommissioning activities. This document contains the results of the work of WGWD in the area of the decommissioning of nuclear installations performed from 2009 to 2011 to improve the Decommissioning Safety Reference Level Report, version 1.0 of March 2007.

This document is based on corresponding IAEA documents (requirements, guidances, etc), that establish an essential basis for safety of all nuclear installations covering also their decommissioning. The WENRA safety reference levels incorporate most of these specific requirements and recommendations stated in version 1.0 taking into account lessons learned from the national benchmarking processes, especially on the implementation of the safety reference levels in the national legal and regulatory framework and feedback from stakeholders.

Part II.

Decommissioning Safety Reference Levels

1. Safety area: Safety management

This safety area covers selected elements of a management system as required by GS-R-3. Especially safety issue 1.4 covers some requirements on the implementation of a safety management system.

1.1. Safety issue: Responsibility

D-01: A licensee[1] shall be responsible for all aspects of nuclear safety on the facility. The continuity of responsibility shall be ensured throughout operation and decommissioning.

*Related **IAEA** safety standards:*

... The operating organization shall also be responsible for all aspects of safety and environmental protection during the decommissioning activities.... (WS-R-5, para. 3.7)

D-02: To fulfil its prime responsibility for safety during decommissioning of the facility, the licensee shall establish and implement safety policies and ensure that safety issues are given the highest priority.

*Related **IAEA** safety standards:*

To fulfil its prime responsibility for safety throughout the lifetime of a fuel cycle facility, the operating organization shall establish, implement, assess and continually improve a management system that integrates safety, health, environmental, security, quality and economic elements to ensure that safety is properly taken into account in all the activities of an organization. (NS-R-5, para 4.1)

The operating organization shall establish and implement safety, health and environmental policies in accordance with national and international standards and shall ensure that these matters are given the highest priority (NS-R-5, para 4.2)

D-03: The ultimate responsibility for safety shall remain with the licensee, although it is permissible to delegate the performance of specific tasks to subcontractors. The licensee shall ensure that the work of contractors is appropriately controlled so that it is conducted safely.

*Related **IAEA** safety standards:*

The ultimate responsibility for safety shall remain with the operating organization, although it is permissible to delegate the performance of specific tasks to a subcontractor. The decommissioning management shall ensure that the work of contractors is appropriately controlled so that it is conducted safely. ... (WS-R-5, para. 7.2)

D-04: In accordance with the national system the licensee or the owner shall provide financial assurances and resources to cover the costs associated with safe decommissioning, including management of resulting radioactive waste.

*Related **IAEA** safety standards:*

[1] Covers the possible change of licensee

... The operating organization shall provide financial assurances and resources to cover the costs associated with safe decommissioning, including management of the resulting radioactive waste. (WS-R-5, para 3.7)

1.2. Safety issue: Organisational structure

D-05: The licensee shall establish an organizational structure for the management and implementation of decommissioning, with the responsibility to ensure that decommissioning will be conducted safely.

Related ***IAEA*** *safety standards:*

An organization for the management and implementation of decommissioning shall be established as part of the operating organization, with the responsibility for ensuring that decommissioning will be conducted safely. ... (WS-R-5, para 7.1)

D-06: The licensee shall assess the adequacy of the organisational structure, for safe and reliable decommissioning of the facility, and for ensuring an appropriate response in emergencies, on a regular basis and in particular, if there is a major change in the plant state or hazard.

Related ***IAEA*** *safety standards:*

A safety assessment should form an integral part of the decommissioning plan. The operating organization is responsible for preparing the safety assessment and submitting it for review by the regulatory body. The safety assessment should be commensurate with the complexity and potential hazard of the installation and, in case of deferred decommissioning, should take into account the safety of the installation during the period leading up to final dismantling. (WS-G-2.1, para 5.3)

In order to control all decommissioning activities, the operating organization should implement an effective management control system. This should include control of preparatory decommissioning activities (such as the installation of new safety systems) and recognition of the risks associated with the changing conditions that arise during decommissioning. (WS-G-2.4, para 7.7)

Administrative measures from the operational phase of the facility may be relevant to the decommissioning. These measures should be reviewed and modified to ensure that they are appropriate and, if necessary, additional administrative measures should be taken. … (WS-G-2.4, para 7.9)

D-07: The licensee shall ensure that there is a clear allocation of authorities and responsibilities, together with the interfaces and communication routes that will be used especially when contractors or outside organizations are used.

Related ***IAEA*** *safety standards:*

… There should be a clear delineation of authorities and responsibilities, together with the interfaces and communication routes that will be used. This is particularly important when contractors or outside organizations are used. (WS-G-2.4, para 7.6)

D-08: The licensee shall evaluate the skills needed for safe decommissioning and shall determine the minimum number and qualification requirements of staff responsible for safety.

Related ***IAEA*** *safety standards:*

The skills needed for decommissioning shall be evaluated and the minimum requirements for qualifications of staff in each position shall be established. ... (WS-R-4, para 7.3)

Decommissioning may be carried out in a sequence of operations separated by one or more periods of time (i.e. phased decommissioning). Some of these periods (i.e. decommissioning phases) may consist of inactive, safe enclosure. In such cases of multiple decommissioning phases, the operating organization should submit to the regulatory body a description of:

> *(a)….*
> …
> *(e) the number of staff needed and their qualifications, during any period of deferment. (WS-G-2.1, para. 5.12)*

1.3. Safety issue: Record and knowledge keeping

D-09: The licensee shall ensure that sufficient knowledge of the facility and technical expertise is maintained during life time of the facility.
The licensee shall ensure that appropriate records and reports that are relevant to decommissioning (e.g. records on the use of the facility, events and incidents, radionuclide inventories, dose rates and contamination levels) shall be retained during life time of the facility. In this way, the design and modifications of the facility and its operating history will be identified and factored into the decommissioning plan.

Related ***IAEA*** *safety standards:*

Provision shall be made, as far as possible, to ensure that key staff are retained and that institutional knowledge about the facility is maintained and is accessible. Appropriate records and reports that are relevant to decommissioning (e.g. records on the use of the facility, events and incidents, radionuclide inventories, dose rates and contamination levels) shall be retained during the lifetime of the facility. In this way the design and modifications of the facility and its operating history will be identified and factored into the decommissioning plan. (WS-R-5, para 5.9)

D-10: The licensee shall maintain an appropriate record system to ensure, before decommissioning, that the radioactive material contained in the facility at the end of the operational phase is accounted for. During decommissioning, this record system shall include an up-to-date inventory of the radioactive material contained in the facility.

Related ***IAEA*** *safety standards:*

Relevant documents and records shall be prepared by the operating organization, shall be kept for an agreed time and shall be maintained to a specified quality by appropriate parties before, during and after decommissioning. (WS-R-5, para 7.6)

1.4. Safety issue: Implementation of a management system

D-11: The licensee shall establish, implement, assess and continually improve a management system. It shall be aligned with the goals of the organization and shall contribute to their achievement. The main aim of the management system shall be to achieve and enhance safety by:

- Bringing together in a coherent manner all the requirements for managing the organization;

- Describing the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied;
- Ensuring that health, environmental, security, quality and economic requirements are not considered separately from safety requirements, to help preclude their possible negative impact on safety.

*Related **IAEA** safety standards:*

A management system shall be established, implemented, assessed and continually improved. It shall be aligned with the goals of the organization and shall contribute to their achievement. The main aim of the management system shall be to achieve and enhance safety by:

- *Bringing together in a coherent manner all the requirements for managing the organization;*
- *Describing the planned and systematic actions necessary to provide adequate confidence that all these requirements are satisfied;*
- *Ensuring that health, environmental, security, quality and economic requirements are not considered separately from safety requirements, to help preclude their possible negative impact on safety. (GS-R-3; para 2.1, also cited in GS-G-3.3, para 2.1)*

Leadership in safety matters has to be demonstrated at the highest levels in an organization. Safety has to be achieved and maintained by means of an effective management system. This system has to integrate all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security, and so that safety is not compromised by other requirements or demands. The management system also has to ensure the promotion of a safety culture, the regular assessment of safety performance and the application of lessons learned from experience. (SF-1, principle 3, para 3.12)

D-12: The licensee shall ensure that the management system is applied to all phases of decommissioning taking into account the continuous change during decommissioning.

*Related **IAEA** safety standards:*

7.7. A comprehensive quality assurance programme under the operating organization's management system [7] shall be applied to all phases of decommissioning ... (WS-R-5, para 7.7)

D-13: The licensee shall ensure, that processes of the management system that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the organization are identified, and their development are planned, implemented, assessed and continually improved. The work performed in each process shall be carried out under controlled conditions, by using approved current procedures, instructions, drawings or other appropriate means that are periodically reviewed to ensure their adequacy and effectiveness.

*Related **IAEA** safety standards:*

The processes of the management system that are needed to achieve the goals, provide the means to meet all requirements and deliver the products of the organization shall be identified, and their development shall be planned, implemented, assessed and continually improved. (GS-R-3; para 5.1)

The work performed in each process shall be carried out under controlled conditions, by using approved current procedures, instructions, drawings or other appropriate means that are periodically reviewed to ensure their adequacy and effectiveness. (GS-R-3; para 5.9)

D-14: The licensee shall ensure that the documentation of the management system includes the following:

- The policy statements of the licensee;
- A description of the management system;
- A description of the organisational structure of the licensee;
- A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
- A description of the interactions with relevant external organisations;
- A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved.

*Related **IAEA** safety standards:*

The documentation of the management system shall include the following:

- *The policy statements of the organization;*
- *A description of the management system;*
- *A description of the structure of the organization;*
- *A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;*
- *A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, assessed and improved. (GS-R-3; para. 2.8)*

2. Safety area: Decommissioning strategy and planning

2.1. Safety issue: Facilitating decommissioning during design, construction and operational phase

D-15: The licensee shall take account of the need to decommission a facility at the time it is being planned, designed, constructed and operated. Measures, including design features, contamination and activation control, shall be described and justified.

Related ***IAEA*** *safety standards:*

The responsibilities of the operating organization include:

- *Establishing a decommissioning strategy and preparing and maintaining a decommissioning plan throughout the lifetime of the facility;*
- *... (WS-R-5, para 3.8)*

D-16: The licensee shall undertake a baseline survey, including radiological conditions of the site before construction, for comparison with the proposed end-state after decommissioning. For those practices for which such a baseline survey has not been done in the past, data from analogous, undisturbed areas with similar characteristics can be used instead of pre-operational baseline data.

Related ***IAEA*** *safety standards:*

A baseline survey of the site, including obtaining information on radiological conditions, shall be performed prior to construction and updated prior to commissioning of a new facility. This information will be used to determine background conditions during the end state survey. For those practices for which such a baseline survey has not been done in the past, data from analogous, undisturbed areas with similar characteristics shall be used instead of pre-operational baseline data. (WS-R-5, para 5.8)

2.2. Safety issue: Decommissioning strategy

D-17: The licensee shall establish a decommissioning strategy for its facility. This decommissioning strategy shall be consistent with existing related national strategies and regulatory requirements, e. g. on decommissioning or radioactive waste management.

Related ***IAEA*** *safety standards:*

The responsibilities of the operating organization include:

- *Establishing a decommissioning strategy and preparing and maintaining a decommissioning plan throughout the lifetime of the facility;*
- *... (WS-R-5, para 3.8)*

... The strategy shall be consistent with national decommissioning and waste management policy. (WS-R-5, para 4.1)

D-18: The decommissioning strategy shall be documented including a description of the options, overall timescales for the decommissioning of the facility and the end-state after completion of all decommissioning activities. The report shall explain the reasons for the preferred option, and options not involving immediate dismantling shall be rigorously justified.

*Related **IAEA** safety standards:*

The preferred decommissioning strategy shall be immediate dismantling. There may, however, be situations where immediate dismantling is not a practical strategy when all relevant factors are considered. These factors may include: the availability of waste disposal or long term storage capacity for decommissioning waste; the availability of a trained workforce; the availability of funds; co-location of other facilities on the same site requiring decommissioning; technical feasibility; and optimization of the radiation protection of workers, the public and the environment. If the deferred dismantling or entombment strategy is chosen, the operating organization shall provide a justification for the selection. The operating organization shall also demonstrate that, for the selected strategy, the facility will be maintained in a safe configuration at all times and will be adequately decommissioned in the future and that no undue burdens will be imposed on future generations. (WS-R-5, para 4.2)

2.3. Safety issue: Facility decommissioning plan during design, construction and operational phases

D-19: Based on the established decommissioning strategy the licensee shall establish an initial decommissioning plan for the facility. The details of the plan shall be commensurate with the type and status of the facility (graded approach).

*Related **IAEA** safety standards:*

The responsibilities of the operating organization include:

- *Establishing a decommissioning strategy and preparing and maintaining a decommissioning plan throughout the lifetime of the facility;*
- *... (WS-R-5, para 3.8)*

D-20: The licensee shall submit the initial decommissioning plan to the regulatory body in support of the licence application for construction for a new facility.

*Related **IAEA** safety standards:*

An initial plan for decommissioning should be prepared and submitted by the operating organization in support of the licence application for the construction of a new reactor. Although the level of detail in the initial plan will necessarily be lower than that in the final decommissioning plan, many of the aspects listed in para. 5.11 should be considered in a conceptual fashion. A generic study showing the feasibility of decommissioning may suffice for this plan, particularly in standardized installations. Depending on applicable regulations, the plan should address the costs and the means of financing the decommissioning work. (WS-G-2.1, para 5.6)

D-21: The initial decommissioning plan shall:

(a) take into account major safety issues;
(b) support the fact that decommissioning can be safely conducted using proven techniques or ones being developed;
(c) include a generic study showing the feasibility of decommissioning;
(d) include consideration of environmental aspects of decommissioning, such as management of waste and radioactive effluents;
(e) provide a basis to assess the costs of the decommissioning work and the means of financing it.

Related ***IAEA*** *safety standards:*
An initial plan for decommissioning shall be prepared which outlines the overall decommissioning process (Ref. [2], para. 3.13). This plan should be submitted by the operating organization to the regulatory body in support of the licence application for commissioning and/or operating the facility. This plan:
(a) Should take into account basic safety issues;
(b) Should support the fact that decommissioning can be safely conducted using proven techniques or ones being developed;
(c) Should include a generic study showing the feasibility of decommissioning;
(d) Should include consideration of environmental aspects of decommissioning, such as management of waste and radioactive effluents;
(e) Should address the costs of the decommissioning work and the means of financing it. (WS-G-2.4, para 5.6).

D-22: If several facilities are located at the same site it shall be ensured that in each facility decommissioning plan any interactions and interdependencies between the facilities are taken into account.

Related ***IAEA*** *safety standards:*

For sites that house more than one facility, a global decommissioning programme shall be developed for the entire site to ensure that interdependences are taken into account in the planning for individual facilities. (WS-R-5, para 4.8)

D-23: During operation the decommissioning plan shall be reviewed by the licensee regularly, at least as frequently as the periodic safety review, and shall be updated as required. These reviews of the decommissioning plan shall consider, in particular, changes in the facility operation experiences or regulatory requirements, and advances in technology to further evolve the decommissioning plan.

Related ***IAEA*** *safety standards:*

During the operation of a reactor, the decommissioning plan should be reviewed, updated and made more comprehensive with respect to technological developments in decommissioning, incidents that may have occurred, including abnormal events, amendments in regulations and government policy, and, where applicable, cost estimates and financial provisions. …. All significant systems and structural changes during plant operation should be reflected in the process of ongoing planning for decommissioning. (WS-G-2.1, para 5.8)

D-24: The decommissioning plan shall be supported by an appropriate safety assessment for the decommissioning activities the details of which are commensurate with the type and status of the facility (graded approach).

Related ***IAEA*** *safety standards:*

…. The decommissioning plan should evolve with respect to safety considerations, based on operational experience and on information reflecting improved technology. All significant systems and structural changes during plant operation should be reflected in the process of ongoing planning for decommissioning. (WS-G-2.1, para 5.8)

The decommissioning plan shall be supported by an appropriate safety assessment covering the planned decommissioning activities and abnormal events that may occur during decommissioning. The assessment shall address occupational exposures and potential releases of radioactive substances with resulting exposure of the public. (WS-R-5, para 5.2)

D-25: The decommissioning plan shall identify existing systems and equipment that will be used during decommissioning to ensure that they are available when needed. The decommissioning plan shall also identify necessary changes or replacements of existing systems. The decommissioning plan shall also identify the need for new facilities to carry out decommissioning and waste management.

*Related **IAEA** safety standards:*

The existing facilities and equipment that will be used during decommissioning should be identified at an early stage in the initial planning phase. This will enable the necessary steps to be taken to ensure that the equipment is available when needed. (WS-G-2.4, para 5.7)

2.4. Safety issue: Final decommissioning plan[2]

D-26: As soon as it has been decided to permanently shut down a nuclear facility, the licensee shall inform the regulatory body.

*Related **IAEA** safety standards:*

The operating organization shall inform the regulatory body prior to shutting down the facility permanently. ... (WS-R-5, para 8.2)

D-27: If a facility is shut down and no longer used for its intended purpose, a final decommissioning plan shall be submitted to the regulatory body not later than two years after the shut down of the facility, unless an alternative schedule for the submission of the final decommissioning plan is specifically authorized by the regulatory body.

*Related **IAEA** safety standards:*

... If a facility is shut down and no longer used for its intended purpose, a final decommissioning plan[5)] shall be submitted for approval within two years of the cessation of the authorized activities, unless an alternative schedule for the submission of the final decommissioning plan is specifically authorized by the regulatory body. The operating organization shall not implement the decommissioning plan until the regulatory body has approved it. Any amendments to this plan shall also be submitted to the regulatory body for approval. The operating organization shall ensure that the facility is maintained in a safe configuration until the approval of the decommissioning plan. (WS-R-5, para 8.2)

[5)] The final decommissioning plan is that version of the decommissioning plan submitted for approval to the regulatory body prior to implementation of the plan. During implementation of this final plan revisions or amendments may subsequently be needed as the activity progresses.

D-28: A final decommissioning plan shall
- be consistent with the decommissioning strategy proposed for the facility,
- be consistent with the safety case for decommissioning (ref. D-50),
- describe the decommissioning activities, including the timeframe and the end-state of the decommissioning project, and the content of the individual phases, if a phased approach is applied,

[2] For explanations on the relation between the final decommissioning plan and the safety case for decommissioning please refer to appendix C.

- describe the facilities, systems and equipment needed to perform the decommissioning project,
- describe the human resources required for safe decommissioning,
- describe the management of residual material and waste in accordance with the national waste strategy,
- describe principles of the management system used, and
- describe the program of the final radiation survey of the end-state of decommissioning.

Related ***IAEA*** *safety standards:*

Prior to the implementation phase of decommissioning activities, a final decommissioning plan shall be prepared and submitted to the regulatory body for approval. This plan shall define how the project will be managed, including: the site management plan, the roles and responsibilities of the organizations involved, safety and radiation protection measures, quality assurance, a waste management plan, documentation and record keeping requirements, a safety assessment and an environmental assessment and their criteria, surveillance measures during the implementation phase, physical protection measures as required, and any other requirements established by the regulatory body. (WS-R-5, para 5.10)

The safety assessment for decommissioning should be consistent with the decommissioning plan [1, 9–11] and with other relevant national and site specific strategies and requirements, for example, with requirements for radioactive waste management and for the release of material and sites from regulatory control. (WS-G-5.2, para 2.2)

The experience from previous decommissioning should be appropriately taken into account as a matter of principle. The following list of items to be considered for the final decommissioning plan should thus be updated whenever previous decommissioning experience permits:
(a) a description of the nuclear reactor, the site and the surrounding area that could affect, and be affected by, decommissioning;
(b) the life history of the nuclear reactor, reasons for taking it out of service, and the planned use of the nuclear installation and the site during and after decommissioning;
(c) a description of the legal and regulatory framework within which decommissioning will be carried out;
(d) explicit requirements for appropriate radiological criteria for guiding decommissioning;
(e) a description of the proposed decommissioning activities, including a time schedule;
(f) the rationale for the preferred decommissioning option, if selected;
(g) safety assessments and environmental impact assessments, including the radiological and non-radiological hazards to workers, the public and the environment; this will include a description of the proposed radiation protection procedures to be used during decommissioning;
(h) a description of the proposed environmental monitoring programme to be implemented during decommissioning;
(i) a description of the experience, resources, responsibilities and structure of the decommissioning organization, including the technical qualification/skills of the staff;
(j) an assessment of the availability of special services, engineering and decommissioning techniques required, including any decontamination, dismantling and cutting technology as well as remotely operated equipment needed to complete decommissioning safely;
(k) a description of the quality assurance programme;
(l) an assessment of the amount, type and location of residual radioactive and hazardous non-radioactive materials in the nuclear reactor installation, including calculational methods and measurements used to determine the inventory of each;
(m a description of the waste management practices, including items such as:
— identification and characterization of sources, types and volumes of waste;
— criteria for segregating materials;
— proposed treatment, conditioning, transport, storage and disposal methods;
— the potential to reuse and recycle materials, and related criteria; and
— anticipated discharges of radioactive and hazardous non-radioactive materials to the environment;
(n) a description of other applicable important technical and administrative considerations such as safeguards, physical security arrangements and details of emergency preparedness;
(o) a description of the monitoring programme, equipment and methods to be used to verify that the site will comply with the release criteria;

(p) details of the estimated cost of decommissioning, including waste management, and the source of funds required to carry out the work; and

(q) a provision for performing a final confirmatory radiological survey at the end of decommissioning. (WS-G-2.1, para 5.11)

2.5. Safety issue: Decommissioning plan update during decommissioning operations

D-29: Depending on the timeframe of decommissioning, the decommissioning plan shall be reviewed regularly by the licensee during decommissioning operations, and shall be updated as required. These updates of the decommissioning plan are to reflect, in particular, changes in the decommissioning strategy, deviations from the scheduled program, experiences from ongoing decommissioning or changes of regulatory requirements and advances in technology.

*Related **IAEA** safety standards:*

The decommissioning plan shall be reviewed regularly and shall be updated as required to reflect, in particular, changes in the facility or regulatory requirements, advances in technology and, finally, the needs of the decommissioning operation. If an abnormal event occurs, a new decommissioning plan or modification of the existing decommissioning plan may be necessary. (WS-R-2. para 6.3).

During the implementation of the decommissioning plan, revisions or amendments may need to be made to the plan in the light of operational experience gained, new or revised safety requirements, or technological developments. (WS-R-2, para 6.4).

3. Safety area: Conduct of decommissioning

3.1. Safety issue: Safety classification

D-30: SSCs may be re-classified as they change in importance to safety in the course of decommissioning activities. The licensee shall reflect this re-classification in the safety case.

*Related **IAEA** safety standards:*

As part of the safety assessment, safety functions and their associated SSCs should be identified, both for planned decommissioning activities and for accident conditions, and their suitability and sufficiency should be demonstrated. The safety functions required to be fulfilled during decommissioning comprise a combination of safety functions that were needed during operation of the facility and additional functions that will be needed as a result of the specific decommissioning activities proposed (e.g. fire detection and suppression during cutting and grinding activities). The effects of decommissioning on the safety functions at adjacent facilities should also be evaluated. In addition, dismantling of major facility structures during decommissioning may involve the deliberate destruction and removal of engineered SSCs that had fulfilled specified safety functions during operation of the facility (e.g. containment, shielding, ventilation, cooling). If these safety functions are still required, the associated SSCs should be maintained in an appropriate state during decommissioning. If this is not practicable, these functions should be provided by suitable alternative means (e.g. tents, temporary facilities, fire systems, electrical systems, administrative procedures) for as long as is required on the basis of the safety assessment. The appropriateness of alternative means of fulfilling these functions should be demonstrated. Any change of safety functions during decommissioning should be justified in advance before its implementation. (WS-G-5.2, para 3.14)

Non-radiological as well as radiological hazards associated with the decommissioning activities should be identified and evaluated in the safety assessment. As a result of this assessment, the protective measures can be defined that will ensure that the regulatory requirements are met. These protective measures may require changes to the existing safety systems that were used during operation. The acceptability of such changes should be clearly justified in the safety assessment. … (WS-G-2.4, para. 5.14)

3.2. Safety issue: On-site emergency preparedness

If for the set of foreseeable accidents considered in the safety case, events requiring protective measures cannot be excluded, planned emergency arrangements will be required. These emergency plans should be proportionate taking account of the magnitude of the accident consequence. For some facilities (such as with low radioactive inventory) an off-site emergency plan may not be required, which must be justified and the off-site aspects of this safety issue will not apply. This site emergency plan can be based on the operational one but modified according to changed hazards during the decommissioning actions. The following SRLs therefore need to be applied in a proportionate manner.

D-31: The licensee shall provide arrangements for responding effectively to reasonably foreseeable events requiring measures at the scene for:

(a) regaining control of any emergency arising at the site, including events related to combinations of non-nuclear and nuclear hazards;

(b) preventing or mitigating the consequences at the scene of any such emergency and

(c) co-operating with external emergency response organizations in preventing adverse health effects in workers and the public.

*Related **IAEA** safety standards:*

Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents. (SF-1, Principle 9)
The primary goals of preparedness and response for a nuclear or radiation emergency are:
- *To ensure that arrangements are in place for an effective response at the scene and, as appropriate, at the local, regional, national and international levels, to a nuclear or radiation emergency;*
- *To ensure that, for reasonably foreseeable incidents, radiation risks would be minor;*
- *For any incidents that do occur, to take practical measures to mitigate any consequences for human life and health and the environment. (SF-1; para 3.34)*

Emergency preparedness and response arrangements commensurate with the threat category of the facility, […], should be developed and implemented. (WS-G-6.1, para 5.14)

Emergency planning arrangements, commensurate with the hazards, shall be established and maintained and incidents significant to safety shall be reported to the regulatory body in a timely manner. Additional requirements for preparedness and response to emergencies are established in another IAEA publication [8]. (WS-R-5, para 8.7)

A programme for emergency planning shall be established (Ref. [2], para. 3.14) and described in the decommissioning plan. This programme should be subject to approval by the regulatory body. Operating organizations should ensure that procedures to deal with unforeseen events are prepared and are put in place. Personnel should be trained in emergency procedures. Provision should be made for regular testing and updating of these procedures by conducting exercises periodically. (WS-G-2.4, para 7.27)

D-32: The licensee shall
- prepare an on-site emergency plan as the basis for preparation and conduct of emergency measures,
- establish the necessary organizational structure for clear allocation of responsibilities, authorities and arrangements for coordinating on-site activities and cooperating with external response agencies throughout all phases of an emergency and
- ensure that, based on the on-site emergency plan trained and qualified personnel, facilities and equipment needed to control an emergency are appropriate, reliable and available at the time.

*Related **IAEA** safety standards:*

The appropriate responsible authorities shall ensure that:
(a) emergency plans [are] prepared and approved for any practice or source which could give rise to a need for emergency intervention;
(b) [response organizations are] involved in the preparation of emergency plans, as appropriate;
(c) the content, features and extent of emergency plans take into account the results of any [threat assessment] and any lessons learned from operating experience and from [emergencies] that have occurred with sources of a similar type […];
(d) emergency plans [are] periodically reviewed and updated." […] (GS-R-2, para 5.17)

Adequate tools, instruments, supplies, equipment, communication systems, facilities and documentation (such as procedures, checklists, telephone numbers and manuals) shall be provided for performing the functions specified in Section 478. These items and facilities shall be selected or designed to be operational under the postulated conditions (such as the radiological, working and environmental conditions) that may be encountered in the emergency response, and to be compatible with other procedures and equipment for the response (such as the communication frequencies of other response organizations), as appropriate. These support items shall be located or provided in a manner that allows their effective use under postulated emergency conditions. (GS-R-2, para 5.25)

The operator and the response organizations shall identify the knowledge, skills and abilities necessary to be able to perform the functions specified […]. The operator and the response organizations shall make arrangements for the selection of personnel and for training to ensure that the personnel have the requisite knowledge, skills, abilities, equipment, and procedures and other arrangements to perform their assigned *response functions. The arrangements shall include ongoing refresher training on an appropriate schedule and the results of which arrangements for ensuring that personnel assigned to positions with responsibilities for emergency response undergo the specified training. (GS-R-2, para 5.31)*

D-33: During decommissioning, the licensee shall review and update as necessary the existing on-site emergency plan, so that it stays appropriate for current and future states of the facility. Experience from recent emergency exercises and reports on real emergency occurrences shall be taken into account.

Related ***IAEA*** *safety standards:*

Emergency planning arrangements, commensurate with the hazards, shall be established and maintained and incidents significant to safety shall be reported to the regulatory body in a timely manner. ... (WS-R-5, para 8.7)

"The operating organization [of a facility or practice in threat category I, II, III or IV] shall prepare an emergency plan that covers all activities under its responsibility, to be adhered to in the event of an emergency. This emergency plan shall be co-ordinated with those of all other bodies having responsibilities in an emergency, including public authorities, and shall be submitted to the regulatory body." (Ref. [12], para. 2.31.) (NS-R-2, para 5.19)

D-34: The licensee shall perform at regular intervals on-site emergency exercises, the results of which shall be reported to the regulatory body. Some of these exercises shall include the participation to the extent possible of external organizations concerned with on-site emergency.

Related ***IAEA*** *safety standards:*

In developing the emergency response arrangements, consideration has to be given to all reasonably foreseeable events. Emergency plans have to be exercised periodically to ensure the preparedness of the organizations having responsibilities in emergency response. (SF-1; para 3.37)

Exercise programmes shall be conducted to ensure that all specified functions required to be performed for emergency response and all organizational interfaces for facilities in threat category I, II or III and the national level programmes for threat category IV or V are tested at suitable intervals[84, 85]. These programmes shall include the participation in some exercises of as many as possible of the organizations concerned. The exercises shall be systematically evaluated and some exercises shall be evaluated by the regulatory body. The programme shall be subject to review and updating in the light of experience gained (see paras 3.8, 3.16, 5.37 and 5.39 for further requirements in relation to exercises). (GS-R-2, para 5.33)

3.3. Safety issue: Decommissioning experience feedback

D-35: The licensee shall establish and implement experience feedback arrangements to collect, screen, analyse and document experience and events at the facility in a systematic way to improve and ensure safe decommissioning. Relevant experience and events reported by other facilities shall also be considered as appropriate.

Related ***IAEA*** *safety standards:*

Leadership in safety matters has to be demonstrated at the highest levels in an organization. Safety has to be achieved and maintained by means of an effective management system. This system has to integrate all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security, and so that safety is not compromised by other requirements or demands. The management system also has to ensure the promotion of a safety culture, the *regular assessment of safety performance and the application of lessons learned from experience. (SF-1, para 3.12)*

The process of safety assessment for facilities and activities is repeated in whole or in part as necessary later in the conduct of operations in order to take into account changed circumstances (such as the application of new standards or scientific and technological developments), the feedback of operating experience, modifications and the effects of ageing. ... (SF-1, para 3.16)

... The feedback of operating experience from facilities and activities — and, where relevant, from elsewhere — is a key means of enhancing safety. Processes must be put in place for the feedback and analysis of operating experience, including initiating events, accident precursors, near misses, accidents and unauthorized acts, so that lessons may be learned, shared and acted upon. (SF-1, para 3.17)

D-36: The licensee shall ensure that results are obtained, that conclusions are drawn, measures are taken, good practices and advances in technology are considered and that timely and appropriate corrective actions are implemented to prevent recurrence and to counteract developments adverse to safety.

*Related **IAEA** safety standards:*

The causes of non-conformances shall be determined and remedial actions shall be taken to prevent their recurrence. (NS-R-3, para 6.11)

Products and processes that do not conform to the specified requirements shall be identified, segregated, controlled, recorded and reported to an appropriate level of management within the organization. The impact of non-conformances shall be evaluated and non-conforming products or processes shall be either:
- *—Accepted;*
- *—Reworked or corrected within a specified time period; or*
- *—Rejected and discarded or destroyed to prevent their inadvertent use. (NS-R-3, para 6.12)*

Corrective actions for eliminating non-conformances shall be determined and implemented. Preventive actions to eliminate the causes of potential non-conformances shall be determined and taken. (NS-R-3, para 6.14)

In many organizations there are several processes to control nonconforming products or processes, for example product inspections. The process or processes should include provisions to prevent the inadvertent use or installation of products or processes that do not conform and to ensure that effective corrective action is taken. (GS-G-3.1, para 6.50)

D-37: Following any abnormal event during decommissioning which is significant for safety the licensee shall carry out an investigation and implement corrective measures to prevent a recurrence and to recover an appropriate level of safety as defined by the safety case for decommissioning.

*Related **IAEA** safety standards:*

Following any abnormal event, the operating organization shall revalidate the safety functions and functional integrity of any component or system which may have been challenged by the event. Necessary remedial shall include inspection, testing and maintenance as appropriate (NS-R-2, 6.9)

3.4. Safety issue: Waste management

Waste from decommissioning shall be safely managed using appropriate routes with respect to their nature and characteristics that have to be determined as soon as possible. Procedures shall be implemented so that waste is segregated as soon as possible to avoid mixing of waste of different natures so as to optimize their management. Whenever categories of waste exit in the national waste management system, procedures shall be such that the waste is segregated in accordance with these categories.

D-38: The licensee shall develop, document and implement arrangements to characterise, segregate and manage the particularly large quantities and different types of radioactive waste and of other material that are produced during decommissioning, in accordance with the requirements set by the national regulatory authority and with the national waste management strategy.

*Related **IAEA** safety standards:*

Decommissioning of nuclear reactors invariably involves the generation of large amounts of radioactive wastes. In the course of decommissioning, waste will be generated in forms that are different from materials and wastes of the types routinely handled during the operational phase of a nuclear power plant or research reactor. Subject to safety considerations, "generation of radioactive waste shall be kept to the minimum practicable". …. (WS-G-2.1, para 2.20)

D-39: The licensee shall develop, document and implement optimized arrangements to segregate radioactive waste and reduce its volume in accordance with the requirements set by the national legal framework and with the national waste management strategy.

*Related **IAEA** safety standards:*

… For example, appropriate decontamination and dismantling techniques and the reuse or recycling of materials can reduce the waste inventory. (WS-G-2.1, para 2.20)

A large part of the waste and other materials arising during the decommissioning process may be sufficiently low in activity concentration for regulatory control to be wholly or partly removed. Some waste may be suitable for disposal in normal landfill sites, while some materials such as steel and concrete may be suitable for recycling or reuse outside the nuclear industry. The removal of regulatory controls should be accomplished in compliance with criteria established by the national regulatory body. (WS-G-2.4, para 7.21)

D-40: The licensee shall keep accurate records of any radioactive decommissioning waste and material removed from regulatory control. The records shall be kept in accordance with national records retention requirements.

*Related **IAEA** safety standards:*

On completion of decommissioning, appropriate records should be retained. In accordance with the national legal framework, these will be held and maintained for purposes such as confirmation of completion of decommissioning activities in accordance with the approved plan, recording the disposal of wastes, materials and premises, and responding to possible liability claims. (WS-G-2.1, para 8.1)

A final decommissioning report shall be prepared (Ref. [2], para. 6.13), on the basis of the records assembled, and should contain the following information:

(a) ...

(g) An inventory of radioactive materials, including amounts and types of waste generated during decommissioning and their locations for storage and/or disposal;

(h) An inventory of non-radioactive materials, including amounts and types of waste generated during decommissioning and their locations for storage and/or disposal;
(i) An inventory of materials, equipment and premises released from regulatory control;
(o) ... (WS-G-2.4, para 8.2)

3.5. Safety issue: On-site and off-site monitoring

D-41: Due to the changes of the facility, specific hazards and effluents associated with decommissioning, the licensee shall apply, review and modify as necessary its on- and off-site monitoring program.

*Related **IAEA** safety standards:*

The radiation protection programme should be clearly set out in the decommissioning plan. Those involved in its execution should be properly trained and have access to appropriate equipment for carrying out radiation surveys, including equipment for measuring external dose rates and surface contamination levels and for sampling air concentrations. (WS-G-2.1, para 7.14)

All decommissioning work should be planned and carried out using work order procedures and radiation work permits, with adequate involvement of radiation protection expertise to determine the required radiation protection measures. Moreover, the promotion of awareness of safety issues should be accorded high emphasis in planning and implementation. Those charged with the day to day responsibility for radiation protection should have the resources, access to decommissioning management and independence necessary to effect an adequate radiation protection programme. (WS-G-2.1, para 7.15)

The decommissioning plan should specify the requirement for on-site and off-site monitoring during decommissioning. On-site monitoring should provide information to identify and assist in mitigating the radiological hazards. It should also be used in the planning of specific decommissioning activities. It should ensure that all potential release points are monitored. On-site monitoring should consist not only of personnel monitoring but also of spatial monitoring for airborne contaminants, such as, having:
- *(a) appropriate monitoring equipment for dose rate and contamination surveys for workplaces, components and materials during decontamination, dismantling and handling;*
- *(b) appropriate monitoring protocols and equipment for packaging and handling of radioactive waste within the site, as well as for transportation of the waste offsite;*
- *(c) appropriate monitoring equipment for airborne contaminants;*
- *(d) appropriate monitoring equipment for timely screening of large quantities of low level radioactive material for clearance purposes; and*
- *(e) appropriate equipment and protocols to monitor the distribution of radionuclides in the installation. (WS-G-2.1, para 7.16)*

The off-site monitoring programme inherited from the operational period will require modification appropriate to the conditions existing during decommissioning. Discharges of radionuclides via airborne and liquid pathways should be controlled, monitored and recorded, as required by the regulatory body or other relevant competent authority. Relevant recommendations are provided in Refs [11, 12, 22]. (WS-G-2.1, para 7.17)

3.6. Safety issue: Maintenance, Testing and Inspection

D-42: The licensee shall prepare, and implement documented programmes for maintenance, testing, surveillance and inspection of SSCs and other equipment significant to safety to ensure that their availability, reliability and functionality remain in accordance with the safety case for decommissioning. The programmes shall take into account operational limits and conditions (OLCs) and be re-evaluated in the light of experience and the continuous changes of the facility during decommissioning.

*Related **IAEA** safety standards:*

The operating organization shall prepare and implement a programme of maintenance, testing, surveillance and inspection of those structures, systems and components which are important to safety. This programme shall be in place prior to fuel loading and shall be made available to the regulatory body. It shall take into account operational limits and conditions as well as any other applicable regulatory requirements and it shall be re-evaluated in the light of experience. (NS-R-2, para 6.1)

The maintenance, testing, surveillance and inspection of all plant structures, systems and components important to safety shall be to such a standard and at such a frequency as to ensure that their levels of reliability and effectiveness remain in accordance with the assumptions and intent of the design throughout the service life of the plant. (NS-R-2, para 6.2)

Effective maintenance, surveillance and inspection (MS&I) are essential for the safe operation of a nuclear power plant. They ensure not only that the levels of reliability and availability of all plant structures, systems and components (SSCs) that have a bearing on safety remain in accordance with the assumptions and intent of the design, but also that the safety of the plant is not adversely affected after the commencement of operation. (NS-G-2.6, para 1.1)

The maintenance programme for a nuclear power plant should cover all preventive and remedial measures, both administrative and technical, that are necessary to detect and mitigate degradation of a functioning SSC or to restore to an acceptable level the performance of design functions of a failed SSC. The purpose of maintenance activity is also to enhance the reliability of equipment. The range of maintenance activities includes servicing, overhaul, repair and replacement of parts, and often, as appropriate, testing, calibration and inspection. (NS-G-2.6, para 2.1)

D-43: The licensee shall address the ageing of SSCs and other equipment significant to safety by establishing, if necessary, provisions for their maintenance, testing and inspection.

*Related **IAEA** safety standards:*

The safety assessment in itself cannot achieve safety. Safety can only be achieved if the input assumptions are valid, the derived limits and conditions are implemented and maintained, and the assessment reflects the facility or activity as it actually is at any point in time. Facilities and activities change and evolve over their lifetimes (e.g. through construction, commissioning, operation, and decommissioning and dismantling or closure) and with modifications, improvements and effects of ageing. Knowledge and understanding also advance with time and experience. The safety assessment has to be updated to reflect such changes and to remain valid. Updating of the safety assessment is also important in order to provide a baseline for the future evaluation of monitoring data and performance indicators and, for facilities for the storage and disposal of radioactive waste, to provide an appropriate record for reference with regard to future use of the site. (GSR-Part 4, para 5.2)

Ageing management of SSCs important to safety should be implemented proactively (with foresight and anticipation) throughout the plant's lifetime, i.e. in design, fabrication and construction, commissioning, operation (including long term operation and extended shutdown) and decommissioning. (NS-G-2.12, para 3.1)

D-44: The licensee shall record, store, analyse and review data on maintenance, testing, surveillance, inspection of SSCs and other equipment relevant for safety. Where necessary corrective measures such as repair, replacement or changes in the maintenance programme shall be implemented.

*Related **IAEA** safety standards:*

Data on maintenance, testing, surveillance and inspection shall be recorded, stored and analysed to confirm that performance is in accordance with design assumptions and with expectations on equipment reliability. (NS-R-2, para 6.10)

The operating organization should monitor the performance or condition of SSCs against the goals it has set to provide reasonable assurance that the SSCs are capable of performing their intended function.(NS-G- 2.6, para 2.7)

A brief but complete review of the repairs carried out should be made and documented. This review should explicitly identify the cause of failure, the remedial action taken, the component that failed and its mode of failure, the total repair time and, if different, the outage time and, finally, the state of the system after repair. Even if a system is found to be within its calibration limits, this fact should be recorded, together with details of any replacement or any adjustment carried out at the discretion of maintenance personnel. (NS-G-2.6, para 5.32)

... For major failures of components important to safety, a root cause analysis should be carried out in order to prevent recurrence. (NS-G-2.6, para 8.47).

A common database should be established in order to share relevant data and evaluations of results among the organizations that are involved in the planning and implementation of MS&I activities. (NS-G-2.6, para 2.16)
An adequate condition monitoring programme should be established in support of optimisation of the maintenance programme. Such a monitoring programme should be based on the following assumptions as a minimum:
- *that the monitored parameters are appropriate indicators for the condition of the SSCs,*
- *that acceptance criteria are available,*
- *that all potential failure modes are addressed,*
- *that the behaviour of the potential failure is traceable and predictable. (NS-G-2.6, para 2.8)*

The maintenance group should periodically review the maintenance records for evidence of incipient or recurring failures. When a need for remedial maintenance is identified, either in this review or during preventive maintenance of the plant, the maintenance group should initiate remedial maintenance in accordance with the administrative procedures mentioned above. If appropriate, the preventive maintenance programme should be revised accordingly. (NS-G-2.6, 8.48)

3.7. Safety issue: Control of decommissioning activities

D-45: The licensee shall control decommissioning operations through the use of written and approved procedures. The licensee shall make and implement arrangements for issuing, modifying and terminating work procedures as part of the management system.

*Related **IAEA** safety standards:*

Decommissioning tasks shall be controlled through the use of written procedures. These procedures shall be subject to review and approval by the appropriate organizations responsible for ensuring safety and practicability. A methodology for issuing, modifying and terminating work procedures shall be established. (WS-R-5, para 7.5)

D-46: No decommissioning activity shall be undertaken without a prior assessment of its impact on safety taking into account the postulated initiating events with internal causes included in the safety case for decommissioning. Due consideration shall be given to different decommissioning activities executed in parallel which might adversely effect safety of each other.

D-47: The licensee shall control modifications of planned decommissioning activities according to their safety significance thereby ensuring that they do not compromise the safety of decommissioning activities.

*Related **IAEA** safety standards:*

In order to control all decommissioning activities, the operating organization should implement an effective management control system. This should include control of preparatory decommissioning activities (such as the installation of new safety systems) and recognition of the risks associated with the changing conditions that arise during decommissioning. (WS-G-2.4, para 7.7)

... The management system should provide assurance that:
...
(h) Appropriate updating and maintenance of safety assessments are performed with due consideration of: changes in the state of the facility as decommissioning progresses; the decommissioning plan; the acquisition of new knowledge; new regulatory concerns; updates of the inventory on the basis of data from sampling and environmental monitoring; measurements of occupational doses; and radioactive releases during decommissioning activities; (WS-G-5.2, para 3.34)

3.8. Safety issue: Period of Deferment

D-48: In case of deferred dismantling the licensee shall make the facility passively safe as far as it is reasonably practicable before entering the period of deferment, so as to minimize the need for active safety systems, monitoring, and human intervention in order to ensure safety.

*Related **IAEA** safety standards:*

If the deferred dismantling strategy has been selected, it shall be demonstrated in the decommissioning plan that such an option will be implemented safely and will require minimum active safety systems, radiological monitoring and human intervention and that future requirements for information, technology and funds have been taken into consideration. The potential aging and deterioration of any safety related equipment and systems shall also be considered. (WS-R-5, para 5.14)

D-49: Before the start of the period of deferment, the licensee shall develop an adequate care-and-maintenance program, the implementation of which ensures safety and does not impair future decommissioning.

*Related **IAEA** safety standards:*

Maintenance may be important during deferred decommissioning since part of the safety of the installation may rely on systems that have to retain their capability to perform for extended periods of time. Periodical monitoring of all the safety related components of the installation should be incorporated into the decommissioning plan. *(WS-G-2.1 para 6.21)*

4. Safety area: Safety verification

4.1. Safety issue: Contents, review and update of the safety case for decommissioning[3]

D-50: The licensee shall provide a safety case, which addresses all issues relevant for safety during decommissioning (for typical contents refer to appendix A). It shall be used as the basis for assessing the safety implications of changes to the facility or to decommissioning practices.

In particular the safety case shall address:
- dynamic changes in facility state,
- new or modified installations, systems and equipment,
- management of large quantities of radioactive material,
- conventional safety and radiation protection issues from demolition and dismantling and also the unusual working environment.

*Related **IAEA** safety standards:*

The decommissioning plan shall be supported by an appropriate safety assessment covering the planned decommissioning activities and abnormal events that may occur during decommissioning. The assessment shall address occupational exposures and potential releases of radioactive substances with resulting exposure of the public. (WS-R-5, para 5.2)

The experience from previous decommissioning should be appropriately taken into account as a matter of principle. The following list of items to be considered for the final decommissioning plan should thus be updated whenever previous decommissioning experience permits:

(a) a description of the nuclear reactor, the site and the surrounding area that could affect, and be affected by, decommissioning;
(b) the life history of the nuclear reactor, reasons for taking it out of service, and the planned use of the nuclear installation and the site during and after decommissioning;
(c) a description of the legal and regulatory framework within which decommissioning will be carried out;
(d) explicit requirements for appropriate radiological criteria for guiding decommissioning;
(e) a description of the proposed decommissioning activities, including a time schedule;
(f) the rationale for the preferred decommissioning option, if selected;
(g) safety assessments and environmental impact assessments, including the radiological and non-radiological hazards to workers, the public and the environment; this will include a description of the proposed radiation protection procedures to be used during decommissioning;
(h) a description of the proposed environmental monitoring programme to be implemented during decommissioning;
(i) a description of the experience, resources, responsibilities and structure of the decommissioning organization, including the technical qualification/skills of the staff;
(j) an assessment of the availability of special services, engineering and decommissioning techniques required, including any decontamination, dismantling and cutting technology as well as remotely operated equipment needed to complete decommissioning safely;
(k) a description of the quality assurance programme;
(l) an assessment of the amount, type and location of residual radioactive and hazardous non-radioactive materials in the nuclear reactor installation, including calculational methods and measurements used to determine the inventory of each;
(m a description of the waste management practices, including items such as:
— identification and characterization of sources, types and volumes of waste;
— criteria for segregating materials;
— proposed treatment, conditioning, transport, storage and disposal methods;
— the potential to reuse and recycle materials, and related criteria; and

[3] For explanations on the relation between the final decommissioning plan and the safety case for decommissioning please refer to appendix C

— anticipated discharges of radioactive and hazardous non-radioactive materials to the environment;
(n) a description of other applicable important technical and administrative considerations such as safeguards, physical security arrangements and details of emergency preparedness;
(o) a description of the monitoring programme, equipment and methods to be used to verify that the site will comply with the release criteria;
(p) details of the estimated cost of decommissioning, including waste management, and the source of funds required to carry out the work; and
(q) a provision for performing a final confirmatory radiological survey at the end of decommissioning. (WS-G-2.1, para 5.11)

As part of the operator's responsibility for all aspects of safety and environmental protection during all phases of decommissioning, as required in Ref. [1], para. 3.8, an appropriate safety assessment should be performed:
(a) To support the selection of the decommissioning strategy, the development of a decommissioning plan and associated specific decommissioning activities;
(b) To demonstrate that exposures of workers and of the public are as low as reasonably achievable (ALARA) and do not exceed the relevant limits or constraints [3]. (WS-G-5.2, para 2.1)

The safety assessment for decommissioning should:
(a) Document how regulatory requirements and criteria are met to support the authorization5 of the proposed decommissioning activities;
(b) Include a systematic evaluation of the nature, magnitude and likelihood of hazards and their radiological consequences for workers, the public and the environment for planned activities and for accident conditions;
(c) Quantify the systematic and progressive reduction in radiological hazards to be achieved through the conduct of the decommissioning activities;
(d) Identify the safety measures, limit controls and conditions that will need to be applied to the decommissioning activities to ensure that the relevant safety requirements and criteria are met and maintained throughout the decommissioning;
(e) Where relevant, demonstrate that the institutional controls applied after decommissioning will not impose an undue burden on future generations;
(f) Provide input to on-site and off-site emergency planning and to safety management arrangements;
(g) Provide an input into the identification of training needs for decommissioning and of competences for staff performing decommissioning activities. (WS-G-5.2, para 2.3)

D-51: The safety case shall be consistent with the final decommissioning plan and its subsequent updates.

*Related **IAEA** safety standards:*

The safety assessment for decommissioning should be consistent with the decommissioning plan [1, 9–11] and with other relevant national and site specific strategies and requirements, for example, with requirements for radioactive waste management and for the release of material and sites from regulatory control. (WS-G-5.2, para 2.2)

D-52: The safety case for decommissioning and any updates of the final decommissioning plan shall be submitted to the regulatory body.

*Related **IAEA** safety standards:*

Prior to the implementation phase of decommissioning activities, a final decommissioning plan shall be prepared and submitted to the regulatory body for approval. This plan shall define how the project will be managed, including: the site management plan, the roles and responsibilities of the organizations involved, safety and radiation protection measures, quality assurance, a waste management plan, documentation and record keeping requirements, a safety assessment and an environmental assessment and their criteria, surveillance measures during the implementation phase, physical protection measures as required, and any other requirements established by the regulatory body. (WS-R-5, para 5.10)

The safety assessment should employ a systematic methodology to demonstrate compliance with safety requirements and criteria for decommissioning throughout the decommissioning process, including the release of material, buildings and sites from regulatory control. In addition, the safety assessment should be used to help ensure that interested parties are confident of the safety of decommissioning. Once developed by the operator, the safety assessment should be reviewed by the regulatory body to ensure compliance with the relevant safety requirements and criteria. (WS-G-5.2, para 1.3)

D-53: To support the safety case for decommissioning, the licensee shall examine records and conduct surveys and measurements to verify the inventory and locations of radioactive, fissile or other hazardous materials in the facility and the surrounding potentially affected areas.

*Related **IAEA** safety standards:*

The responsibilities of the operating organization include:
- *...*
- *Performing safety assessments and environmental impact assessments related to decommissioning;*
- *Performing appropriate radiological surveys in support of decommissioning;*
- *... (WS-R-5, para 3.8)*

During the preparation of the final decommissioning plan, the extent and type of radioactive material (irradiated and contaminated structures and components) at the facility shall be determined by means of a detailed characterization survey and on the basis of records collected during the operational period. If nuclear material or operational waste remains at the facility, this radioactive material shall be included in the characterization survey. (WS-R-5, para 5.11)

D-54: The licensee shall review and update the safety case for decommissioning
- at major steps in the decommissioning project and
- when changes of the decommissioning plan are intended or changes of regulatory requirements or other safety relevant information arise

to ensure the safety case is still valid and appropriate to support the safe conduct of the decommissioning work.

*Related **IAEA** safety standards:*

The safety assessment for decommissioning should be consistent with the decommissioning plan [1, 9–11] and with other relevant national and site specific strategies and requirements, for example, with requirements for radioactive waste management and for the release of material and sites from regulatory control. (WS-G-5.2, para 2.2)

The safety assessment for decommissioning should be reviewed and updated, as appropriate, to ensure that it remains an accurate representation of the physical, chemical and radiological state of the facility as the decommissioning activities proceed. (WS-G-5.2, para 2.4)

At facilities for which a phased (step by step) approach to decommissioning has been selected, account should be taken in the safety assessment of the phases, the nature of the decommissioning activities and the hazards they entail, which may differ for each phase. A graded approach should be applied to each decommissioning phase. (WS-G-5.2, para 3.4)

D-55: The licensee shall carry out at regular intervals a review of the safety of the facility under decommissioning at a frequency established by the regulatory body.

An update of the safety case according to D-54 that also fulfils the requirements of D-56 is equivalent to the review required above.

D-56: The review according to D-55 shall confirm the compliance of the decommissioning activities and states with regulatory requirements and any deviations shall be resolved. It shall also identify and evaluate the safety significance of deviations from applicable current safety standards and best practices and take into account the cumulative effects of changes to procedures, modifications to the facility and the decommissioning organization, technical developments, decommissioning experience accumulated and ageing of SSCs. The safety case shall be updated accordingly.

4.2. Safety issue: Decommissioning reporting

D-57: The licensee shall review the progress in decommissioning against the plan and shall report periodically on the results to the regulator as required.

Related IAEA safety standards:

The responsibilities of the operating organization include:

- *...*
- *Keeping records and submitting reports as required by the regulatory body (WS-R-5, para 3.8)*

D-58: The licensee shall prepare a final decommissioning report to demonstrate, that the decommissioning has been completed and the proposed end state of the facility or site has been achieved.

*Related **IAEA** safety standards:*

A final decommissioning report shall be prepared that documents, in particular, the end state of the facility or site, and this report shall be submitted to the regulatory body for review. (WS-R-5, para 9.3)

On completion of decommissioning, appropriate records should be retained as specified by the regulatory body. These records should be held and maintained for purposes such as confirmation of the completion of decommissioning in accordance with the approved plan. The confirmation of the completion of decommissioning should include information on the disposition of waste, materials and premises. (WS-G-2.4, para 8.1)

D-59: The licensee shall ensure that relevant records and the final decommissioning report are available and accessible at the end of decommissioning according to the national regulatory system.

Related IAEA safety standards:

A system shall be established to ensure that all records are maintained in accordance with the records retention requirements of the quality assurance system and the regulatory requirements. (WS-R-5, para 9.4)

On completion of decommissioning, appropriate records should be retained as specified by the regulatory body. These records should be held and maintained for purposes such as confirmation of the completion of decommissioning in accordance with the approved plan. The confirmation of the completion of decommissioning should include information on the disposition of waste, materials and premises. (WS-G-2.4, para 8.1)

4.3. Safety issue: License termination conditions

D-60: Before a facility or site can be released from regulatory control, the licensee shall perform a final survey to demonstrate that the end-state, as approved by the regulatory body, has been met.

Related ***IAEA*** *safety standards:*

The responsibilities of the operating organization include:

- *...*
- *Ensuring that end state criteria have been met by performing a final survey;*
- *... (WS-R-5, para 3.8)*

The facility shall not be released from regulatory control, nor shall authorization be terminated until the operating organization has demonstrated that the end state in the decommissioning plan has been reached and that any additional regulatory requirements have been met. The regulatory body shall evaluate the end state of the site by performing a thorough inspection of the remainder of the facility after decommissioning activities have been completed to ensure that the end point criteria have been met. (WS-R-5, para 9.2)

D-61: At the completion of decommissioning, the licensee shall not be relieved of responsibility for the facility or site unless the regulatory body has agreed.

Related ***IAEA*** *safety standards:*

On completion of decommissioning it shall be demonstrated that the end state criteria as defined in the decommissioning plan and any additional regulatory requirements have been met. The operating organization shall only be relieved of further responsibility for the facility after approval by the regulatory body. (WS-R-5, para 9.1)

D-62: In the case of restricted use the licensee shall provide a long term impact assessment, an appropriate surveillance regime and any proposed land use restrictions.

Related ***IAEA*** *safety standards:*

If a facility cannot be released for unrestricted use, appropriate controls shall be maintained to ensure the protection of human health and the environment. These controls shall be specified and shall be subject to approval by the regulatory body. Clear responsibility shall be assigned for implementing and maintaining these controls. The regulatory body shall ensure that a programme has been established to apply the remaining regulatory requirements and to monitor compliance with them. (WS-R-5, para 9.6)

Appendix A: Example for a safety case for decommissioning

A typical safety case for decommissioning includes:

- description of the site, the facility layout (including the radiological characterisation plan of the facility) and facility performance during decommissioning activities,
- demonstration how safety is achieved (for normal operation and accidental situations, addressing radiological hazards and conventional hazards[1)], that may result in radiological consequences, and related scenario[2)]),
- detailed descriptions of the safety functions; all safety systems and safety-related SSCs; their design basis and functioning in all decommissioning states including anticipated decommissioning occurrences and accidents identify applicable regulations codes and standards,
- description of the relevant aspects of the decommissioning organization and the management of safety,
- documentation on the evaluation of the safety aspects related to the site,
- outline of the general safety objectives of decommissioning, design concept and the approach adopted to meet the fundamental safety objectives,
- description of the safety analyses performed to assess the safety of the facility in response to postulated initiating events against safety criteria and radiological release limits (see Appendix B),
- description of the on-site emergency operation procedures and accident management guidelines, the inspection and testing provisions, the qualification and training of personnel, the decommissioning experience feedback programme, and the management of ageing,
- technical bases for and description of the operational limits and conditions (OLCs),
- description of the policy, strategy, methods and provisions for radiation protection,
- description of the emergency preparedness arrangements,
- description of the on-site radioactive waste management provisions.

1) Significant conventional hazards that are of particular importance in the case of decommissioning are e.g. include: lifting and handling of heavy loads, use of hazardous materials for decontamination, stability of decontaminated structures, demolition.

2) Scenario related to radiological hazards that are of particular importance in the case of decommissioning are e.g. include: extensive cutting of activated and contaminated material, modification of safety barriers, entry into areas of the plant that were normally inaccessible, decontamination of large items, dispersion of contamination during demolition.

Appendix B: Postulated initiating events

External postulated events

Special attention shall be given to complex sites, where external events are likely to affect also neighbouring installations which could cause additional stress on the safety of the facility under decommissioning.

<u>Natural phenomena</u>

- Extreme weather conditions (precipitation: rain, snow, ice, frazil, wind, lightning, high or low temperature, humidity)
- Flooding
- Earthquake
- Natural fires
- Effect of terrestrial and aquatic flora and fauna (blockage of inlet and outlets, damages on structure)
- Possible combinations of such conditions

<u>Human induced phenomena</u>

- Fire, explosion or release of corrosive/hazardous substance (from surrounding industrial and military installations or transport infrastructure)
- Aircraft crash (accidents)
- Missiles due to structural/mechanical failure in surrounding installations
- Flooding (failure of a dam, blockage of a river)
- Power supply and potential loss of power
- Civil strife (infrastructure failure, strikes and blockages)
- Possible combinations of such conditions

Internal postulated events

- Loss of energy and fluids: Electrical power supplies, air and pressurised air, vacuum, super heated water and steam, coolant, chemical reagents, and ventilation;
- Improper use of electricity and chemicals
- Mechanical failure including drop loads, rupture (pressure retaining vessels), leaks (corrosion), plugging
- Instrumentation and control, human failures
- Internal fires and explosions (gas generation, process hazards)
- Flooding, vessel overflows

*Related **IAEA** safety standards:*
Selected postulated initiating events (DS 316 Appendix 1)

External postulated initiating events

<u>Natural phenomena</u>

- *Extreme weather conditions*
  precipitation : rain, snow, ice, frazil, wind, tornadoes, hurricanes, cyclones, dust or sand storm, lightning, high or low temperature, humidity

- *Flooding*
- *Earthquake and eruption of volcano*
- *Natural fires*
- *Effect of terrestrial and aquatic flora and fauna (blockage of inlet and outlets, damages on structure)*

<u>*Human induced phenomena*</u>

- *Fire, explosion or release of corrosive/hazardous substance*
- *(from surrounding industrial and military installations or transport infrastructure)*
- *Aircraft crash*
- *Missiles due to structural/mechanical failure in surrounding installations*
- *Flooding (failure of a dam, blockage of a river)*
- *Power supply and potential loss of power*
- *Civil strife (terrorism, sabotage, infrastructure failure, strikes and blockages)*

Internal postulated events

- *Loss of energy and fluids : Electrical power supplies, air and pressurized air, vacuum, super heated water and steam, coolant, chemical reagents, and ventilation;*
- *Use of electricity and chemicals*
- *Mechanical failure including drop loads, rupture (pressure retaining vessels), leaks (corrosion), plugging*
- *Instrumentation and control, human failures*
- *Internal fires and explosions (gas generation, process hazards)*

Flooding, vessel overflows

Appendix C:
Explanation of the relationship between
Final Decommissioning Plan and Safety Case

The WENRA WGWD applies in its reference levels for the safety during decommissioning a concept of decommissioning plan and safety case for decommissioning to address aspects of importance for safety of decommissioning in all phases of a facility lifetime.

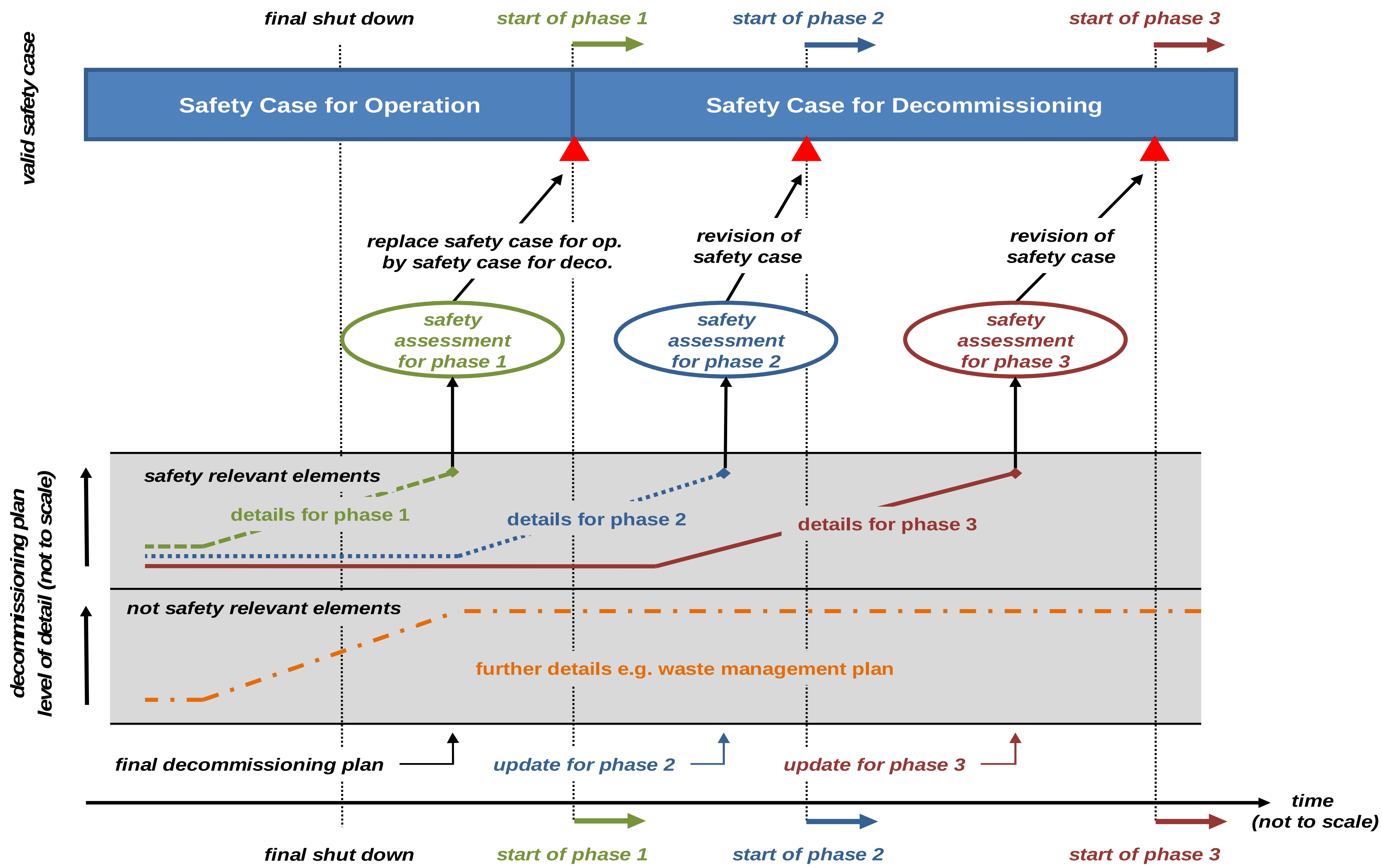
During the operational period the initial and updated decommissioning plan is addressing on a low level of details basic aspects of a future decommissioning of the facility. At the time of application for authorisation the aspects of planning of the individual decommissioning activities are addressed in the final decommissioning plan. Typical elements of the final decommissioning plan are:

- a detailed description of the intended decommissioning activities,
- information on the timeframe for the decommissioning,
- a description of the end-state of decommissioning,
- description of the content of phases, in case of a decommissioning project structured in different phases, and
- a description of the waste management programme.

Those parts relevant for safety during decommissioning, e.g. description of the intended decommissioning activities, are subject to a safety assessment and become part of the safety case for decommissioning which is the collection of arguments and evidence in support of the safety during decommissioning. An example of a safety case for decommissioning is already provided in Appendix A. Typical elements of a safety case are:

- a description of the legal framework,
- a facility description (incl. the radioactive and hazardous materials inventory),
- a description on the safety assessment for normal operation and accident situations and its results,
- a description of structures, systems and components and operational limits and conditions,
- the radiation protection programme,
- description of the on-site emergency planning.

Often large decommissioning projects are divided into different phases. In such cases the final decommissioning plan addresses in detail the first phase while the subsequent phases are addressed on a lower level; following figure illustrates this situation. Accordingly, the level of detail for future phases needs further evolution during conduct of decommissioning resulting in updates of the decommissioning plan for the specific phases. The safety relevant elements of the updated decommissioning plan become subject to related safety assessments which might result in revisions of the safety case for decommissioning



valid safety case
final shut down
start of phase 1
start of phase 2
start of phase 3
Safety Case for Operation
Safety Case for Decommissioning
replace safety case for op.
by safety case for deco.
revision of
safety case
revision of
safety case
safety
assessment
for phase 1
safety
assessment
for phase 2
safety
assessment
for phase 3
decommissioning plan
level of detail (not to scale)
safety relevant elements
details for phase 1
details for phase 2
details for phase 3
not safety relevant elements
further details e.g. waste management plan
final decommissioning plan
update for phase 2
update for phase 3
time
(not to scale)
final shut down
start of phase 1
start of phase 2
start of phase 3

WENRA
WESTERN EUROPEAN NUCLEAR
REGULATORS ASSOCIATION

RHWG
REACTOR HARMONISATION
WORKING GROUP

WGWD
WORKING GROUP ON WASTE
AND DECOMISSIONING

WIG
WENRA INSPECTION
WORKING GROUP