Western European

WENRA

Nuclear Regulator's Association

Safety Objectives for New Power Reactors

Study by

WENRA Reactor Harmonization Working Group

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RHWG

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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:
    - o accidents with core melt which would lead to early[20] or large[21] releases have to be practically eliminated[22] ;
    - o 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 :
    - o individual and collective doses for workers;
    - o radioactive and non radioactive discharges to the environment;
    - o quantity and activity of radioactive waste.

O7. Management of safety

- ensuring effective management of safety from the design stage. This implies that the licensee:
    - o 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;
    - o 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.

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[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:

	Level of defence in depth	Objective of the level	Essential means	Associated plant condition categories	Radiological consequences
Original design of the plant	Level 1	Prevention of abnormal operation and failure	Conservative design and high quality in construction and operation	Normal operation	Regulatory operating limits for discharge
	Level 2	Control of abnormal operation and failure	Control, limiting and protection systems and other surveillance features	Anticipated operational occurrences	Regulatory operating limits for discharge
	Level 3 (1)	Control of accident to limit radiological releases and prevent escalation to core damage conditions (2)	Safety systems Accident procedures	DiD Level 3.a Postulated single initiating events	No off-site radiological impact or only minor radiological impact (see NS-G-1.2/4.102)
		Control of accident to limit radiological releases and prevent escalation to core melt conditions (3)	Engineered safety features (4) Accident procedures	DiD Level 3.b **Selected multiples failures events** including possible failure or inefficiency of safety systems involved in DiD level 3.a	
	Level 4	**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**	**Engineered safety features to mitigate core melt** **Management of accidents with core melt (severe accidents)**	**Postulated core melt accidents (short and long term)**	Limited protective measures in area and time
Emergency planning	Level 5	Mitigation of radiological consequences of significant releases of radioactives materials	Off-site emergency response Intervention levels	-	Off site radiological impact necessitating protective measures

(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:
  - check that the CDF and radiological consequences are indeed reduced;
  - identify complementary design provisions where needed;
  - identify where diversity is needed in the design of safety systems, in complement to redundancy;
- Reduction of human-induced failures through:
  - more automatic or passive safety systems;
  - longer grace period for operators;
  - improved man-machine interface;
- Use of improved materials such as thermal insulation materials, to reduce the clogging phenomena on the sump filters.

Accidents with core melt (O3)

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:
  - effective reduction of loads on the containment arising from severe accidents situations and/or increased resistance of the containment to such loads
  - leaktightness of the containment in case of severe accident, including in the long term
  - provisions to avoid penetration of the corium through the containment basemat
  - 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:
  - for PWR steam generators, to reduce the probability of tube failures during core melt, before core relocation,
  - for vessel internals and reactor cavity (sacrificial and basemat materials) to take the core melt phenomena into account.

Independence between all levels of defence-in-depth (O4)

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**