Western European
WENRA
Nuclear Regulator's Association

Harmonization of Reactor Safety in WENRA Countries

Report by

WENRA Reactor Harmonization Working Group

January 2006

RHWG
Reactor Harmonization Working Group

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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 2006

Executive Summary

One of the aims of the Western European Nuclear Regulatory Association (WENRA) that now comprises the Chief Regulators of 17 European Nuclear Regulatory Authorities is to develop a harmonized approach to reactor safety. In order 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 scrutinised 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 results indicate that the majority of the Reference Level requirements are implemented in nuclear power plants in WENRA countries. All countries have some work to do for full harmonization, both to validate their implementation results and to align their regulatory requirements with the very strict harmonization definition.

The group's work signifies a considerable amount of effort and commitment by the participating organisations 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 achieved should be seen as a step towards meeting WENRA's commitment to achieving continuous improvement of reactor safety within Europe through mutual learning, and 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 the 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 coordinate the necessary actions within the participating organisations, a working group was formed (Reactor Harmonization Working Group, RHWG) with representatives from all 17 countries (Annex 4). It has taken the RHWG over 2½ 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;

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[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.

- 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
- 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 side 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 authorisation. 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

[6] It is not WENRA's primary goal to develop safety standards for the European Union.

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 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.

[7] Meeting between representatives from EUR and WENRA 03/2002.

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.
2. Countries assessed themselves against these Reference Levels on both the legal and implementation side and documented their position.
3. The national positions were scrutinised 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).

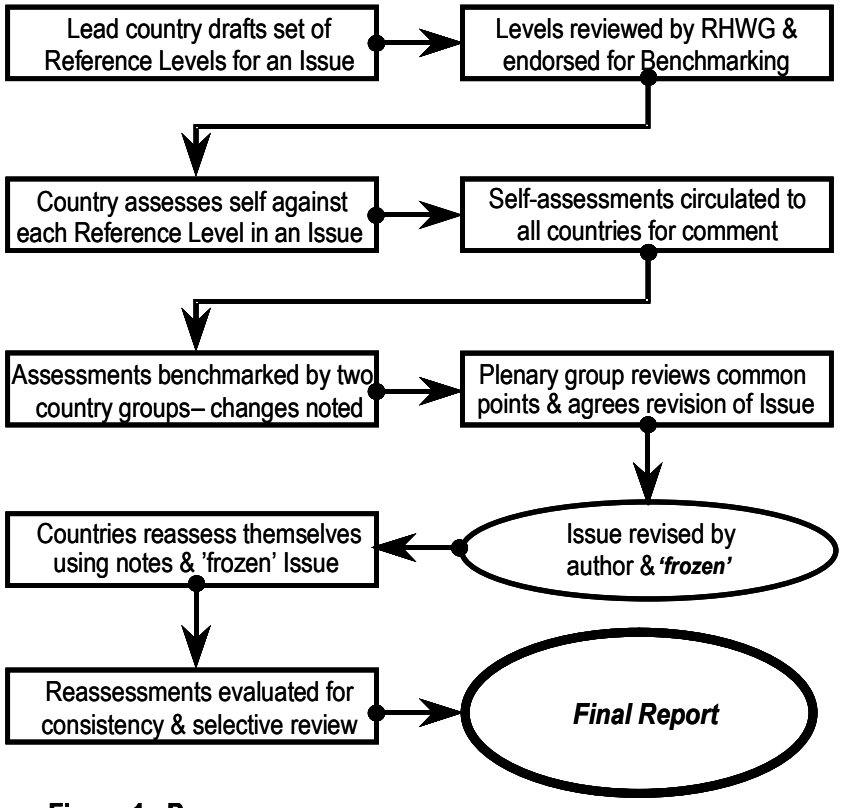


Figure 1: Process summary

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, 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, are both 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?

[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.

(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.

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 include the missing Reference Level requirement(s) by the end of 2005 at the latest;*
- *The Reference Level requirement is covered by a different national requirement to such an extent that the added safety value of the reference requirement is minor;*
- *Specifically for issue N: The Reference Level is included in 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 of a reference requirement is lacking in an older plant for which a shut down decision has been taken;*
- *Implementation of a reference requirement is in progress and will be completed before the end of 2005; or*
- *Implementation of a Reference Level requirement has been exempted because the regulatory body has accepted a technical justification.*

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 on 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 4.3, 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.

Figure 2(a):
Overview - All legal requirement results

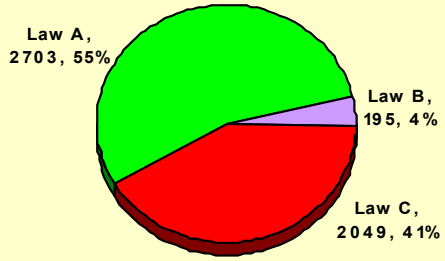
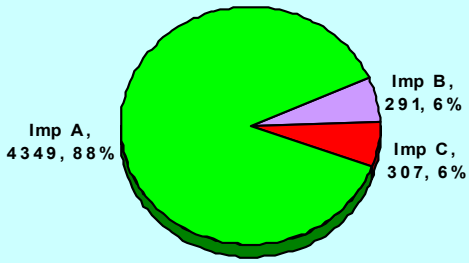


Figure 2(b):
Overview - All implementation results



5.1 General overview

Fig 2(a) summarises 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 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 Overview of safety issues

Grand totals by topic

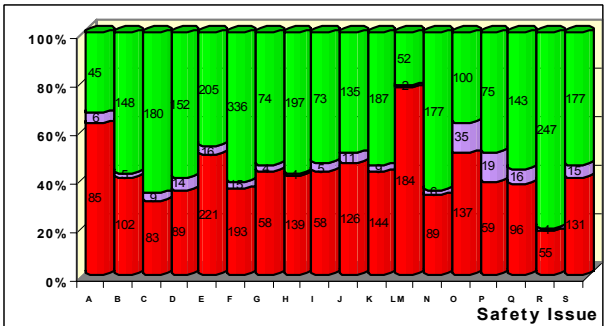


Figure 3(a): **Law: Grand totals by topic (%)**

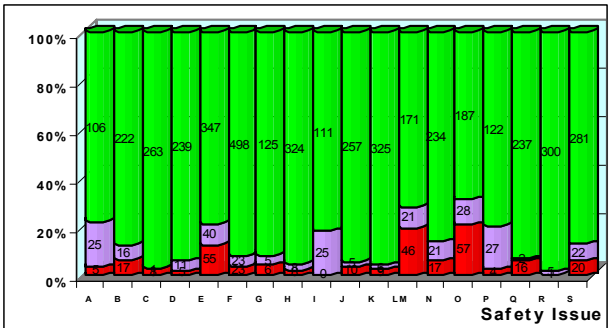


Figure 3(b): **Imp: Grand totals by topic (%)**

Figures 3(a) and 3(b) give the overall situation for all countries on the 18 safety 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 Authorisation* (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 cooperate 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

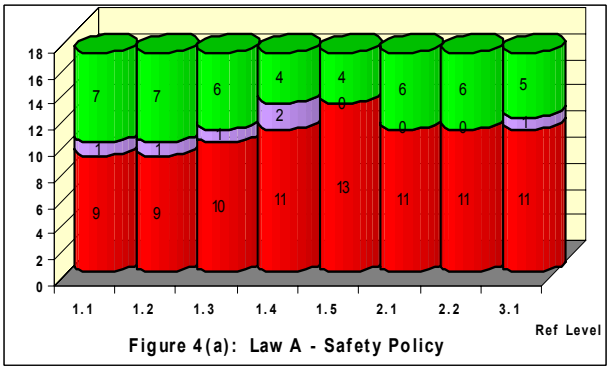


Figure 4(a): Law A - Safety Policy

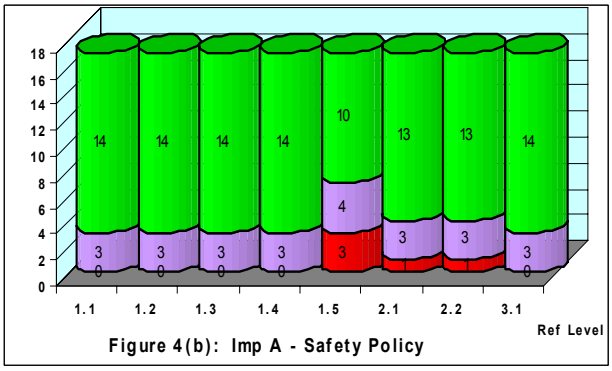


Figure 4(b): Imp A - Safety Policy

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

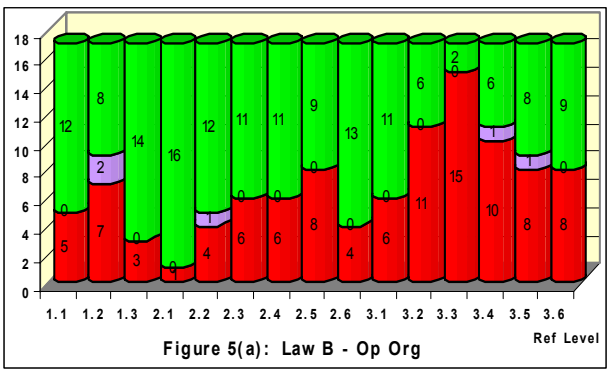


Figure 5(a): Law B - Op Org

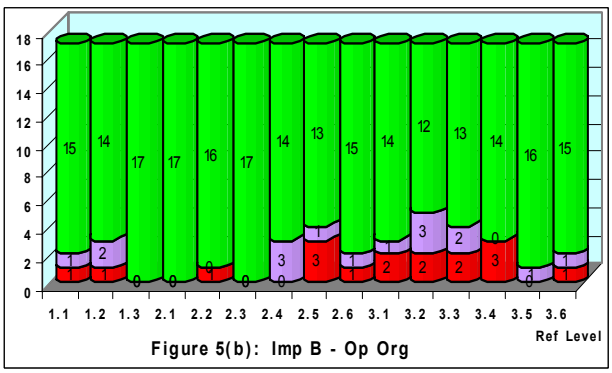


Figure 5(b): Imp B - Op Org

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

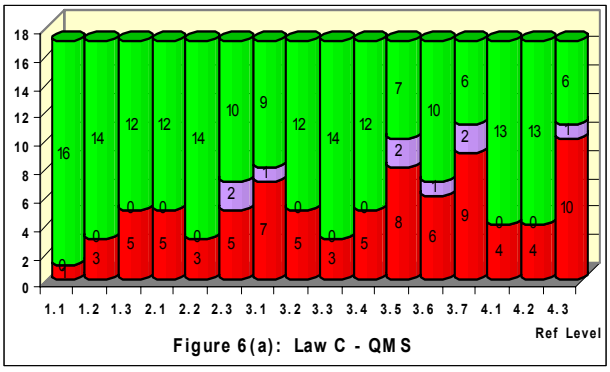


Figure 6(a): Law C - QMS

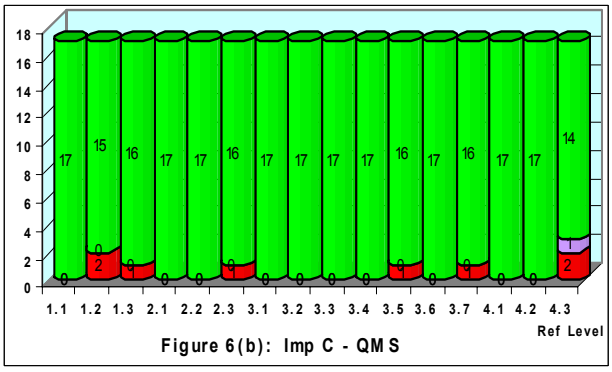


Figure 6(b): Imp C - QMS

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

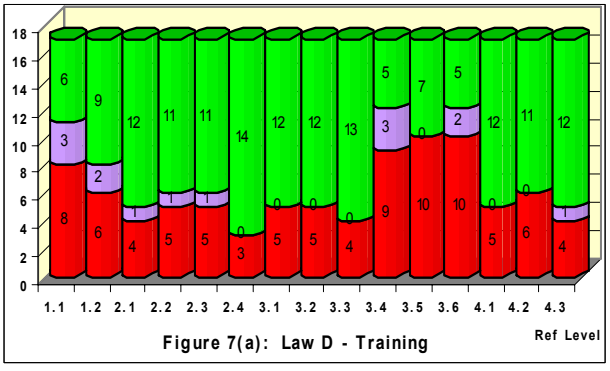


Figure 7(a): Law D - Training

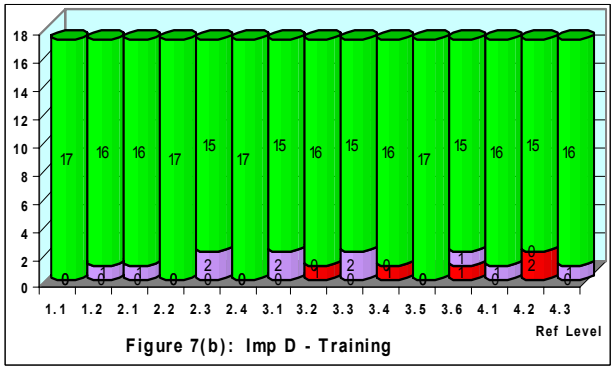


Figure 7(b): Imp D - Training

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

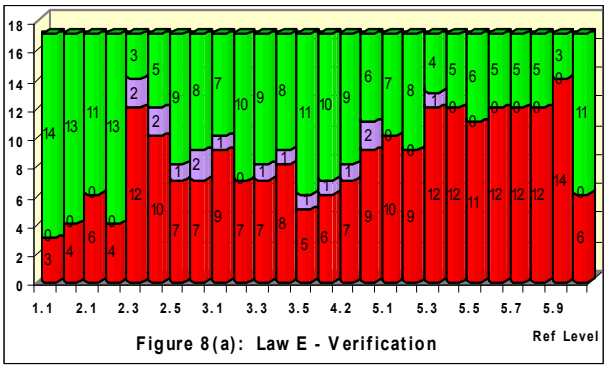


Figure 8(a): Law E - Verification

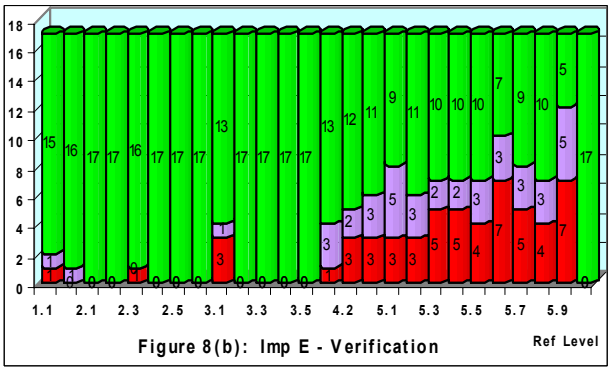


Figure 8(b): Imp E - Verification

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

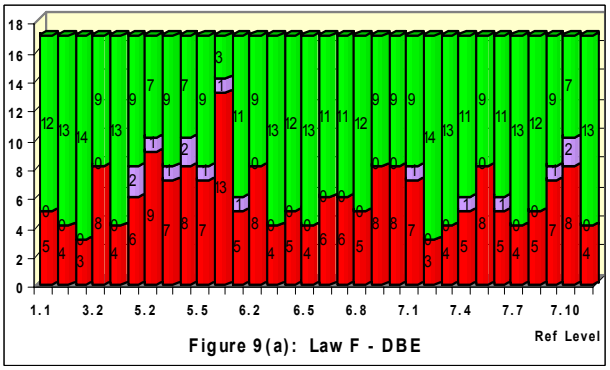


Figure 9(a): Law F - DBE

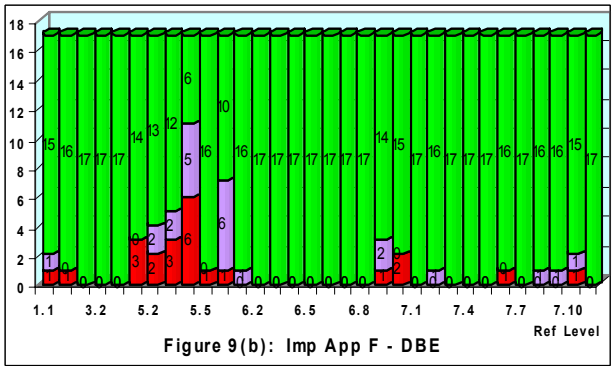


Figure 9(b): Imp App F - DBE

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

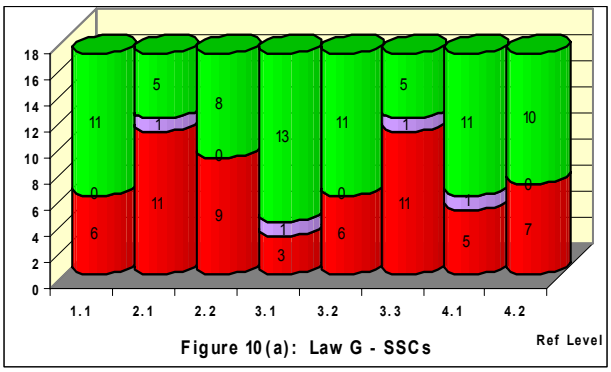


Figure 10(a): Law G - SSCs

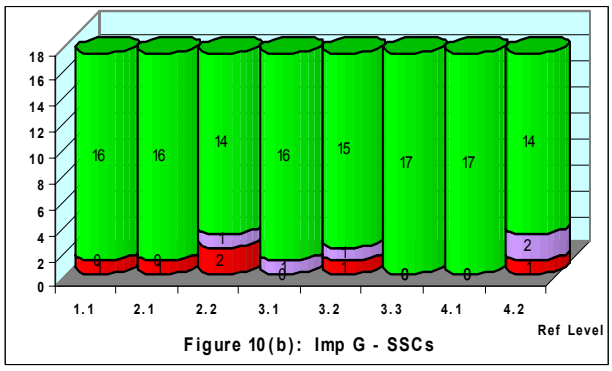


Figure 10(b): Imp G - SSCs

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

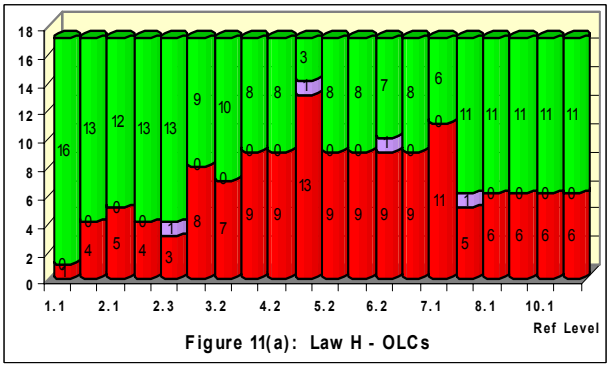


Figure 11(a): Law H - OLCs

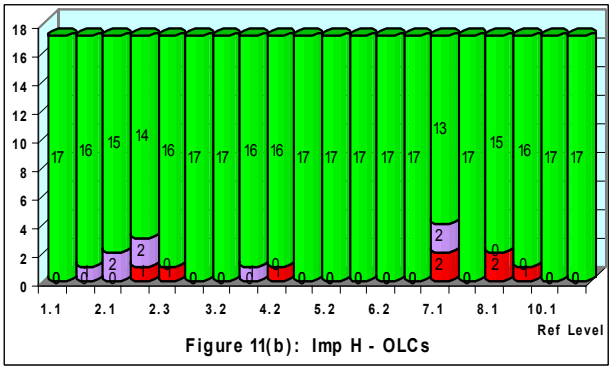


Figure 11(b): Imp H - OLCs

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.

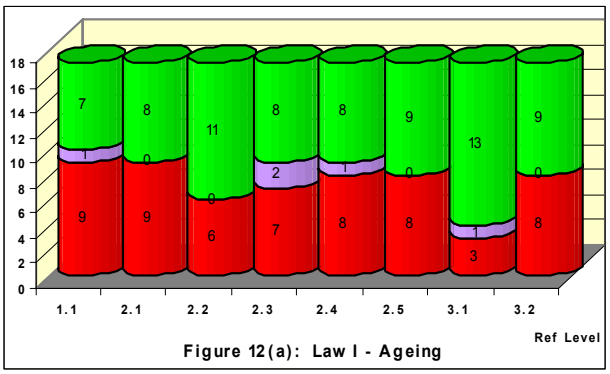


Figure 12(a): Law I - Ageing

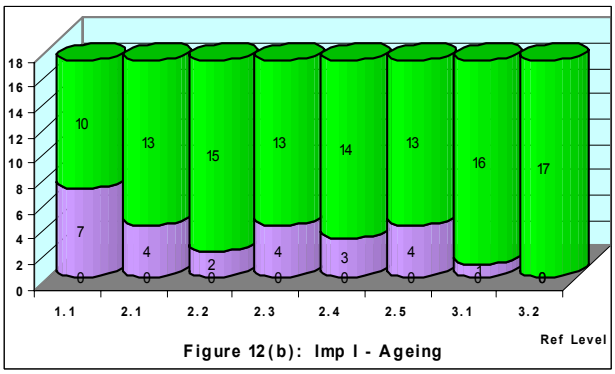


Figure 12(b): Imp I - Ageing

J – System for investigation of events and Operational Experience Feedback

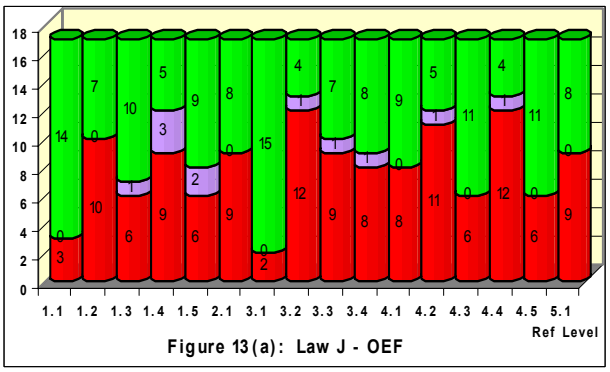


Figure 13(a): Law J - OEF

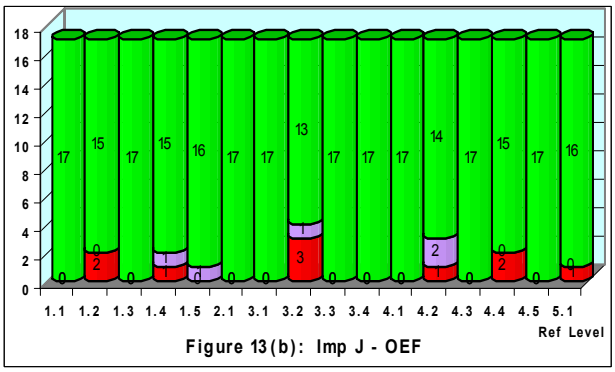


Figure 13(b): Imp J - OEF

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.

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

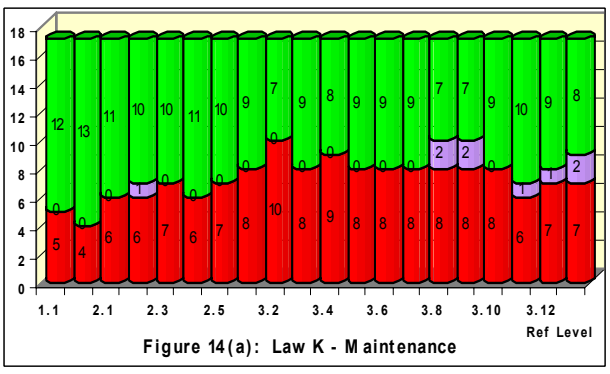


Figure 14(a): Law K - Maintenance

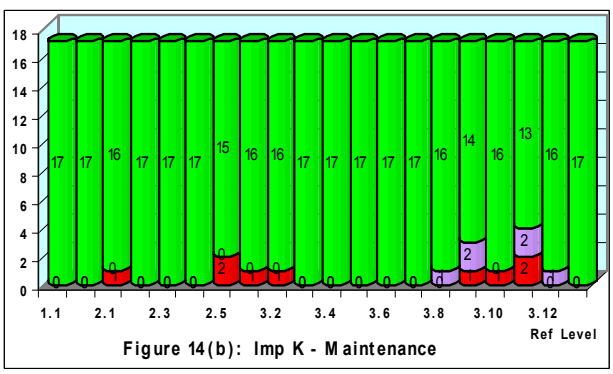


Figure 14(b): Imp K - Maintenance

In Figure 14(a), about 50% of the countries require the *Maintenance and Inspection* Reference Levels.

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

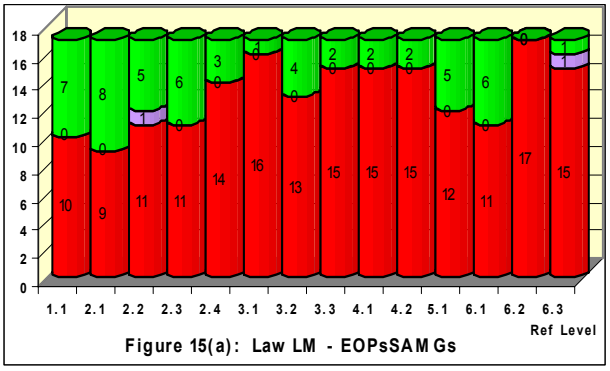


Figure 15(a): Law LM - EOPsSAMGs

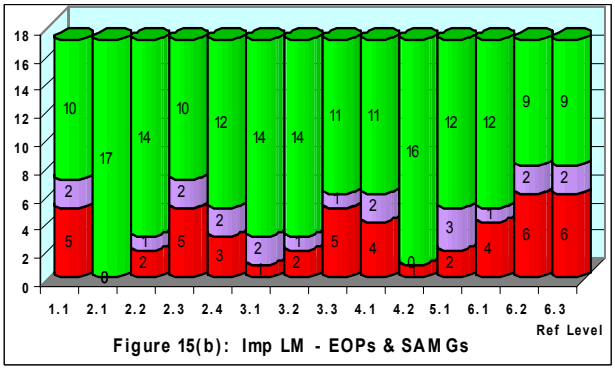


Figure 15(b): Imp LM - EOPs & SAMGs

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.

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

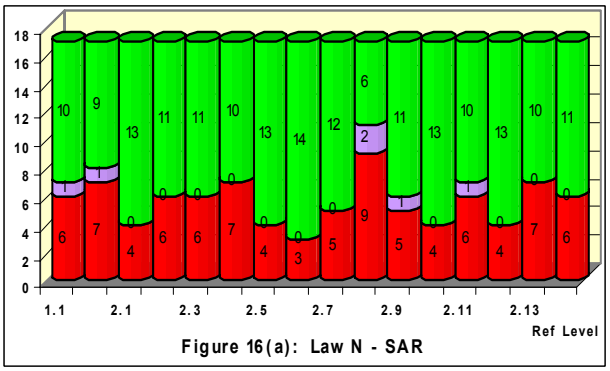


Figure 16(a): Law N - SAR

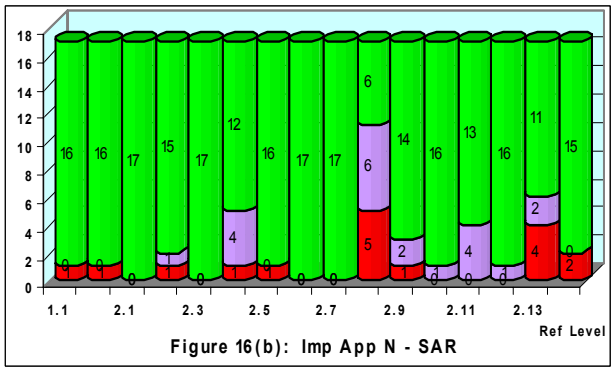


Figure 16(b): Imp App N - SAR

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

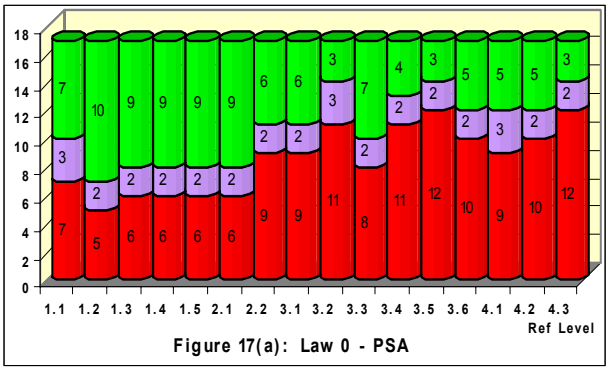


Figure 17(a): Law O - PSA

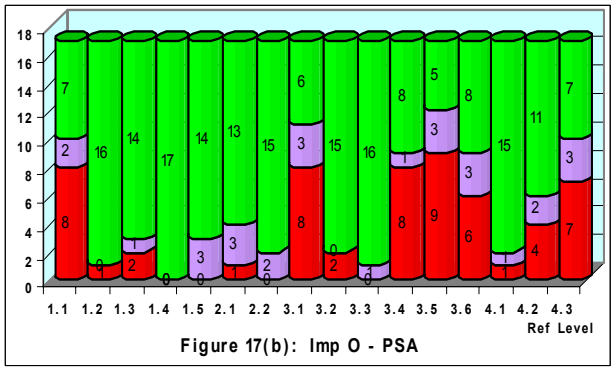


Figure 17(b): Imp O - PSA

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 have shown to be important to safety.

Figure 17(b) shows 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

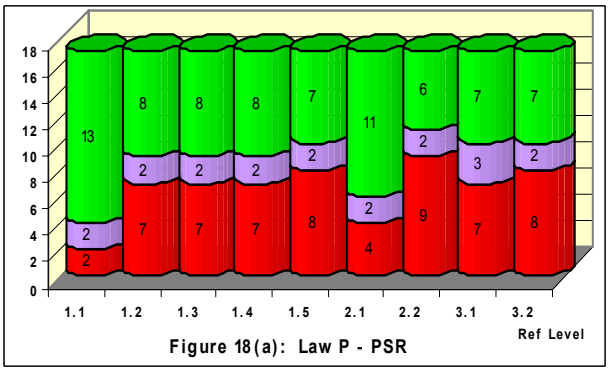


Figure 18(a): Law P - PSR

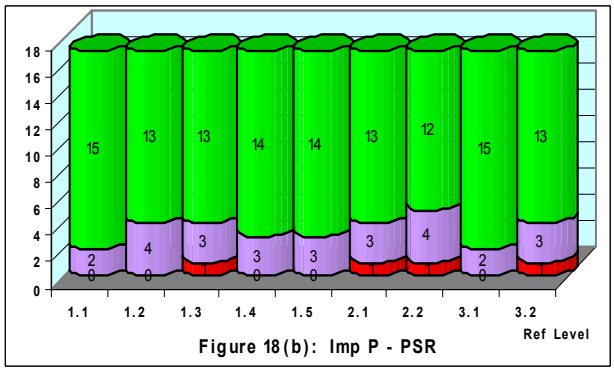


Figure 18(b): Imp P - PSR

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 has implemented four levels; the rest are implemented with some justifications (Figure 18(b)).

Q – Plant Modifications

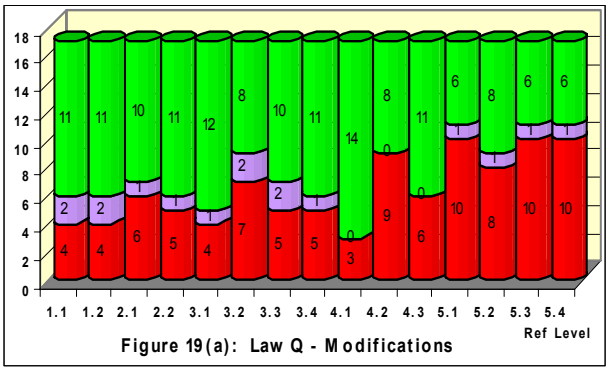


Figure 19(a): Law Q - Modifications

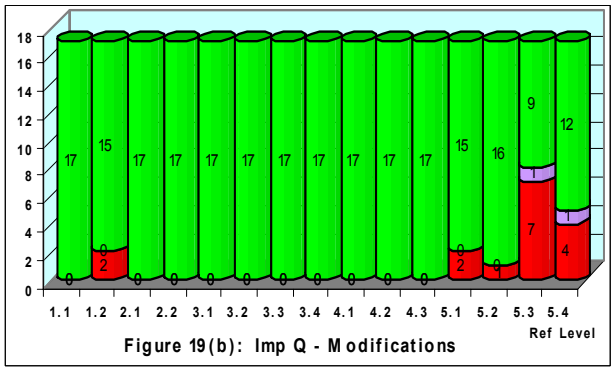


Figure 19(b): Imp Q - Modifications

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

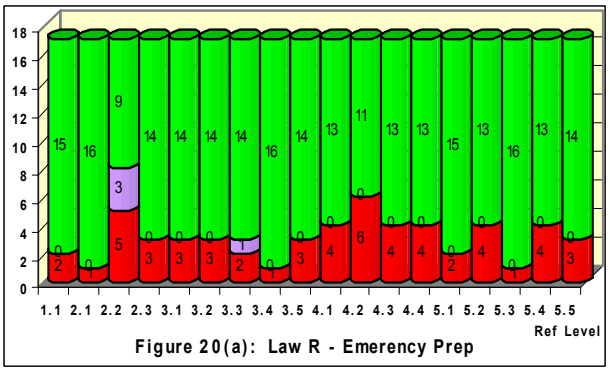


Figure 20(a): Law R - Emerency Prep

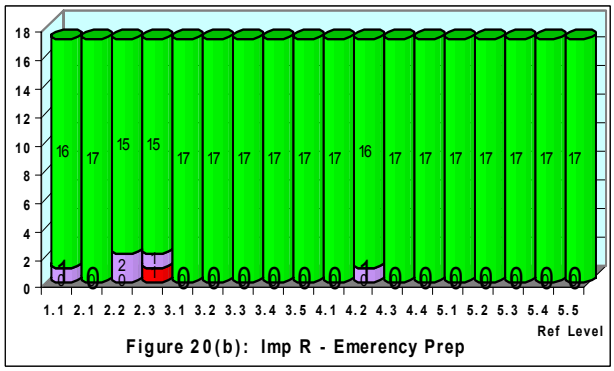


Figure 20(b): Imp R - Emerency Prep

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 organisation 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.

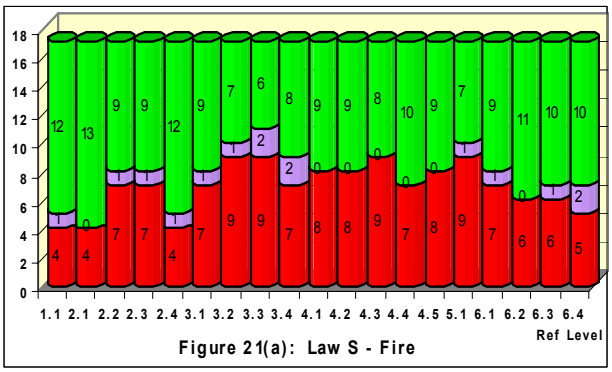


Figure 21(a): Law S - Fire

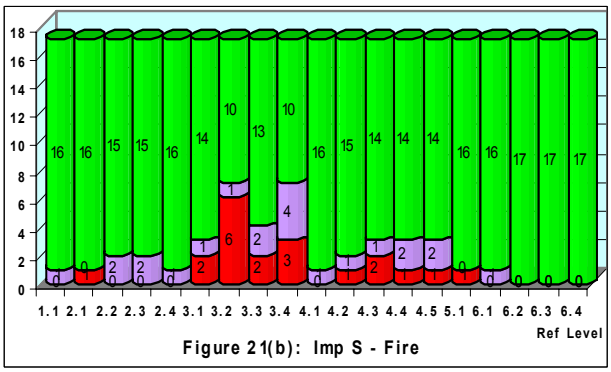


Figure 21(b): Imp S - Fire

A large majority of the countries has implemented most levels (Figure 21(b)). Least implemented levels relate to fire hazard analysis and coverage of water based fire protection systems.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

Issue B: Operating Organisation	
Safety area: Safety Management	Document status: Final

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[11] 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[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.

[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.

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.

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.

[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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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.

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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.

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 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 organisational 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

[18] Collaboration is taken to mean that all groups are involved in the process.

organisational 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.

[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

3.1 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 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.

4.4 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 penalising 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

[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.

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 3.3 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].

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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.

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.

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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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.

[26] Under the conditions specified in section 6.

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 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;
- 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 extension28. 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 penalising single failure shall be applied to an equipment used to achieve the safety function;

[27] In comparison with historical weather data for the site region.

[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.

- 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;
- 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.

[29] Proven by experience under similar conditions or adequately tested and qualified.

[30] Within 4-6 seconds, e.g. Scram system.

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.

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

[31] By computer sampling and/or print outs

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.

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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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.

[32] SSCs include software for I&C.

[33] Postulated Initiating Event – as defined by IAEA – includes consequences following on from event.

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.]

[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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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.

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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)*

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.

[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.

Western European Nuclear Regulators' Association
REACTOR HARMONISATION WORKING GROUP

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 organisation 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.

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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.

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 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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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;
- 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 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.

Western European Nuclear Regulators' Association
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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

[39] It is anticipated that such reviews are carried out more frequently than the 10-yearly Periodic Safety Reviews.

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.

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.

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).

[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

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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.

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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).

3.2 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.

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.

Western European Nuclear Regulators' Association
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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.

[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.

2.10 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.

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.

[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.

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.

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.

[46] Small deviations in the plant parameters that could give rise to severely abnormal plant behaviour.

Western European Nuclear Regulators' Association
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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;
- 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.

Western European Nuclear Regulators' Association
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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.

[47] Para 4.4 of IAEA Guide NS-G-2.3 contains information about possible categories.

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

4.3 Implementation and testing of plant modifications shall be performed in accordance with relevant work control and plant testing procedures.

4.3 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.

[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

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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:

(d) Prompt recognition and classification of emergencies;

(e) Timely notification and alerting of response personnel;

(f) Ensuring the safety of all persons present on the site, including the protection of the emergency workers;

(g) Informing the authorities and the public, including timely notification and subsequent provision of information as required;

(h) Performing assessments of the situation on the technical, & radiological points of view (on and off site);

(i) Monitoring radioactive releases;

(j) Treatment and first aid of a limited number of contaminated and/or overexposed workers/persons; and

(k) Plant management and damage control[49].

[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.

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.

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].

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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.

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.

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.

Western European Nuclear Regulators' Association
REACTOR HARMONIZATION WORKING GROUP

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]

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.

[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 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 scrutinised; 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 summarised 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 finalising 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 website: 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 website: www.asn.gouv.fr

Germany

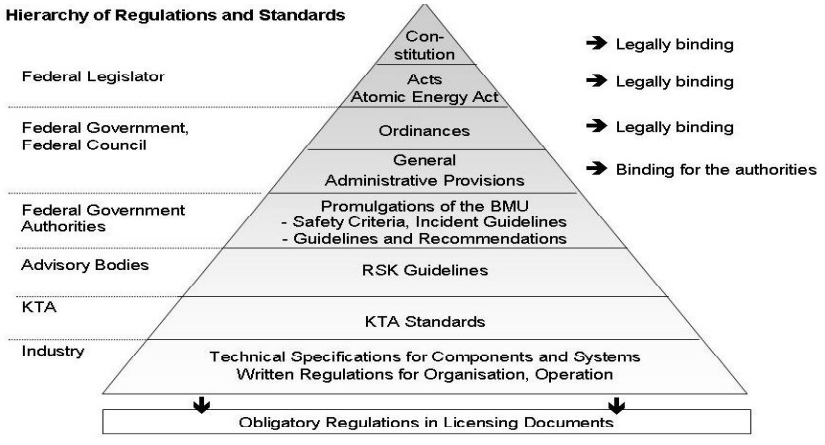
In accordance with the federal structure of the Federal Republic of Germany, its Constitution (Article 74 (1) 11a Basic Law) contains detailed provisions on the legislative and administrative competencies of the Federation and the individual Federal States (Länder). The Federation has enacted the "Federal 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 websites: 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

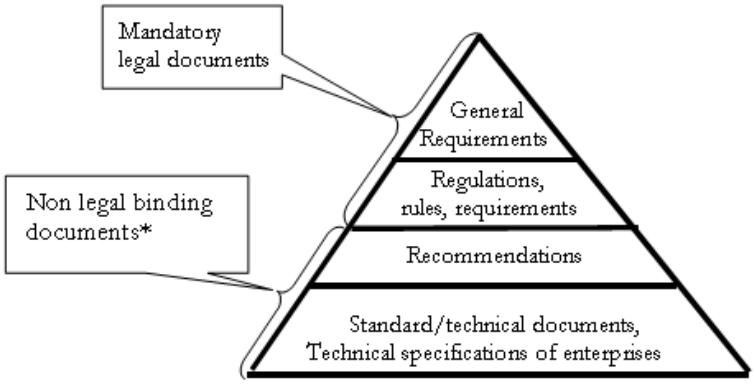
In Lithuania, the scope of the legislation covers the construction, operation and decommissioning of nuclear facilities and control of nuclear materials and wastes. The legal acts are updated, as necessary. The main legal document governing nuclear energy is the **Law on Nuclear Energy** passed by the 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 website 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 website 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 website: 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 websites: www.ski.se and www.ssi.se

Switzerland

With respect to nuclear installations, the legislative and regulatory framework in Switzerland is established by the federal constitution, federal laws, 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), which organisation 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 life. The Swiss regulatory body is composed of the HSK as the supervisory authority for nuclear safety and the Section for Nuclear Energy (again part of BFE) as the supervisory authority for nuclear security and safeguards. By law, nuclear safety is the full responsibility of the licence holder i.e. the NPP.

Licences (for siting, construction, operation, and decommissioning) are issued at the federal level. Each licence contains licence conditions that are mandatory for the licence holder. Within the licence conditions, the HSK issues permits / approvals e.g. for safety limit settings, plant safety system modifications or cycle start-up after refuelling outage.

The legal provisions for authorisation / regulation / supervision / inspection of nuclear installations in Switzerland have been established with the federal Atomic Energy Act of 1959 (Atomgesetz, AtG) and its associated Ordinance of 1984, as well as the Radiological Protection Act of 1991 (Strahlenschutzgesetz, StSG). The status of legal requirements, as identified in the reference levels, was assessed on this basis. The AtG – founded legal situation represents only principal or basic requirements, without addressing the very detailed reference level conditions. Some of these conditions are addressed in existing Regulatory Guidelines; however, such Guidelines were neither connected with the AtG nor mandatory. Thus, the assessment shows a lack of legal basis in a large number of cases.

With the enactment of the Federal Law on the use of Nuclear Energy (Kernenergiegesetz, KEG) and its associated Ordinance (Kernenergieverordnung, KEV) on February 1 2005, the legal basis has changed dramatically. KEG/KEV contain a multitude of detailed requirements, which relate to a significant part of the harmonization issues. The process of implementation of the new legislation, which entails the issuing of additional ordinances and Regulatory Guidelines – now linked with the new legislation - as well as rewriting the existing Guidelines, has already started; addressing the harmonization issues identified in this report is intended to be part of this process.

More detailed information about the HSK and its mandate may be found at www.hsk.ch.

This website includes a link to the 3rd Swiss report on the implementation of the obligations of the international “Convention on Nuclear Safety”, which report provides full details on the legislative and regulatory framework, and a link to all 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**