FINAL REPORT

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BENCHMARKING OF NUCLEAR TECHNICAL REQUIREMENTS AGAINST WENRA SAFETY REFERENCE LEVELS, EU REGULATORY FRAMEWORK AND IAEA STANDARDS

(ENER/2017/NUCL/SI2/760935)

Ref: ENER/D2/2016-677

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Benchmarking of nuclear technical requirements against WENRA safety reference levels, EU regulatory framework and IAEA standards

FINAL REPORT on Tasks:

Task 1: Benchmarking of the European Utility Requirements

Task 2: Possible application of the Franco German ETC

Task 3: A common European pre-licensing process: scope, content, implementation

Task 4: Benchmarking of the national long term operation programmes

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PROJECT BASIC DATA

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ABSTRACT

The present study supports past and on-going discussions with several stakeholders (ENSREG, ETSON, ENISS/EUR) and aims at bringing insights and material to Member States in line with 2017 Nuclear Illustrative Programme of the Commission which recommends that the cooperation on harmonisation of licensing process should be fostered and additional efforts to standardise codes and components should be done.

The project is composed of four tasks:

Task 1 addresses the benchmarking of the European Utility Requirements (EUR) against international safety standards, namely the requirements of the IAEA, WENRA and the amended NSD.

Task 2 is a feasibility study on the possibility to extend the work performed by the French and German TSOs in 1990s to other national legal frameworks and other type of reactors.

Task 3 is a detailed description of the technical content that an EU common pre-licensing process should include.

Task 4 developed an approach for benchmarking the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87. In this approach, all plant safety improvements that have been implemented PSR, LTO, the Stress test or that resulted from a national regulator or operator initiative, are taken into account.

EXECUTIVE SUMMARY

The 2017 Nuclear Illustrative Programme of the Commission[1] (PINC) provides a basis for discussion on the role of nuclear energy trends and related investments for the period up to 2050, including on how nuclear energy can help achieve the EU's energy objectives. As nuclear safety remains the Commission's absolute priority, it specifically includes investments related to post-Fukushima safety upgrades and those related to the long-term operation of existing nuclear power plants.

The PINC indicates how many countries in Europe and in the rest of the world intend to rely on nuclear energy to produce part of their electricity for the coming decades. This will imply new builds and Long Term Operation Programmes.

An important obstacle to a timely licensing is that the harmonization developed by national nuclear safety authorities mainly concerns high level safety standards. Whilst this is certainly useful to attain a similar level of safety at EU level, it does not contribute significantly to more rapid or generic licensing of new builds or to extend the operational life of existing reactors.

Hence the PINC recommends that the cooperation on harmonisation of licensing process should be fostered and additional efforts to standardise codes and components should be done.

The present study supports past and on-going discussions with several stakeholders (ENSREG, ETSON, ENISS/EUR) and aims at bringing insights and material to Member Sates which have an exclusive competence in licensing Nuclear installations.

The project ENER/2017/NUCL/SI2/760935 entitled “Benchmarking of nuclear technical requirements against WENRA safety reference levels, EU regulatory framework and IAEA standards” is composed of four tasks:

Task 1: Benchmarking of the European Utility Requirements;
Task 2: Possible application of the Franco German ETC;
Task 3: A common European pre-licensing process: scope, content, implementation;
Task 4: Benchmarking of the national long term operation programmes.

Task 1 addresses the benchmarking of the European Utility Requirements (EUR) against international safety standards. The purpose of the EUR document is to present clear and complete statements of utility expectations for the design of Generation III nuclear power plants. The objective of this task is to verify the EUR organisation’s claims that the EUR Revision E is in line with the requirements of the IAEA, WENRA and the amended Nuclear Safety Directive of the EU. The overall benchmarking of European Utility Requirements against these international standards has resulted in the conclusion that the European Utility Requirements are in full compliance with the international standards. Even though certain differences exist, they are seen as being there in most cases due to limited different interpretation or different level of depth in addressing certain requirements. However the amplitude or the nature of those differences does not jeopardize the compliance conclusions. On few occasions EUR make a simple statement that if certain requirements are not sufficiently specified the corresponding IAEA safety standards shall be applied. In practical terms it means that in such cases the IAEA Safety Standards are to be followed and they supersede or complement the EUR requirements. This approach has been accepted and judged as being adequate for assurance of the completeness of the EUR standards. An example of this approach can be found in EUR Chapter 2.15 on Quality Assurance where it is stated that the overall Quality Assurance Programme (QAP) shall be in compliance with the IAEA Safety Requirements GSR Part 2 on “Leadership and Management for Safety”. This statement puts GSR Part 2 as an umbrella document for the overall QAP.

A detailed comparison of each individual requirement is presented in tabular form in the main body of the report in which also observed differences are described. A special chapter at the end of the Task 1 presentation is devoted to the analysis of selected, most important requirements that were introduced

[1] COM(2017) 237 Final

in all international standards after the Fukushima accident. It should be pointed out that in these, most recent and most important international standards full compliance has been identified with no significant observations, as obviously the authors of the EUR have also devoted needed attention to these emerging issues. The overall conclusion of the study is that EUR are in full compliance with the benchmarked international standards.

These results indicate that the EUR may be considered by national authorities in supporting their licensing process of new reactors.

The aim of Task 2 is to perform a feasibility study on the possibility to extend the work performed by the French and German TSOs in 1990s to develop a common understanding and practical description of safety requirements to other national legal frameworks and other type of reactors. At the time it was achieved, this work was considered sufficiently detailed to support efficiently licensing detailed review and was integrated into the AFCEN relevant codes.

Under this task the experience from experts involved in this Franco-German collaboration at that time was solicited and extensive surveys of available literature, published papers and official publications dealing with the issue were conducted. Based on the information gathered, and based on the knowledge of experts involved in this process about the current status of international codes and standards, a feasibility study was performed on the way that Franco-German experience can be utilized for possible future cooperation between or among different European countries in designing, licensing or constructing a nuclear power plant of selected reactor types. The work undertaken under this Task 2 was performed in several steps:

1. Description of the process which led to the establishment of the "Guidelines" and ETCs. Both positive aspects and challenges are addressed as well as lessons learned.
2. How was the work performed and how this successful output was achieved?
3. How did the TSOs and regulators come up with a set of codes which were acceptable to both sides, in spite of well-established and detailed codes being in existence in France (RCC) and in Germany (KTA rules)?
4. Feasibility study; could this process be replicated today, for any given pair or group of regulators and any given design, and under which boundary conditions it could succeed?

At the end, steps were detailed that would be necessary to be undertaken if a full study on the way to implement such cooperation would be performed.

The feasibility study identified that commendable element of the Franco-German cooperation was the strong and well-organised implementation, based on a governmental declaration and on agreements between all parties. The Franco-German process was initiated by industry but in parallel supported by politics at the highest level.

The main conclusions of the feasibility study are the following:

For any replication, such strong political support seems essential. In addition, the following boundary conditions must be considered:

- Strong support by governments;
- Strong role of industry;
- Experienced regulators and TSOs;
- Technical approach : Develop common Safety Objectives;
- Good agreement on work conditions and organisation of the cooperation.

Before the work on the development of common safety standards and requirements begins, the following issues would need to be addressed as a minimum:

- Define precisely the area and objectives of common work;
- Define duties and responsibilities of all stakeholders involved;
- Clarify compatibilities and non-compatibilities of existing regulatory frameworks;
- Clarify the existing structure of all organizations involved (TSO, advisory, regulatory) in all countries involved;
- Define the level of professional capabilities and experience for the experts involved;
- Evaluate possible impact of different culture/mentality/language in the interaction of the experts;
- Establish clear rules and procedures for interaction, communication and documentation.

If a full study were to be undertaken on the way to implement such cooperation would be performed, based on the findings of the feasibility study, several steps would need to be investigated based on the experience of the Franco-German project:

- Step 1: Investigation of the status of the nuclear programs and regulatory frameworks in participating countries;
- Step 2: Compatibility check of basic principles and safety objectives set in relevant national legislations;
- Step 3: Determination of professional profile of available experts, possible impact of different culture/mentality/language, rules and procedures for interaction;
- Step 4: Overall schedule of the cooperative project.

Given all these factors, the general conclusion of Task 2 is that it would be worthwhile to undertake a full study with the aim of setting out the conditions for possible future cooperation between two or among several countries to establish a common ground for designing, licensing or constructing selected nuclear power plant technologies.

Within the Task 3 a detailed description of the technical content that a EU common pre-licensing process should include has been developed, taking into account the different types of reactors, the applicable safety standards and (to a certain extent) the diversity of Member States’ national framework.

To achieve this objective, the existing approaches, both on national and on international level, towards a pre-licensing reactor design evaluation and "acceptance" were analysed, with a specific focus on the scope and content both of the review and the resulting statement by the participating authority/authorities. This is a desktop study, based on publicly available information, especially on information published by the relevant regulators. Also, the professional experience of the authors in regulatory matters on an international scale provides an essential contribution to the study.

Based on the analyses of the various approaches and a discussion of their practical advantages and drawbacks, a concept is developed for a Joint Overall Design Assessment, to be possibly performed by the regulators of several Member States, leading to a Common Opinion on Design Acceptability.

The focus of this task is put on scope and technical content of the pre-licensing assessment.

Basically, the scope/broadness of such a pre-licensing should be “full scope” and encompassing the entire range of safety analysis of a reactor design, as it is defined, in a more or less coherent fashion, by national and international assessment standards such as the IAEA's SSR-2/1, Safety of Nuclear Power Plants: Design, and MDEP's General Terms of Reference for design-specific working groups. This can be adjusted in details to fit the specific needs of the participating regulators and the vendor by entering into contractual arrangements. These arrangements could incorporate elements facilitating the project-independent pre-licensing assessment, such as envelope criteria – especially site criteria – or the possibility for the designer to present alternative solutions for certain design elements in order to fulfil the requirements under different national regulations.

Concerning the depth of the assessment, a level of detail such as that required for the issuing of a construction licence will probably not be achievable in the framework of a multinational cooperation and may not even be useful; on the other hand, the assessment must be "meaningful", i.e. substantial and technical enough to effectively front-load national licensing processes. Therefore, this Report proposes that the technical content to be assessed for all topics should in substance be "Grade 2" as defined here. Grade 2 comprises the evaluation whether the general plant design, according to the claims of the designer, complies with the main safety goals in relevant legislation and regulation (Grade 1) and whether the claims can be demonstrated to be based on compliance with basic requirements as contained in national regulations (Grade 2). For certain issues defined in each JODA process, the assessment could also be taken to Grade 3, comprising evaluation of the main design features against more detailed regulations and codes and standards. This proposed level of detail ensures meaningfulness of the assessment and enables, as far as possible, a smooth interface with the subsequent national licensing processes.

Thus, JODA can be summarised as featuring a "full scope but limited depth" assessment, subject to a certain degree of flexibility. For any particular assessment, this basic flexibility requires contractual agreements to freeze the criteria and to provide certainty to all parties involved.

The flexibility also applies to the reactor designs eligible for the process. Under the general guidance mentioned above (supplemented by SMR-specific aspects if necessary), the process would be open to all technologies.

Concerning the safety requirements utilised for design assessment, the situation in the EU seems favourable in that high-level requirements have been established in a uniform fashion by the revised Nuclear Safety Directive 2009/71 (as revised by directive 87/2014), the WENRA reference levels and the common adherence to IAEA Safety Standards; this is complemented by industry's EUR, as discussed in detail in the Task 1 Report. This level of harmonisation should greatly facilitate the joint assessment. Remaining differences in detailed requirements, and in general regulatory approach and "philosophy" as well as in assessment methodology need to be discussed in advance and settled in the contractual arrangements. If necessary, it could also be considered that individual regulators, to an extent agreed beforehand, can utilise their own national requirements. They could still accept the design, based on "different grounds" by comparison to their peers; if necessary, they can express, in the CODA, some reservations on single points and define issues to be clarified in their subsequent national licensing process.

Finally, in terms of process, the task 3 advocates a realistic and pragmatic approach which does not necessitate a new legislative framework. The process relies on the factual importance of the joint outcome (the CODA). As practice with existing national and international pre-licensing processes has shown, participating regulators will not arbitrarily deviate from their final statement issued under the pre-licensing process even if this has no legal force; they will rely on it unless new information arises which casts doubt on the results and necessitates a new assessment. Likewise, no mandatory legal interface to national licensing processes is proposed. In this concept, the diversity of Member States' national frameworks can be taken into account as such and does not need to be addressed by changes in national legislation and regulation which would be difficult to achieve.

All in all, JODA would offer a voluntary and flexible but at the same time meaningful and effective tool for performance of a joint design assessment by the regulators of several EU Member States. However, if a given design is to be implemented in one Member State only or if implementation in further Member States is still uncertain, a JODA would not seem to provide an added value. Instead, there may be more suitable means for the regulator tasked with assessing the design – especially if it is the regulator of a somewhat smaller country – to derive benefits from international cooperation, e.g. to collaborate with a (EU or non-EU) regulator who has already licensed a power plant of this design and to make use of the reference plant concept.

The objective of Task 4, Approach for Benchmarking of LTO Programmes, is to propose an approach for benchmarking/assessing the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87. The task is aimed at addressing the present situation, i.e. although the NSD clearly established the requirement that the high level of safety is achieved in all MSs aimed at practical exclusion of releases that would have long term environmental effects, the level of fulfilment of the Articles 8a and 8b of the amended NSD by EU NPPs is not known.

Unlike in other industries, benchmarks in nuclear industry and in particular benchmarks related to nuclear safety are rarely undertaken. Even those that are used (e.g. WANO performance indicators for safety systems) are very narrowly focused and set against strictly defined definitions and criteria. To assess the safety level of an NPP against the NSD, there is a need to look at a plant safety in totality – design, internal and external events, human factors, ageing, etc. The study found that the best framework for such a "totality" would come from a PSR performed according to IAEA Safety Guide. In such an approach, the areas of review addressed by the LTO benchmark would consist of 14 PSR Safety Factors that cover all aspects important to safety of an operating NPP.

Benchmarking compares a status of an organization with a particular standard and shows where it is in relation to that standard. In Task 4, each EU NPP is to be benchmarked against Articles 8a and 8b NSD 2014/87. Since these Articles establish high level requirements (mainly related to what needs to be done) rather than defining specific safety level to be achieved, practical benchmarking criteria – LTOC need to be developed for each of the PSR Safety factors. The LTOC would be derived from applicable current safety standards and internationally recognised good practices, in line with the IAEA Safety Guides and in particular with WENRA Reference Levels. In the proposed benchmarking concept, relevant attributes of each NPP (including the original design and all subsequent safety improvements/measures, regardless whether they originate in a PSR, LTO upgrades, the stress test or national regulator or operator initiatives) are compared with LTOC providing three possible outcomes: LTOC fully met/ LTOC partly met/LTOC not met.

Task 4 outlined a proposed benchmarking approach, presents examples of LTOC for the review areas of Ageing and External hazards (sub-area of PSR Safety factor Hazard analysis) and provides results of a test application on two NPPs. The proposed approach for benchmarking LTO programmes includes an overall assessment of a plant against the above LTOC. This is a PSR-like global assessment that summarizes the outcomes of individual LTOC reviews and their interaction - indicating all significant shortfalls/deficiencies, safety improvements planned and the overall level of plant safety with respect to Articles 8a and 8b of the amended NSD 2014-87. The data used in the test application came from the Topical Peer Review on Ageing Management and EU Post-Fukushima Stress Test. Although on a limited sample, the test application supported the viability of the proposed LTO benchmarking method.

To facilitate achieving a harmonised high level of nuclear safety throughout the EU consistent with the NSD and as discussed by ENSREG, the proposed LTO benchmark approach (method) would need to be further developed under the auspices of ENSREG and/or WENRA. The steps of the development, suggested in the report, include development of practical LTOC for all 14 Safety Factors/ areas of review, a pilot study of the LTO benchmarking method implemented by 2 – 3 NPPs, and finalization of the methodology in a guidance document based on the experience from the pilot study.

Envisaged benefits of implementation of the proposed LTO Benchmarking approach would include a qualitative assessment of the safety level of individual EU NPPs against the high level requirements of the NSD, providing an overall picture of the safety level of EU NPPs, without intention to rank them, and a significant incentive to EU regulators and NPP operators to harmonize their PSR implementation in line with the IAEA PSR Safety Guide and thus with the WENRA Reference Level Issue P because it would facilitate their implementation of the LTO benchmark. The LTO benchmark outcome would further add to motivation for safety improvements where indicated, and enhance transparency and trust among the EU public that a harmonised high level of safety is being achieved across the EU Member States.

TASK 1 - Benchmarking of the European Utility Requirements

TENDER SPECIFICATIONS OF TASK 1

EU tender ENER/D2/2016-677 specifies the requirements for Task 1 as follows:

"E.U.R: The European Utility Requirements developed since early nineties by ENISS (European Nuclear Installations Safety Standards Initiative) within Foratom, the nuclear lobby. ENISS claims that the detailed safety requirements comply with the requirements set by regulators within WENRA or the IAEA. They constitute a comprehensive set of more than 5 000 technical requirements sorted by safety themes and also reactors type that aims at facilitating the licensing both from the vendor's point of view and from that of the safety authority point of view. Recently several vendors have requested ENISS to benchmark their designs against the EUR. This is a clear sign of the credibility they expect to gain to enter the EU market with such a label.

Deliverable: a benchmarking of the EUR documents against a) all applicable WENRA reference levels, b) IAEA standards and c) requirements of the amended directive on nuclear safety as detailed by ENER/JRC (in development). It will be submitted to both NPPs vendors and nuclear safety authorities for comments/endorsement".

1 DESCRIPTION OF THE PROCESS

Task 1 addresses the benchmarking of the European Utility Requirements (EUR) against international safety standards. The objective of this task is to corroborate the EUR organisation's claims that the EUR Revision E is in line with the requirements of the IAEA, WENRA and the amended Nuclear Safety Directive of the EU.

The work on the European Utility Requirements started in 1991 when major electricity producers joint effort to harmonize national efforts in the development, design and licensing of Light Water Reactor plants. The EUR organization started with 5 partners in 1991 and is comprising of 14 partners today. In the period from 1991 until today, several revisions took place, Revision A was published in 1994, Revision B in 1995, Revision C in 2001, Revision D in 2012 and the last Revision E in 2016. It is this last Revision E (whose development started only 2 years after the publication of the previous Revision D, in order to capture the lessons learned from the Fukushima Daiichi accident) which is subject of the review under the Task 1.

The aim of the European Utility Requirements is to promote harmonization of requirements for Generation III Nuclear Power Plants (NPPs) across Europe (and worldwide). Generation III NPPs are considered to be an evolution from the Generation II NPPs characterized by:

- Improvements in nuclear safety with additional redundancy and diversity and inclusion of passive systems;
- Having standardized design that would reduce the licencing time;
- Higher availability and operational life of typically 60 years;
- Increased fuel and thermal sufficiency; and
- More robust design.

The aim is to harmonize the requirements to which Light Water Reactor NPP to be built in Europe will be designed, built, commissioned, operated and maintained. In addition to harmonization that will bring also the economic benefits by reducing costs in design, commissioning and operation, the EUR requirements require NPPs to be designed in a way to have low impact on the environment and population by minimizing radioactive and chemical releases.

Revision E, being prepared with a view of incorporating lessons learned from the Fukushima Daiichi accident has incorporated also the revised post-Fukushima international safety standards such as the IAEA Safety Requirements SSR 2/1 Rev.1, WENRA Report on Safety of New NPP design, WENRA Report on Safety Reference Levels for Existing Reactors and the amended Nuclear Safety Directive. The level to which these standards have been incorporated into the EUR is the subject of this Task 1.

The European Utility Requirements consists of 4 Volumes:

- **Volume 1:** There are no requirements in Vol. 1; two new chapters have been added to Revision E: general information about E.U.R. policies and the new list of 53 "key issues" covering 190 requirements from Volume 2 (out of 5500) to be used for a new design pre-assessment. It contains also the list of acronyms and definitions used in the E.U.R. document.
- **Volume 2:** The most important Volume with 4500 requirements in 20 Chapters; the new structure of Chapter 2.1 (Safety Requirements) in Revision E was prepared to be in line with the IAEA Safety Requirements SSR 2/1 Rev.1.; it includes new classification approach to be in line with the IAEA Safety Guide SSG-30 and details the radiological Objectives and Targets to be respected, in line with WENRA objectives O2 and O3. Each requirement of Volume 2 is labelled as N, T or C (Nuclear Island, Turbine or Common) according to its scope of application.

Requirements in Vol 2 are divided into 20 Chapters. However, they can be grouped into 7 different subjects:

- The general safety approach (2.1)
- Design objectives: performance and safety targets (2.2)
- General design requirements: design basis, grid requirements, codes and standards, layout rules (2.3, 2.4, 2.5, 2.11)
- Functional requirements: systems, containment, I&C (2.8, 2.9, 2.10)
- Requirements relating to equipment: components, materials, I&C (2.6, 2.7, 2.10)
- Requirements relating to the design process and construction: design process, Quality Assurance, constructability, decommissioning (2.12, 2.13, 2.14, 2.15, 2.16)
- Requirements relating to assessment methodologies: Deterministic and Probabilistic approaches, performance, cost, Environmental Impact assessment (2.17, 2.18, 2.19, 2.20)

- **Volume 3:** Volume 3 is out of the scope of this project; this Volume contains several subsets presenting the results of all design assessments done by the E.U.R. Organisation since the establishment of E.U.R.
- **Volume 4:** New structure of the Vol. 4 in Revision E contains requirements applicable to the conventional part of an NPP and minor update of the text. As such it is not highly relevant to this project, but nevertheless all requirements, if any, with N or C have also to be addressed.

The significant changes in Revision E in comparison to earlier Revisions were is the following fields:

- After the Fukushima Daiichi accident both the IAEA and WENRA have updated their standards. In order to capture the lessons learned from the Fukushima Daiichi accident, and to align its standards with the revisions of IAEA and WENRA standards, EUR completely restructured its Chapter 2.1 on Safety requirements.
- Safety Classification was brought in line with the SSG-30 document of the IAEA.
- Criteria for limited impact for Severe Accidents were brought in line with the new WENRA objectives.
- External Hazards address two distinct levels, one on Design Basis External Hazards and Rare and Severe External Hazards.
- Seismic approach brought in line with the world best practices.
- I&C and HMI requirements brought in line with IEC Standard 61513
- Leak Before Break and Pipe Break Preclusion concepts introduced.
- Layout requirements and Grid Connection requirements updated with new regulations.
- State-of-the art PSA methodology and its applications included.

The first step in Task 1 was to establish the compliance of EUR Key issues from Volume 1 and EUR Chapter 2.1 from Volume 2. In this way a bulk of safety related requirements were covered.

The IAEA Safety Series start with the top document, SF-1 on Safety Fundamentals. The IAEA Safety Fundamentals consist of 10 safety principles. Under the Safety Fundamentals, the IAEA Safety Requirements are produced, with the objective to set up the requirements which are necessary to fulfil 10 safety principles.

The IAEA Safety Requirements are divided into two parts; General Safety Requirements (GSR) and Specific Safety Requirements (SSR).

There are 7 General Safety Requirements; Part 1: Governmental, Legal and Regulatory Framework for Safety, Part 2: Leadership and Management for Safety, Part 3: Radiation Protection and Safety of Radioactive Sources, Part 4: Safety Assessment of Facilities and Activities, Part 5: Predisposal Management of Radioactive Waste, Part 6: Decommissioning and Termination of Activities and Part 7: Emergency Preparedness and Response.

Specific Safety Requirements comprise of 6 volumes: NS-R-3 (Rev 1): Site Evaluation for Nuclear Installations, SSR-2/1(Rev 1): Safety of Nuclear Power Plants: Design, SSR-2/2 (Rev 1): Safety of nuclear power plants: Commissioning and Operation, SSR-3: Safety of Research Reactors, NS-R-5 (Rev 1): Safety of Nuclear Fuel Cycle facilities, SSR-5: Disposal of Radioactive Waste and SSR-6: Regulations for the Safe Transport of Radioactive Material.

Underneath of these Safety Requirements there is an entire fleet of corresponding Safety Guides which provide guidance on how to fulfil the higher level requirements. While not all Safety Requirements and corresponding Safety Guides are directly applicable to the set of European Utility Requirements, they have all been screened for their applicability in the performance of this task. The most relevant Safety Requirements to compare with the European Utility Requirements are in SSR-2/1 Rev 1 on NPP Design. All 82 detailed requirements for design captured in SSR-2/1 Rev 1 provide the most up-to-date, state of the art, internationally accepted safety standards for design of nuclear power plants.

The major changes in SSR-2/1 Rev 1 as compared to the earlier version, relate to the following thematic fields:

- Prevention of severe accidents by strengthening the design basis for the plant;
- Prevention of unacceptable radiological consequences of a severe accident for the public and the environment;
- Mitigation of the consequences of a severe accident to avoid or to minimize radioactive contamination off the site.

The Safety Fundamentals and 13 Safety Requirements documents have been screened for this benchmark exercise. The Safety Guides which are produced under the Safety Requirements are only the guidance documents explaining how to fulfil the corresponding safety requirements and as such were not directly applicable for this study. However in many cases they have been used as only there equivalent level of details is to be found. For example, the Specific Safety Guide SSG-30 on Safety Classification of Structures, Systems and Components in Nuclear Power Plants has been used as basis for the development of the equivalent classification in the European Utility Requirements and as such was of great importance for this benchmarking study.

WENRA is the Western European Nuclear Regulators' Association that comprises of 17 members and 9 observers. It has two main Working Groups; RHWG – Reactor Harmonization Working Group and WGWD – Working Group on Waste and decommissioning (there is also the WENRA Inspection Working Group (WIG) but it is of less importance for this project). After the Fukushima Daiichi accident they have created 5 ad-hoc sub-groups of RHWG to deal with: Natural hazards, Containment integrity, Accident management, Periodic safety review and Mutual assistance. WENRA RHWG has issued two documents that are of high relevance to this task:

- Report on Safety Reference Levels for Existing Reactors and
- Report on Safety of new NPP designs

The objective of the development of Safety Reference Levels was to increase harmonization within WENRA countries on safety requirements issued by the regulatory bodies and their implementation in existing nuclear power plants. Initially they identified 18 areas where harmonization was considered as necessary. After the Fukushima Daiichi accident an additional 19th issue has been included to cover natural hazards (extreme weather conditions, external flooding, and seismic events). Major changes after the Fukushima Daiichi accident have also been in the area of Design Extension Conditions DEC – Issue F, where clear differentiation between DEC without core melt (DEC A) and DEC with core melt

(DEC B) has been introduced and in addressing the DEC aspects for the spent fuel pool and multi-unit sites. NPP autonomy for a justified time has also been introduced.

The Safety Reference Levels' emphasis is on nuclear safety, primarily focusing on safety of the reactor core and spent fuel. They specifically exclude nuclear security and, with few exceptions, radiation safety. Their latest version represents, in addition to good practices in WENRA countries, objectives for safety improvements taking into account the lessons learned from the Fukushima Daiichi accident.

The intention of the document on Safety of new NPP design was to develop common positions on selected key safety issues and safety expectations for the design of new nuclear power plants. The document addresses:

- WENRA safety objectives for new NPPs
- Selected key safety issues divided into 7 Positions and
- Lessons learned from the Fukushima Daiichi accident (covering 6 conclusions on issues presented in the above 7 positions).

The EU Council Directive 2014/87/Euratom of 8 July 2014 brings the amendment to the Directive 2009/71/Euratom which established a Community framework for the nuclear safety of nuclear installations (it is commonly referred to as the amended Nuclear Safety Directive). The original Council Directive 2009/71/Euratom imposes obligations of the Member States to establish and maintain a national framework for nuclear safety and reflects the provisions of the main international instruments in the field of nuclear safety, namely the Convention on Nuclear Safety, as well as the IAEA Safety Fundamentals. The amendment came as after the Fukushima Daiichi accident the European Council called on the Commission to review, as appropriate, the existing legal and regulatory framework for the safety of nuclear installations and propose any improvements that may be necessary. The amended Nuclear Safety Directive, which includes also all the elements of the original Directive from 2009, was used to benchmark the European Utility Requirements.

The European Utility Requirements were taken as basis and compared to the IAEA, WENRA and EU standards. The grouping of requirements as done in Volumes 1, 2 and 4 was used. The work started by benchmarking the E.U.R. Volume 1 Chapter 4 Key issues requirements. These issues, 53 in total, have been selected to represent the principal requirements to be met by a LWR design to be built in Europe. They represent a selection of EUR requirements from other volumes and as such provide a high level overview.

This was followed by benchmarking Volume 2 Chapter 2.1 Safety requirements which are seen as the most relevant for the study. It was followed by benchmarking the rest of Volume 2, all together 20 Chapters, which represent 95% of relevant material. It was not seen practicable to have every single European Utility Requirement compared with other documents, as it would have constituted a large multidimensional matrix, which would be hard to manage. Grouping them thematically as they appear in individual Chapters was seen as a manageable effort without losing any information in following the process.

Volume 4 was next to being screened. All requirements in Volume 4 were marked as T (turbine) and as such not applicable to this benchmarking study as only those marked as N (nuclear) or C (common) were to be included into the study.

The next benchmarking involved WENRA Safety Reference Levels for Existing Reactors and WENRA Report on Safety of New NPP Designs. In this benchmarking it was seen as more convenient to verify if each requirement from WENRA documents was addressed in the EUR documents. Therefore in the Tables that are used in these two comparisons, the WENRA requirements are on the left hand side of the applicable Tables.

In this process, E.U.R. requirements were first compared with the relevant IAEA Safety Standards and WENRA documents. Wherever an observation of differences was detected it was adequately documented and each finding was accompanied by a technical justification.

The E.U.R.s were subsequently compared to the requirements of the amended Nuclear Safety Directive, with particular emphasis on new provisions in Articles 6, 8a, 8b, 8c, 8d and 8e,. They deal with the nuclear safety objective for nuclear installations (8a), its implementation (8b), initial assessment and periodic safety reviews (8c), on-site emergency preparedness and response (8d) and peer reviews (8e). The EURs compliance with the provisions of the amended Nuclear Safety Directive were then verified and documented.

Following this, the EUR requirements were briefly compared with the provisions of the Vienna Declaration on Nuclear Safety.

At the end of the exercise some general conclusions have been made. It is seen as worthwhile to highlight the most important elements which were introduced or highlighted in all the above standards after the Fukushima Daiichi accident. These elements have been reflected in their latest revisions. This has been decided in order to see the comparison and compliance in the most important (or at least the newest) safety issues. Therefore in this last chapter, we highlighted the comparison of European Utility Requirements with the international standards in the following selected key areas:

1. Plant States considered in the design of nuclear power plants;
2. Application of defence-in-depth and independence of levels of defence-in-depth;
3. The concept of practical elimination;
4. Early and large releases;
5. Postulated initiating events;
6. Internal and external hazards;
7. Aircraft crash;
8. Design extension conditions;
9. Safety analysis of the plant design;
10. Containment;
11. Emergency power supply;
12. Use of non-permanent equipment for accident management;
13. Integrated Management System vs. Quality Management System.

In these 12 selected key areas a comprehensive comparison has been made between the European Utility Requirements and international standards with full explanation of issues addressed and principles introduced.

The final product of Task 1 is a document presenting the comparison and compliance of European Utility Requirements with the EU amended Nuclear Safety Directive, the IAEA Safety Standards, WENRA Safety Reference Levels and WENRA Report on safety of new NPP design and Vienna Declaration on Nuclear Safety with appropriate technical justifications.

Note: throughout the document E.U.R. or EUR are used to mean European Utility Requirements and there is no distinction in both abbreviations.

2 BENCHMARKING OF VOLUME 1 (CHAPTER 4) OF E.U.R. AGAINST THE IAEA SAFETY STANDARDS

2.1 E.U.R. CHAPTER 1.4 EUR KEY ISSUES

The Volume 1 of EUR gives main Policies and Objectives with Introduction, policies, synopsis, key issues, definitions and acronyms. The most important part is Chapter 4 of Volume 1 which presents 53 key issues for the design of LWRs in Europe. Key issues are selected principal requirements that should be met by the LWR design to be built in Europe. All of them are also covered in Volume 2 but have been singled out in order to provide a high level overview that a new design can be quickly benchmarked against to establish the first impression on the design acceptance. The benchmarking against key issues should be performed by vendors as a pre-assessment of their design, as an initial step of the overall assessment. These 53 key issues reflect the need for the design to:

- Achieve high level of safety and environmental protection;
- Respect the limitations of human performance;
- Meet operational performance targets;
- Provide a robust investment proposition.

E.U.R Chapter 1.4 EUR KEY ISSUES	**IAEA Safety Standards**
Issue 1: Application of Defence-in-Depth EUR Section 2.1.1.4A This Issue is in **full compliance** with the international standards. IAEA SSR 2/1 treats Design Extension Conditions (DEC) as DEC A without core melt and DEC B with the core melt. The EUR equivalents are complex sequences as DEC A, and severe accidents as DEC B, which are treated in the DiD concept as Levels 3a and Level 4 respectively. Both EUR and SSR 2/1 address design basis accidents (DBA). Equivalents in the WENRA documents are Postulated single initiating events to cover DBA, Postulated multiple failure events for complex sequences or DEC A and Postulated core melt accidents for severe accidents or DEC B.	SSR 2/1 Req. 7 TECDOC-1791 Table 4

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 2: Design Basis Accidents and Complex Sequences 2.1.2.3 A-B-C 2.1.2.4.1 A-B-C-D-E This issue is in **full compliance** with the international standards. Design basis accidents described in the EUR are in full compliance with the Req. 19 of SSR 2/1. As discussed above under Issue 1, complex sequences are the equivalent of DEC without core melt and requirements presented in the EUR are in full compliance with the Req. 20 of the SSR 2/1 in parts dealing with DEC without the core melt. WENRA equivalents are described under the Issue 1.	SSR 2/1 Req. 19 para 5.24-5.26 SSR 2/1 Req. 20
Issue 3: Severe Accidents 2.1.2.4.2 A-B-C-D-E Issue 3 is in **full compliance** with the international standards. Requirements for severe accidents in the EUR are equivalent to the Req. 20 of the SSR 2/1 in its parts dealing with DEC with core melt. WENRA equivalents are described under the Issue 1.	SSR 2/1 Req. 20
Issue 4: Reference Source Term and PSA Evaluation of Source Term 2.1.2.4.3.2 A-B-C 2.1.2.4.3.4 A-B Issue 4 is in **full compliance** with the international requirements. SSR 2/1 requires containing the source term within the containment and as such, the containment and its safety features shall be able to withstand extreme scenarios that include also the core melt. Such scenarios shall be determined using engineering judgement and probabilistic safety assessments. EUR is more specific in this respect and requires a design-specific determination of the reference source term and reference severe accident. Design-specific issues are not addressed and any international standards.	SSR 2/1 para 2.14 SSR 2/1 Req. 20 para 5.30

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 5: Practical elimination 2.1.2.5 A-B-C 2.9.3.1.7.1B Issue 5 on practical elimination is in **full compliance** with the international standards. List of sequences to be practically eliminated are practically the same in EUR 2.1/2.5B and in the TECDOC-1791 Appendix 4. Principles of practical elimination are also the same in both documents i.e. EUR 2.1/ 2.5A-B-C and TECDOC-1791 Chapter 7. Practical elimination of sequences that could lead to the containment failure are equivalent in EUR 2.9/ 3.1.7B and SSR 2/1 Req. 20 para 5.30. More on practical elimination principles and compliance is written in the last chapter of this document under point 3 of Chapter 10.	SSR 2/1 para 2.13(4) footnotes 3 and 4 TECDOC 1791 Chapter 7 and Appendix 4 SSR 2/1 Req. 20 para 5.30
Issue 6: External Hazards 2.1.2.6.2.1 A-B 2.1.2.6.4 A-B-C-D-E-F-H The Issue 6 on External Hazards is in **full compliance** with the international standards. EUR document distinguishes between Design Basis External Hazards (DBEH) for which safety assessment is preformed conservatively and Rare and Severe External Hazards (RSEH) for which safety assessment is performed using realistic, Best Estimate approach. IAEA TECDOC-1791 uses the same distinction labelling them Design Basis External Events (DBEE) and Beyond Design Basis External Hazards (BDBEE), having the same meaning and requirements. More on internal and external hazards principles and compliance is written in the last chapter of this document under point 6 of Chapter 10.	SSR 2/1 Req. 17 TECDOC-1791 Chapter 9 NS-R-3 Chapters 3 & 5
Issue 7: Operational Staff Doses & Doses to the Public During Normal Operation and AOOs. 2.1.3.2.2 A-B 2.1.3.2.3 A-B The Issue 7 is in **full compliance** with the international standards. In the EUR values are derived on the basis of ALARA principle and are set at 0.3 mSv/year, whereas values in the ICRP and GSR Part 3 are set at 1 mSv/year for public exposure during NO and AOO.	GSR Part 3, SCHEDULE III-1 GSR Part 3, SCHEDULE III-3

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
For operational staff doses in NO and AOO the values are set at 5 mSv/year, whereas values in the ICRP and GSR Part 3 are set at < 50 mSv/year, with average over 5 years < 20 mSv/year. The values in EUR document are therefore stricter than those prescribed in the GSR Part 3 standard.	
Issue 8: Safety Objectives & Off-Site Targets for Accidents 2.1.3.3 A-B (without core melt) 2.1.3.4 A-B-E (with core melt) Issue 8 is in **full compliance** with the international standards. EUR specifies more precisely the requirements to be met for accidental conditions in terms of evacuation, sheltering, iodine prophylaxis, permanent relocation or long time restriction of food consumption, whereas the IAEA specifies only that in the event of core melt accidents, only emergency countermeasures that are of limited scope in terms of area and time are necessary.	TECDOC-1791, App.2, Table 5
Issue 9: Categorization of Safety Functions and Classification of SSCs. 2.1.5.1.1 A-B-C Issue 9 is in **full compliance** with the international standards. Fulfilment of three fundamental safety functions (reactivity control, cooling of fuel and confinement of radioactive material) is the basic requirement in all documents. Classification of structures, systems and components (SSC) is the same in EUR and SSG-30 documents.	SSR 2/1 Req. 18 para 5.23 SSR 2/1 Req. 11 SSG-30 para 2.3
Issue 10: Safety Classification 2.1.5.1.2 A-B-C 2.1.5.1.3 A-B 2.1.5.1.5 B-C 2.1.5.1.7 A 2.1.5.1.7.1 A Issue 10 on Safety Classification is in **full compliance** with international standards. Classification methods and rules follow the same principles in EUR and IAEA documents. Basically design provisions are classified in safety class 1, 2 or 3 according to the severity of consequences of their failures.	SSR 2/1 Req. 22 para 5.34 SSG-30 para 2.6; 2.12 SSG-30 para 3.23 SSR 2/1 Req. 9 SSG-30 paras 3.8 – 3.9 SSG-30 paras 3.23; 4.5

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
In general, EUR is following the SSG-30 Guide to the extent possible.	
Issue 11: Autonomy Objectives 2.1.6.7.1 B-C 2.1.6.7.2 A 2.1.6.7.3 A-B-F-G Issue 11 on Autonomy Objectives is in **in full compliance** with international standards and in this subject area provides additional guidance to designers in terms of prescribed time scales for fulfilment of actions. This section of EUR gives time limits in respect of operators and plant personnel actions, availability of ultimate heat sink and of power supply systems. All of these are also covered in IAEA SSR 2/1 but without concrete numbers in terms of time units (minutes, hours or days).	SSR 2/1 paras 4.11(d); 5.12; 5.13; 5.58; 6.33 SSR 2/1 Req. 53 SSR 2/1 Req. 68
Issue 12: Non-Permanent Equipment 2.1.6.8.3 A-B-C-E 2.1.6.8.4 A 2.1.6.8.5 C Issue 12 on Non-Permanent Equipment is **in full compliance** with the international requirements. SSR 2/1 addresses non-permanent equipment in para 6.28B for the removal of heat from the containment, para 6.45A for restoration of the electrical power supply and para 6.68 to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation. Chapter 10 of the TECDOC-1791 is entirely devoted to Non-Permanent Equipment for accident management and supports the requirements presented in the EUR document.	SSR 2/1 paras 6.28B; 6.45A; 6.68 TECDOC-1791 Chapter 10
Issue 13: Performance of Fuel Elements and Assemblies and Structural Capability of the Reactor Core 2.1.7.1.1.1 A 2.1.7.1.1.2 A-B 2.1.7.1.1.3 A-B-C 2.1.7.1.2 A Issue 13 on Performance of fuel elements and assemblies is **in full compliance** with the Requirements 43 and 44 of SSR 2/1 and recognizes the existence of "*the process of deterioration*".	SSR 2/1 Req. 43 SSR 2/1 Req. 44

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
It however does not specify which these processes are as does SSR 2/1 in para 6.1 where it states: 6.1. The processes of deterioration to be considered shall include those arising from: — Differential expansion and deformation; — External pressure of the coolant; — Additional internal pressure due to fission products and the build-up of helium in fuel elements; — Irradiation of fuel and other materials in the fuel assembly; — Variations in pressure and temperature resulting from variations in power demand; — Chemical effects; — Static and dynamic loading, including flow induced vibrations and mechanical vibrations; — Variations in performance in relation to heat transfer that could result from distortion or chemical effects. Allowance shall be made for uncertainties in data, in calculations and in manufacture. However, the design limits, addressed in para 6.2 of SSR 2/1 are described in much greater detail in sections 2.1.7.1.1.1 for Normal operation, 2.1.7.1.1.2 for Anticipated Operational Occurrences and in 2.1.7.1.1.3 for Design Basis Accidents.	
Issue 14: Containment Bypass Accidents 2.1.7.3.2.1 A-B-C Issue 14 on Containment bypass accidents is in **full compliance** with the international standards. EUR addresses the effectiveness and means to reduce the probability of containment bypass accidents and similar reasoning can be found in TECDOC-1791 for practical elimination of containment bypass during severe accidents.	SSR 2/1 Req. 55 and 56 TECDOC-1792 Ch. 7 and Appendix 4 Ch. 10 on practical elimination
Issue 15: Instrumentation and Control Systems 2.1.7.4.6 A-B-C-D 2.1.7.4.7 A Issue 15 on Instrumentation and Control Systems is if **full compliance** with the international standards. The EUR Sections 2.1.7.4.6 Main control room and 2.1.7.4.7 Emergency control room are **identical** with the SSR 2/1 Requirements for Control room and Supplementary control room, only the names for both control rooms are different. SSR 2/1 requirement under 6.41 that "*taking appropriate measures and providing adequate information for the*	SSR 2/1 Req. 65 SSR 2/1 Req. 66

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
protection of occupants against hazards also apply for the supplementary control room at the nuclear power plant", is not explicitly stated in the E.U.R. requirements for the Emergency control room but it is taken to be covered by other requirements for the Emergency control room.	
Issue 16: Electrical Power Supply in AOO and Accident Conditions 2.1.7.5 A-B-E-F-G Issue 16 on Electrical power supply in AOO and accident conditions is in **full compliance** with the international standards. Issue 16 addresses design for the Loss of Off-site Power (LOOP). Both EUR and SSR 2/1 require: - Emergency Power Supply to cope with LOOP in AOO and DBA and - Alternate Power Supply to cope with LOOP in DEC. Both should be independent and physically separated from each other. More on this issue is written in the last chapter of this document under point 11 of Chapter 10.	SSR 2/1 Req. 68
Issue 17: Fuel Storage and Handling System. Stored Fuel Heat Removal 2.1.7.7 A 2.1.7.7.1 A-B-C-D-E 2.8.2.2.2.1 F-I Issue 17 on Fuel storage and handling systems and Stored fuel heat removal is in **full compliance** with the international standards. The requirements in EUR use practically the same wording as in SSR 2/1 Req. 80 and requires that the systems be such to ensure integrity and properties of the fuel are maintained at all times during fuel handling and storage.	SSR 2/1 Req. 80 SSR 2/1 - Req. 80 para 6.68A
Issue 18: Intentional Aircraft Crash 2.1.8.3.3 A-B-D-E-F Issue 18 on Intentional aircraft crash is in **full compliance** with the international standards as it provides much more detail on the requirements in such cases then the IAEA and WENRA (Position O3.7). More detail is provided under the point 7 in the last chapter 10 of this document.	NS-R-3 Rev. 1 paras 3.44 – 3.47 NSS-10

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 19: Duration of Cooldown 2.2.2.1.1 A-B Issue 19 on duration of cooldown is very specific in term of temperatures and duration hours and as such much more detailed than general requirements that can be found in international standards. Being more specific than other standards it can be concluded that they are in **full compliance.**	N/A
Issue 20: Duration of Start-up and Loading 2.2.2.1.2 B-C Issue 20 is also very specific in terms of necessary time intervals and as such much more detailed than general requirements that can be found in international standards. Being more specific than other standards it can be concluded that they are in **full compliance.**	N/A
Issue 21: Type of Fuel, Refuelling Cycle and Expected Thermal margins 2.2.3.1 A-B 2.2.3.4.1 A 2.8.2.2.1.1 A 2.2.3.2 A Issue 21 is also very specific in terms of the fuel types and as such much more detailed than general requirements that can be found in international standards. Being more specific than other standards it can be concluded that they are in **full compliance.**	N/A
Issue 22: Low Boron Capability 2.2.3.7 B 2.2.3.5 A Issue 22 deals with the requirement to have negative power reactivity coefficient during DBA and DEC. It can be found in the IAEA NS-G-1.12 (Design of the Reactor Core for NPPs) in its Appendix I which describes the reactivity coefficients. Such requirement is in **full compliance** with the international standards.	NS-G-1.12 Appendix I

E.U.R Chapter 1.4 EUR KEY ISSUES	**IAEA Safety Standards**
Issue 23: Load Following and Manoeuvring Capability 2.2.3.4.1 A 2.2.3.8 A Issue 23 is in **full compliance** with international standards. NS-G-1.12 in its paras 3.103 and 3.107 cover all reactor core calculation analysis for all modes of reactor operation for the entire operating cycle.	 NS-G-1.12 paras 3.103 and 3.107
Issue 24: Spent Fuel Storage Capacity 2.2.5.1 A Issue 24 addresses <u>the capacity</u> of the spent fuel storage. These requirements are more detailed then the general Spent Fuel Pool requirements described in SSR 2/1 Req. 80 which is more connected with the Issue 17. The level of detail for the capacity of the spent fuel pools as described in Issue 24 cannot be found in the international standards and therefore being more detailed it can be concluded that Issue 24 is in **full compliance.**	 SSR 2/1 Req. 80
Issue 25: Overall Availability, Outage Duration 2.2.7.2.1 A 2.2.7.2.2 B In issue 25, the overall availability factor target is given as 90% and duration of maintenance outage (16 days), T-G overhaul (24 days) and ISI outage (36 days). Such time duration specifications are greater level of detail than specifications found in international safety standards and as such can be classified as **full compliance.**	N/A
Issue 26: Response to Los of External Grid 2.2.2.11 A Issue 26 requires that plant be switched to house load operation without reactor trip in case of the loss of external grid. As such level of detail is not prescribed in the safety requirements it can be concluded that Issue 26 is in **full compliance.** (For Westinghouse plants such tests were performed under SU-PA-12 procedure, but were never made a requirement due to stresses that it imposes on the entire installation).	N/A

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 27: Grid Interface 2.2.2.8 A 2.3.2.2.1 A 2.3.2.2.2 A 2.3.3.1.2 A Requirement 31 para 8.11 from SSR 2/2 requires that operator shall have arrangements with the grid operator to ensure connections of the plant to the external grid. The level of detail in Issue 27 is much greater than that in international standards and therefore the **full compliance** can be established for Issue 27.	N/A (SSR 2/2 Req. 31 para 8.11)
Issue 28: Plant Design Life 2.4.2 A-C-E Design life in EUR is established as 60 years. As international safety standards do not prescribe any duration it can be concluded that Issue 28 is in **full compliance.**	N/A
Issue 29: Seismic Design Levels and Soil Properties for Seismic Analysis 2.4.1.2.1.3 A 2.4.6.4.2 A-B-C NS-R-3 requires for earthquakes to be analysed, historical data recorded and hazards assessed (para 3.1 – 3.4). Similarly for soil properties, hazards need to be assessed (para 3.38 – 3.43). EUR gives much more detail then NS-R-3 Rev 1 in specifying requirements for seismic design and soil properties and as such is in **full compliance** with the international standards.	NS-R-3 paras 3.1 – 3.4 (seismic) NS-R-3 paras 3.38 – 3.43 (soil)
Issue 30: Codes and Standards 2.5.2 A-B 2.5.3 B Levels of rules as presented in the EUR Chapter 2.5 in Figure 1 are **in full compliance** with the relevant parts of IAEA GSR Part 1 on Governmental, Legal and Regulatory Framework for Safety as well as WENRA requirements and EU directives.	GSR Part 1

E.U.R Chapter 1.4 EUR KEY ISSUES	**IAEA Safety Standards**
Issue 31: Materials Selection 2.6.3.2 A Requirements set out in Chapter 2.6 of E.U.R. are much more detailed then Requirements in the IAEA Safety Standards and are **in full compliance** with Requirements stipulated in SSR 2/1 and SSR 2/2. More details are addressed in Chapter 4 of this document under the benchmarking of the EUR Chapter 2.6 with the international requirements.	SSR 2/1 SSR 2/2
Issue 32: Fuel Compatibility 2.7.11.4 A 2.2.4.1 A E.U.R. are much more detailed in functional requirements for fuel assembly design for nuclear power plants and are **in full compliance** with other reviewed safety requirements, including the IAEA Safety Guides.	NS-G-1.12
Issue 33: Diesels Generators I&C Principles 2.7.13.2.2 A-B-C-D-E SSR 2/1 addresses the requirements for emergency power supply (for AOO and DBA) and alternate power source (for DEC). It does not however go into details on when and how they will be started. SSR 2/1 requirements in this area correspond to the EUR requirements made in 2.1.7.5 A,B,C,E,J,F,I,D,K. E.U.R. are much more detailed in functional requirements for components to be used in the design of nuclear power plants like those in EUR 2.7.13.2.2A-B-C-D-E. As such, issue 33 is **in full compliance** with other reviewed safety requirements, including the IAEA Safety Guides.	SSR 2/1 Req. 68
Issue 34: Reliability of Shutdown Capabilities & Shutdown Margin 2.8.2.1.1.5 A-B-D 2.8.2.1.1.6 B SSR 2/1 Req. 46 specifies requirements for reactor shutdown. EUR Sections 2.8.2.1.1.5 and 2.8.2.1.1.6 are in **full compliance** with SSR 2/1 Req.46 requiring two independent and diverse shutdown systems and having adequate shutdown margins. These requirements apply only to PWRs.	SSR 2/1 - Req. 46 paras 6.7 – 6.12

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 35: Environmental protection 2.8.2.2.4.3 A 2.2.5.2 A-B-C-D Issue 35 deals primarily with the radioactive waste and provides practical and detailed requirements in terms of waste volume and segregation. It is in **full compliance** with international standards.	SSR 2/1 Reqs. 12 and 78 SSR 2/2 Req. 21 GSR Part 5 Reqs. 4, 10, 11 and 12
Issue 36: Reactor Cooling System Arrangements and Mid-Loop Operation 2.8.3.3.1.3.2 A-AA-AB-AC-AD SSR 2/1 Req.47 requires that design be such to avoid loss of coolant, limitation of material flaws and embrittlement and standards for component parts within the RCS. SSR 2/1 Req. 48 requires overpressure protection. NS-G-1.9 gives more precise guidance on the design of RCS. EUR Issue 36 is more precise in requiring specific conditions in the RCS and mid-loop operation and these conditions are in **full compliance** with the international standards.	SSR 2/1 Reqs. 47-48 NS-G-1.9
Issue 37: Independence and separation of System Divisions 2.8.4.1.5 A-B-C SSR 2/1 Req. 21 deals with physical separation and independence of safety systems and SSR 2/1 Req. 64 requires separation of protection systems and control systems. A term “elements of safety systems” is used in SSR 2/1 which corresponds to the term used in the EUR document as “system divisions”. Requirements on separation and independence are equivalent in both documents and therefore Issue 37 is in **full compliance** with the international standards.	SSR 2/1 Reqs. 21 and 64
Issue 38: Containment System General Configuration 2.9.2.1 A-B-C Both NS-G-1.10 and EUR define primary and secondary containment in the same way (NS-G-1.10 para 4.147). Acceptable technologies for the primary containment are also the same in both documents (Annex I; I-2). Issue 38 is therefore in **full compliance** with international standards.	NS-G-1.10 paras 4.6, 4.147 Annex I; I-2

E.U.R Chapter 1.4 EUR KEY ISSUES	**IAEA Safety Standards**
Issue 39: Secondary Containment Performance 2.9.2.2.1.3 A 2.9.2.2.1.4 A NS-G-1.10 specifies that in accidental conditions the secondary containment should be able to withstand conditions i.e. hold up fission products similar to EUR requirement. Further, NS-G-1.10 specifies that design should minimize the secondary containment bypass leakage whereas EUR puts the limit to 10% and is therefore more precise in establishing the limit. Issue 39 is therefore in **full compliance** with the international requirements.	 NS-G-1.10 paras 4.33 and 4.34 NS-G-1.10 para 3.23
Issue 40: In-Vessel and Ex-Vessel Debris Cooling 2.9.3.1.8.2 A 2.9.3.1.8.3 A-B-C Both SSR 2/1 and EUR require that containment should withstand core melt scenarios. EUR in addition requires cooling of corium in or ex vessel. Further, both IAEA and EUR documents require limitation of core-concrete interaction. As such, Issue 40 is in **full compliance** with international standards.	 SSR 2/1 Req. 20 para 5.30 NS-G-1.10 para 6.12
Issue 41: Containment Spray System Configuration 2.9.4.1.2.1 B In addition to other requirements EUR stipulates that the CSS design should avoid recirculation of contaminated water outside the containment system. Such additional requirement has not been found elsewhere and it can be concluded that Issue 41 is in **full compliance** with the international standards.	SSR 2/1 Req. 58 para 6.28 NS-G-1.1- paras 4.90 – 4.95
Issue 42: High-Energy Lines within the Secondary Containment 2.9.4.2.2 B Requirements for high energy pipe breaks are addressed in both NS-G-1.10 and EUR documents in a similar manner and it can be concluded that Issue 42 is in **full compliance** with the international standards.	NS-G-1.10 paras 3.3(1); 3.20; 4.57(b)

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 43: Overall I&C and System Life Cycle 2.10.6.1 A-B Issue 43 requires that the design of the overall safety I&C be compliant with International Electrotechnical Commission IEC 61513 document. Similar specifications for a systematic approach can be found also in SSG-39. As such, Issue 43 is in **full compliance** with the international standards.	SSG-39 paras 2.10 – 2.23 IEC 61513
Issue 44: Output Documentation of I&C Requirements Specification Phase 2.10.6.3 B EUR 2.10.6.3B and SSG-39 para 2.90 provide the list of specifications to be provided as a minimum. Lists are not the same but cover the same topics and therefore Issue 44 is in **full compliance** with the international standards.	SSG-39 para 2.90
Issue 45: Qualification Programme 2.1.8.1.2 A 2.10.7.2.2.7 A-B-C 2.4.8.1.1 A-B-C EUR 2.1.8.1.2 A is identical with SSR 2/1 Req. 30 and requires a qualification programme to be established for Items Important to Safety. The same applies for I&C qualification which is described in SSG-39. Seismic qualification of electrical equipment refers to IEC 60980 standard in both EUR 2.4.8.1.1 and in SSG-39 Table I-1. Issue 45 is therefore in **full compliance** with the international standards.	SSR 2/1 Req. 30 SSG-39 paras 6.77 – 6.134 SSR 2/2 Req. 13 paras 4.48 – 4.49
Issue 46: Fire 2.11.2.4.2 A-J-K-L-N-P-Q EUR requirements on Fire are more detailed than SSR 2/1 Req. 74 on Fire protection systems as well as those in NS-G-2.1 and therefore it can be concluded that Issue 46 is in **full compliance** with the international standards.	SSR 2/1 Req. 74 NS-G-2.1 para 2.6 – 2.7
Issue 47: Information Management System 2.12.12 A Issue 47 requires designer to use appropriate computer hardware and software from the early stages of the deign process. Such requirements are not found in international standards and hence this issue can also be classified as **full compliance.**	N/A

E.U.R Chapter 1.4 EUR KEY ISSUES	**IAEA Safety Standards**
Issue 48: Construction time targets/ Project time schedule optimization 2.13.2.2 A Construction and project schedule and targets are not addressed in international safety standards and being additional requirements, Issue 48 can be classified as **full compliance.**	N/A
Issue 49: Long Term Safety & Provisions for Replacement of Major Components 2.1.8.1.1 A-B 2.1.8.1.3 A-B 2.14.5 A EUR 2.1.8.1.1 and SSR 2/1 Req. 29 are identical and deal with calibration, testing, maintenance, repair and replacement. EUR 2.1.8.1.3 and SSR 2/1 Req. 31 are also identical and deal with the design life of items important to safety. Therefore Issue 49 is in **full compliance** with the international standards.	 SSR 2/1 Req. 29 SSR 2/1 Req. 31 SSR 2/2 Reqs. 14 and 16
Issue 50: Quality assurance 2.15.1 A-E-F-M E.U.R Chapter 2.15 recognizes the developments in safety management and clearly states that "the overall QAP of the project, as well as management processes and QAPs of Contractors, and review organisations, shall be in accordance with the requirements of the nuclear specific safety standard IAEA GSR Part 2: Leadership and Management for Safety (2016)" (EUR 2.15.1E). It further states, that the Quality Assurance programme is part of the Management system (EUR 2.15.1 A) and further, on few places emphasises, for example, that "where potential contradiction between ISO 9001 (2015) – being a general quality standard for products and services - and IAEA GSR Part 2 (2016) – being nuclear safety standards - exists in the implementation phase, the nuclear specific features of the IAEA GSR Part 2 (2016) standard shall be mandatory QAP input for nuclear related Structures, Systems and Components (SSCs)" (EUR 2.15.1 F). As it clearly refers to compliance with the GSR Part 2 principles it can be concluded that Issue 50 is in **full compliance** with the international standards. More on this issue can be found in the last chapter 10 under point 13 Integrated Management System vs. Quality Management System.	GSR Part 2 GS-G-3.1 GS-G-3.5

E.U.R Chapter 1.4 EUR KEY ISSUES	IAEA Safety Standards
Issue 51: Design for Decommissioning 2.16.3.J Issue 51 formulates requirements that should be met in the design stage in order to facilitate easier decommissioning. It describes system independence for easier dismantling, surface characteristics for contaminated components and accessibility for removal of components. IAEA GSR Part 6 and WS-G-2.1 determine conditions to be fulfilled already in the design stage and E.U.R. Issue 51 can be considered to be in **full compliance** with the international standards.	GSR Part 6, Chapter 2, Req. 12 SSR 2/1 Req. 12 para 4.20 WS-G-2.1 paras 4.1 - 4.4
Issue 52: Probabilistic Safety Analysis 2.1.4.3 A-B-C-D 2.17.2.1 B EUR 2.1.4.3 and SSR 2/1 Req. 42 para 5.76 are identical and require PSA to be performed for all modes of operation and for all plant states, including shutdown. Provisions of EUR 2.17.2.1 are covered in SSG-3 Chapter 5 and it can be concluded that Issue 52 is in **full compliance** with the international standards.	SSR 2/1 Req. 42 para 5.76 GSR Part 4 Req. 15 para 4.55 SSG-3 Chapter 5
Issue 53: Human Factors 2.1.8.2 C-F 2.17.3.5 A-B 2.14.1.1 F Human factors are addressed in SSR 2/1 in Requirement 32 dealing with design for optimal operator performance (paras 5.53 – 5.62). These are the main human factor and human-machine interface requirements to be considered in the design stage. In addition, SSR 2/1 Requirements 65 and 66 address Main Control Room and Supplementary Control Room (Emergency Control Room in the EUR document) requirements. All these aspects of Human Factors in design have been adequately addressed by Issue 53 and are in **full compliance** with the international standards.	SSR 2/1 Reqs. 32, 65 and 66 SSG-3 para 5.96 – 5.113 SSR 2/1 Req. 32

CONCLUSIONS

All 53 key issues presented in Volume 1, Chapter 4 are selection of issues from Volume 2 “Generic and Nuclear Island Requirements” with almost half of them being taken from Chapter 2.1 which covers “Safety Requirements”. Selection of 53 key issues has been made with the objective to establish the first impression on the design acceptance. Benchmarking against the 53 key issues is to be performed by vendors as a pre-assessment of their design. Such assessment should build a confidence that the design will achieve high level of safety and environmental protection, acknowledge the limitations of human performance and meet operational performance targets. Benchmarking against international standards has shown a **full compliance** of all 53 key issues with international standards.

3 BENCHMARKING OF VOLUME 2 OF E.U.R. AGAINST THE IAEA SAFETY STANDARDS

3.1 E.U.R. CHAPTER 2.1 SAFETY REQUIREMENTS

E.U.R Chapter 2.1 is intended to provide the basic requirements to be placed by utilities on the design organization. The objective of this chapter is to provide the assurance that if complied with, then the design will be ready to be licensed, built and operated in the majority of European countries with only minor modifications. In developing this chapter, the authors have heavily relied on the IAEA Safety Requirements SSR 2/1, Rev 1: Safety of Nuclear Power Plants; Design and GSR Part 4, Rev 1: Safety Assessment for Facilities and Activities as well as on the IAEA Specific Safety Guide SSG-30 on Safety Classification of Structures, Systems and Components in Nuclear Power Plants. The authors also claim the full compliance with the amended Nuclear Safety Directive and with the WENRA RHWG Report on Safety of new NPP design and updated WENRA Reference Levels.

As a result of the Fukushima Daiichi accident the requirements on extended plant autonomy and design features which allow connection of the non-permanent equipment in case of rare and severe external hazards have been introduced in EUR Chapter 2.1 of the latest Revision E.

Another important addition from lessons learned from the Fukushima Daiichi accident was that the fundamental safety objective for the design of future nuclear power plants is to restrict the off-site radiological releases in case of a severe accident to an acceptable level for the environment and the public. This would in turn also provide technical basis for a possible simplification of the off-site emergency planning.

The safety requirements presented in this chapter are predominantly based on deterministic methods and supplemented where necessary by probabilistic methods using appropriate numerical targets and analysis.

In the table below as well as throughout this document, E.U.R. requirements have been compared with the equivalent topics in the IAEA Safety Standards. Not all IAEA items are in the form of requirements, some of them are found in Safety Guides or other IAEA documents. Nevertheless the authors of this report have tried to cover all E.U.R. requirements with the equivalent IAEA statements regardless of the level that they have in the IAEA Safety Standards hierarchy.

The equivalents are presented in the below tables. Only the observations of any differences are then commented in the footnote statements. These should be viewed as working technical notes that will be summarized in a special chapter 10 on General Conclusions at the end of this report.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
1. GENERAL SAFETY REQUIREMENTS(1)	
1.1 General safety objectives	TITLE
1.1A	SSR 2/1 paras 2.2;2.6
1.1B	SSR 2/1 Req. 5; para 2.3

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
1. GENERAL SAFETY REQUIREMENTS[(1)]	
1.1C	SSR 2/1 para 2.8
1.1D	SSR 2/1 Req. 5 para 4.3; paras 2.11; 2.13(4); 4.3
1.1E	SSR 2/1 para 2.13(4)
1.2 Fundamental safety functions	TITLE
1.2A	SSR 2/1 Req. 4
1.2B	SSR 2/1 Req. 4 para 4.1
1.2C	SSR 2/1 Req. 4 para 4.2
1.3 Physical barriers	TITLE
1.3A	SSR 2/1 para 4.11(a)
1.3B	SSR 2/1 para 2.9
1.3C	SSR 2/1 para 4.12
1.3D	SSR 2/1 para 4.10
1.4 Application of Defence-in-Depth	TITLE
1.4A	SSR 2/1 Req. 7
1.4A1	SSR 2/1 para 2.12
1.4A2	SSR 2/1 para 2.13(1)
1.4A3	SSR 2/1 para 2.13(2)
1.4A4	SSR 2/1 para 2.13(3) TECDOC-1791 SSR 2/1 para 2.14
1.4A5	SSR 2/1 para 2.13(4)
1.4A6	SSR 2/1 para 2.13(5)
1.4A7	SSR 2/1 para 2.12
1.4AA	SSR 2/1 para 2.13 TECDOC-1791
1.4B	SSR 2/1 para 2.10
1.4C	SSR 2/1 paras 4.11(b),(c),(d),(e)
1.4D	SSR 2/1 para 4.13A
1.4E	SSR 2/1 para 4.13

[(1)] 2.1.1 GENERAL SAFETY REQUIREMENTS

General safety requirements in E.U.R. Chapter 2.1, Section 1 describing General safety objectives, Fundamental safety functions, Physical barriers and Application of Defence-in-depth are in **full compliance** with the same principles outlined in the IAEA Safety Requirements SSR 2/1 on Safety of Nuclear Power Plants: Design.

In the area of Application of defence-in-depth, the IAEA SSR 2/1 in paragraph 2.14 specifies that "*A relevant aspect of the implementation of defence in depth for a nuclear power plant is the provision in the design of a series of physical barriers,* ***as well as a combination of active, passive and inherent safety feature that contribute to the effectiveness of the physical barriers in confining radioactive material at specific locations…".*** These provisions are not explicitly mentioned in the E.U.R. document, except in 2.1.1.4.A4 where Inherent Safety Characteristics are required in Level 3a and 3b of defence-in-depth application. The E.U.R. document requires the above combination of active, passive and inherent safety features in particular for Level 3a and 3b. It is however evident that they have also been implicitly included elsewhere bearing in mind that the level of detail in the E.U.R. document is much greater in this subject area then in the IAEA SSR 2/1.

In the IAEA SSR 2/1 Requirement 7 on Application of defence in depth, para 4.11 (f), states that design "*shall provide* ***multiple means*** *for ensuring that each of the fundamental safety functions is performed, thereby ensuring effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation".* The requirements from SSR 2/1 - 4.11 (a), (b), (c), (d), and (e), are captured in E.U.R. document in sections 2.1.1.3A and 2.1.1.4C. The requirement SSR 2/1 4.11(f) which follows the above requirements (a) to (e) is however not reflected immediately in corresponding E.U.R. sections. Requirement for multiple means for ensuring that each of the fundamental safety functions is performed is however implicit in detailed elaboration of requirements for SSCs design specifications.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	**IAEA Safety and Security Standards**
2. DESIGN CONDITIONS	
2.1 Plant states	SSR 2/1 Req. 13 para 5.1
2.2 Postulated initiating events	SSR 2/1 Req. 16 paras 5.5 – 5.15 SSR 2/1 Req. 13 para 5.1
2.3 Design basis accidents	SSR 2/1 Req. 19
2.3A	SSR 2/1 Req. 19
2.3B	SSR 2/1 Req. 19 para 5.24
2.3C	SSR 2/1 Req. 19 para 5.25
2.3D (2)	SSR 2/1 Req. 19 para 5.26
2.4 Design extension conditions (3)	SSR 2/1 Req. 20; SSR 2/1 Req. 6, para 4.6
2.4.1 Complex sequences	SSR 2/1 Req. 20
2.4.2 Severe accidents	SSR 2/1 Req. 20
2.4.3 Severe accidents in-containment source term quantification	SSR 2/1 Req. 20 para 5.30
2.5 Practical elimination	SSR 2/1 para 2.13(4) footnotes 3 and 4 TECDOC-1791 Chapter 7 and Appendix 4
2.6 Internal and external hazards	SSR 2/1 Req. 17 NS-R-3 Chapters 3 & 5
2.6.1 Considerations of internal hazards	SSR 2/1 para 5.16
2.6.2 Identification of external hazards	SSR 2/1 paras 5.17 – 5.22

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
	TECDOC-1791 Chapter 9
2.6.2.1 Identification of external hazards for standard design	NS-R-3 Chapters 3 & 5
2.6.2.1.1 Accidental aircraft crash	NS-R-3 paras 3.44 – 3.47
2.6.2.2 Identification of external hazards for site-specific evaluation	NS-R-3 Chapters 3 & 5.1A
2.6.3 Screening for site-specific evaluation	NS-R-3 2.14 – 2.21
2.6.4 External hazard design rules for standard design and site-specific design	NS-R-3 Chapter 3.

(2) 2.1.2.3 Design basis accidents

In terms of Deterministic safety analysis to be used for Design Basis Accidents (DBA) and Design Extension Conditions (DEC), there might be a slight difference between E.U.R. and SSR 2/1 requirements but it is more a matter of interpretation then actual inconsistency.

E.U.R. requires in 2.1.4.2.3A that for AOO and DBA three types of deterministic safety analysis shall be used:

- Best Estimate Plus Uncertainty (BEPU)
- Best Estimate combined with conservative Initial and Boundary (I&B) conditions
- Full conservative analysis: conservative computer codes with conservative I&B conditions

In comparison, SSR 2/1 in 5.26 requires that *“the DBA shall be analysed in a conservative manner. This approach involves postulating certain failures in safety systems, specifying design criteria and using conservative assumptions, models and input parameters in the analysis”.*

For Design Extension Conditions (DEC), both E.U.R. (2.1.4.2.3B) and SSR 2/1 (5.27) allow Best Estimate approach to be used. In addition, SSR 2/1 in 5.27 footnote 13 suggests that “*more stringent approaches may be used according to States’ requirements”.*

Note that SSR 2/1 requires that analysis shall be carried out in a **conservative manner.** The IAEA Safety Guide SSG-2 on Deterministic Safety Analysis already recognized that some sequences treated using purely conservative inputs do not necessarily produce conservative results and therefore cannot be considered as calculations being performed in a conservative manner. Many regulatory bodies throughout the world are accepting the three options described in E.U.R. 2.1.4.2.3A and SSG-2 as valid licensing basis, recognizing, that such calculations are performed in a conservative manner. Therefore it can be concluded, the both E.U.R. and SSR 2/1 are **in full compliance** with each other in the way deterministic safety analysis are performed.

(3) 2.1.2.4 DEC – Design Extension Conditions

Definitions of the design extension condition are the same in both, the E.U.R. document and the IAEA SSR 2/1 Requirements and all requirements that relate to DEC **are in full compliance,** in spite of different titles that are being used for conditions that comprise DEC. In the E.U.R document DEC consists of *“complex sequences” and “severe accidents”,* whereas in the IAEA SSR 2/1 DEC comprise of *“events without significant fuel degradation” and “events with core melting”.* In spite of different names, the concepts are practically the same. In its definition the complex sequences consist of events “which have the potential to lead to significant releases but do not involve core melt” and the definition of severe accident stipulates the events involving significant core melt which have the potential to lead to significant releases.

Another minor difference can be identified between the SSR 2/1 Requirement 20, para 5.30 which stipulates that "*the containment and its safety features shall be able to withstand extreme scenarios that include, among other things, melting of the reactor core. These scenarios shall be selected by using engineering judgement and input from probabilistic safety assessments".* E.U.R. in its Chapter 2.9 on Containment in section 2.9.3.1.8.1C states that "*the primary containment design shall be such that the primary containment can withstand any of the severe accidents addressed in DEC, without operator action during the first 12 hours from the beginning of severe accident conditions".* It is considered that both requirements are **in compliance** as E.U.R. only specifies conditions in more detail allowing for no operator action for a given time period.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	**IAEA Safety and Security Standards**
3. QUANTITATIVE SAFETY OBJECTIVES	
3.1 Overall approach to targets	INSAG-12 para 2.2
3.2 Radiological impact during Normal Operation and Incident Conditions	TECDOC-1791, App.2, Table 5
3.2.1 Radioactive discharge criteria	GSR Part 3, Req. 13, 3.33; Req. 17 para 3.52
3.2.2 Doses to the public during Normal Operation and Anticipated Operational Occurrences(4)	GSR Part 3, SCHEDULE III-3
3.2.3 Operational staff doses during Normal Operation and Anticipated Operational Occurrences(4)	GSR Part 3, SCHEDULE III-1
3.3 Safety objectives and off-site release targets for accidents without core melt	TECDOC-1791, App.2, Table 5
3.4 Safety objectives and off-site release targets for accidents with core melt	TECDOC-1791, App.2, Table 5
3.5 Probabilistic safety targets	INSAG-12 para 100

(4) Doses to that public and Operational staff doses during Normal Operation and Anticipated Operational Occurrences are much stricter in the E.U.R. then in the IAEA GSR Part 3 and ICRP 103 recommendations.

In the E.U.R. values are derived on the basis of ALARA principle and are set at 0.3 mSv/year, whereas values in the ICRP and GSR Part 3 are set at 1 mSv/year for public exposure during NO and AOO.

For operational staff doses in NO and AOO the values are set at 5 mSv/year, whereas values in the ICRP and GSR Part 3 are set at < 50 mSv/year, with average over 5 years < 20 mSv/year.

The difference is also due to the definitions; in the E.U.R. documents these values are set as targets, whereas in the ICRP and GSR Part 3 documents they are set as objectives.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
4. SAFETY ANALYSIS	
4.1 General	SSR 2/1 Req. 42 paras 5.71-5.74 SSR 2/1 Req. 10 GSR Part 4 Req. 15 para 4.53
4.2 Deterministic safety analysis	SSR 2/1 Req. 42 para 5.75 GSR Part 4 Req. 15 para 4.54
4.3 Probabilistic safety analysis	SSR 2/1 Req. 42 para 5.76 GSR Part 4 Req. 15 para 4.55

See the comments under (2) above.

In addition, the Rev 1 of GSR Part 4 has two new paragraphs added to the Requirement 10; para 4.36a and para 4.36b which specify the conditions for safety analysis in case of multiple units on one site in terms of external hazards and in case when resources (human and material) are being shared in accident conditions.

Such conditions are not found in the E.U.R. Section 2.1.4 on Safety Analysis. However E.U.R. Section 2.1.6.6 addresses sharing of SSCs between units.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
5. SAFETY CLASSIFICATION (5)	
5.1 Categorization of Safety Functions and Classification of SSCs	SSR 2/1 Req. 22
5.1.1 Introduction	SSR 2/1 Req. 18 para 5.23 SSR 2/1 Req. 11 SSG-30 para 2.3
5.1.2 Categories of safety functions	SSR 2/1 Req. 22 5.34 SSG-30 paras 2.6; 2.12
5.1.2.1 Safety Category 1 Functions	SSG-30 para 3.15
5.1.2.2 Safety Category 2 Functions	SSG-30 para 3.15
5.1.2.3 Safety Category 3 Functions	SSG-30 para 3.15
5.1.3 Design provisions	SSG-30 para3.23
5.1.4 Assignment of Safety Class to SSCs	SSG-30 paras 2.3-2.7; 3.17-3.26; Chapter 4
5.1.5 Requirements on SSCs according to Safety Class	SSR 2/1 Req. 18 SSR 2/1 Req. 9 SSG-30 paras 3.8, 3.9

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
5.1.6 Classification of SSCs according to the design and construction codes	SSG-30 paras 2.6, 2.9
5.1.7 Environmental conditions resistance levels	SSG-30 paras 3.23; 4.5

(5) 2.1.5 SAFETY CLASSIFICATION

The E.U.R. Section of Safety Classification is **in full compliance** with the IAEA Safety Requirements SSR 2/1 in particular with Requirement 4 on Fundamental safety functions, Requirement 18 on Engineering design rules and Requirement 22 on Safety classification. In addition, a full compliance is found also with the Specific Safety Guide SSG-30 on “Safety Classification of Structures, Systems and Components in Nuclear Power Plants”. The E.U.R. section on Safety Classification is more detailed then guidance provided in SSG-30 and it covers all aspects described in SSG-30. It is perhaps the best match between the E.U.R. document and the IAEA Safety Standards publications.

Under 2.1.5.1.5 compliance with the SSR 2/1 Requirement 9 was established in addition to Requirement 18 and SSG-30. However, paragraph 4.16 in the SSR 2/1 Requirement 9 gives guidance on what to do if an unproven design or feature is introduced. It states *“Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, safety shall be demonstrated by means of appropriate supporting research programmes, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications. The new design or feature or new practice shall also be adequately tested to the extent practicable before being brought into service, and shall be monitored in service to verify that the behaviour of the plant is as expected”.*

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
6. ENGINEERING DESIGN REQUIREMENTS	
6.1 General	SSR 2/1 Req. 6 para 4.5 SSR 2/1 Req. 14
6.2 Design limits	SSR 2/1 Req. 15
6.3 Resilience to failures	TITLE
6.3.1 Singe failure criteria	SSR 2/1 Req. 25
6.3.1.1 Redundancy	SSR 2/1 Req. 21 para 24
6.3.2 Common cause failures	SSR 2/1 Req. 24
6.3.2.1 Independence	SSR 2/1 Req. 21 para 24
6.3.2.2 Functional isolation	SSR 2/1 Req. 21 para 24
6.3.2.3 Diversity	SSR 2/1 Req. 24
6.4 Reliability of items important to safety	SSR 2/1 Req. 23
6.5 Fail safe design	SSR 2/1 Req. 26
6.6 Sharing of SSCs between units	SSR 2/1 Req. 33 para 5.63
6.7 Autonomy objectives (6)	TITLE

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
6. ENGINEERING DESIGN REQUIREMENTS	
6.7.1 Autonomy objectives in respect of operators and plant personnel	SSR 2/1 paras 4.11(d); 5.12; 5.13; 5.58; 6.33
6.7.2 Autonomy objectives in respect of ultimate heat sink	SSR 2/1 Req. 53
6.7.3 Autonomy objectives in respect of power supply system	SSR 2/1 Req. 68
6.7.4 Compressed air	SSR 2/1 Req. 72
6.7.5 Autonomy objectives in respect of MCR	SSR 2/1 Req. 65
6.8 Non-Permanent Equipment [(7)]	SSR 2/1 paras 6.28B, 6.45A; 6.68; TECDOC-1791 Ch.10

[(6)] 2.1.6.7 Autonomy objectives

This section of E.U.R. gives time limits in respect of operators and plant personnel actions, availability of ultimate heat sink and of power supply systems. All of these are also covered in IAEA SSR 2/1 but without concrete numbers in terms of time units (minutes, hours or days) as prescribed in the E.U.R. document. E.U.R. document is in this area **in full compliance** with the IAEA SSR 2/1 Requirements and provides additional guidance to designers in terms of prescribed time scales.

[(7)] 2.1.6.8 Non-permanent equipment

This section of E.U.R. is **in full compliance** with the IAEA SSR 2/1 requirements. SSR 2/1 addresses non-permanent equipment in 6.28B for the removal of heat from the containment, 6.45A for restoration of the electrical power supply and 6.68 to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation. All three uses of non-permanent equipment are covered in E.U.R. 2.1.6.8.5 C. More explanation on the use of non-permanent equipment is elaborated in the IAEA TECDOC-1791, Chapter 10, which in general provides more explanation on the application of SSG 2/1. Elaboration in this TECDOC is in line with the E.U.R. requirements provided under 2.1.6.8.1 – 2.1.6.8.7.

Both documents E.U.R. and IAEA recognize that all plant conditions must be taken care of by safety systems and safety features installed at the nuclear power plant. There should be no need for installation of any additional equipment in order to comply with the acceptance criteria. Non-permanent equipment is considered as complimentary essential means and provides additional robustness. Clear criteria must be established to determine, if non-permanent equipment should be installed on-site or stored off-site. These criteria consider coping time, installation time and flexibility among other parameters. These provisions are addressed both in the IAEA and the E.U. documents.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
7. DESIGN OF SPECIFIC SYSTEMS [(8)]	TITLE
7.1 Reactor core [(9)]	TITLE
7.1.1 Performance of fuel elements and assemblies	SSR 2/1 Req. 43
7.1.2 Structural capability of the reactor core	SSR 2/1 Req. 44

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
7. DESIGN OF SPECIFIC SYSTEMS (8)	TITLE
7.1.3 Control of reactor core	SSR 2/1 Req. 45
7.1.4 Reactor shutdown	SSR 2/1 Req. 46
7.2 Reactor coolant systems	TITLE
7.2.1 Design of reactor coolant systems	SSR 2/1 Req. 47
7.2.2 Overpressure protection of reactor coolant pressure boundary	SSR 2/1 Req. 48
7.2.3 Inventory of the reactor coolant	SSR 2/1 Req. 49
7.2.4 Clean-up of reactor coolant	SSR 2/1 Req.50 para 6.17
7.2.5 Removal of residual heat from the reactor core	SSR 2/1 Req. 51
7.2.6 Emergency cooling of the reactor core	SSR 2/1 Req. 52 paras 6.18; 6.19
7.2.7 Heat transfer to an ultimate heat sink	SSR 2/1 Req. 53 paras 6.19A; 6.19B
7.3 Containment	TITLE
7.3.1 Containment system for the reactor	SSR 2/1 Req. 54
7.3.2 Control of radioactivity release from the containment	SSR 2/1 Reqs. 55 and 56 TECDOC-1792 Ch. 7 and Appendix 4 Ch. 10 on practical elimination
7.3.3 Isolation of the containment	SSR 2/1 Req. 56
7.3.4 Control of containment conditions	SSR 2/1 Req. 58
7.4 Instrumentation and control systems	TITLE
7.4.1 Provision of instrumentation	SSR 2/1 Req. 59
7.4.2 Control systems	SSR 2/1 Req. 60
7.4.3 Protection systems	SSR 2/1 Req. 61
7.4.4 Reliability and testability of instrumentation and control systems	SSR 2/1 Req. 62
7.4.5 Separation of protection system and control system	SSR 2/1 Req. 64
7.4.6 Main Control Room (10)	SSR 2/1 Req. 65
7.4.7 Emergency Control Room	SSR 2/1 Req. 66
7.4.8 Emergency Response facilities on the site	SSR 2/1 Req. 67 GSR Part 7 Req. 24 para 6.25 (a),(b),(c)
7.5 Electrical power supply in AOO and accident conditions	SSR 2/1 Req. 68
7.6 Supporting systems and auxiliary systems	TITLE
7.6.1 Performance of supporting systems and auxiliary systems	SSR 2/1 Req. 69

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
7. DESIGN OF SPECIFIC SYSTEMS [(8)]	TITLE
7.6.2 Heat transport systems	SSR 2/1 Req. 70
7.6.3 Process sampling systems and post-accident sampling systems	SSR 2/1 Req. 71
7.6.4 Compressed air systems	SSR 2/1 Req. 72
7.6.5 Air-conditioning and ventilation systems	SSR 2/1 Req. 73
7.7 Fuel Storage and Handling Systems	SSR 2/1 Req. 80
7.8 Means of radiation monitoring	SSR 2/1 Req. 82
7.9 Treatment of radioactive effluents and radioactive waste	TITLE
7.9.1 System for treatment and control of waste	SSR 2/1 Req. 78 SSR 2/1 Req. 6 para 4.8 INSAG-12 para 4.5.8
7.9.2 System for treatment and control of effluents	SSR 2/1 Req. 79 SSR 2/1 Req. 6 para 4.8 INSAG-12 para 4.5.8

[(8)] 2.1.7 DESIGN OF SPECIFIC SYSTEMS

DESIGN OF SPECIFIC SYSTEMS as specified in 2.1.7 of E.U.R. is **in full compliance** with SSR 2/1 Chapter 6 DESIGN OF SPECIFIC PLANT SYSTEMS (Requirements 43 – 82). Requirements in both documents are **identical** except for SSR 2/1 Requirements 74-77 and 81 which are not addressed in E.U.R. section 2.1.7 but elsewhere. These requirements relate to:

- 74. Fire protection systems,
- 75. Lighting systems,
- 76. Overhead lifting equipment,
- 77. Steam supply system, feedwater system and turbine generators and
- 81. Design for radiation protection.

[(9)] 2.1.7.1 Reactor core

E.U.R. Section 2.1.7.1.1 on Performance of fuel elements and assemblies is **in full compliance** with the Requirements 43 and 44 of SSR 2/1 and recognizes the existence of "*the process of deterioration*". It however does not specify which these processes are as does SSR 2/1 in para 6.1 where it states:

The processes of deterioration to be considered shall include those arising from:

- Differential expansion and deformation;
- External pressure of the coolant;
- Additional internal pressure due to fission products and the build-up of helium in fuel elements;
- Irradiation of fuel and other materials in the fuel assembly;

- Variations in pressure and temperature resulting from variations in power demand;
- Chemical effects;
- Static and dynamic loading, including flow induced vibrations and mechanical vibrations;
- Variations in performance in relation to heat transfer that could result from distortion or chemical effects.

Allowance shall be made for uncertainties in data, in calculations and in manufacture.

However, the design limits, addressed in para 6.2 of SSR 2/1 are described in much greater detail in EUR Sections 2.1.7.1.1.1 for Normal operation, EUR Section 2.1.7.1.1.2 for Anticipated Operational Occurrences and in EUR Section 2.1.7.1.1.3 for Design Basis Accidents.

(10) 2.1.7.4.6 Main control room and 2.1.7.4.7 Emergency control room

E.U.R. requirements for the Main control room and for the Emergency control room are **identical** with the SSR 2/1 Requirements for Control room and Supplementary control room, only the names for both control rooms are different.

SSR 2/1 requirement under 6.41 that "*taking appropriate measures and providing adequate information for the protection of occupants against hazards also apply for the supplementary control room at the nuclear power plant",* is however not explicitly stated in the E.U.R. requirements for the Emergency control room.

E.U.R. section 2.1.7.1.3 Control of reactor core is **in compliance** with the SSR 2/1 requirement 45. However, requirements on reactivity control devices are not covered in E.U.R. section 2.1.7.1.3. The Requirement 45 stipulates that "*The demand made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized".* In addition SSR 2/1 in para 6.5 (that belongs to the same Requirement 45) stipulates that "*In the design of reactivity control devices, due account shall be taken of wear out and of the effects of irradiation, such as burnup, changes in physical properties and production of gas".*

The design of the reactivity control devices are however addressed in E.U.R. section 2.8.2.1.1.5 where it stipulates:

"The reactivity control systems shall be designed to ensure and sustain the sub-criticality in the reactor core:

- *after planned reactor shutdown during Normal Operation and after Anticipated Operational Occurrences, as long as needed; and*
- *after a transient period (if any) following a Design Basis Accident or Design Extension Conditions".*

Requirements on the reactivity control devices can be considered to be equivalent.

Additional Requirements from SSR 2/1 under Requirement 46 on Reactor Shutdown, namely:

- Requirements under para 6.9, 6.10 and 6.11 are covered by E.U.R. section 2.8.2.1.1.5, and
- Requirements under para 6.7, 6.8 and 6.12 are covered by E.U.R. section 2.8.2.1.1.6.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
8. OTHER CONSIDERATIONS	TITLE
8.1 Long term safety	TITLE
8.1.1 Inspection, on line monitoring, testing and maintenance	SSR 2/1 Req. 29

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
8.1.2 Qualification of items important to safety	SSR 2/1 Req. 30
8.1.3 Ageing management	SSR 2/1 Req. 31
8.2 Human factors	SSR 2/1 Req. 32
8.3 Security [11]	TITLE
8.3.1 General considerations	NSS-13; SSR 2/1 Req. 8 SSR 2/2 Req. 17
8.3.2 Design of physical protection sys.	NSS-13, SSR 2/1 Req. 38, Req. 39
8.3.3 Intentional aircraft crash	NSS-10

[11] 2.1.8.3 Security

E.U.R Section on Security is **in adequate compliance** with the recommendations given in the IAEA Nuclear Security Series, especially NSS-13 - "Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities".

However, SSR 2/1 Requirement 8 addresses Interfaces of safety with security and safeguards and states that *"Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another*". This important interface which calls for caution that all three important activities should not compromise each other is not addressed in the E.U.R. document. Therefore an adequate compliance has been assigned to this important issue.

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
9. TABLES	
9.1 Table 2 – Radiological criteria for radioactive releases in normal operation and AOO per unit	GSR Part 7; Req. 8, 5.25(c); Req. 9 paras 5.32(b), 5.40
9.2 Table 3 – Frequencies and general acceptance criteria for plant states [12]	SSG-2 Table 2 TECDOC-1791 Table 2, App.2, Table 5
9.3 List of hazards	SSR 2/1 Req. 17 SSG-3 Annex 1
9.3.1 List of internal and external hazards for standard design	SSR 2/1 Req. 17 SSG-3 Annex 1
9.3.2 List of external hazards for site specific evaluation	SSR 2/1 Req. 17 SSG-3 Annex 1
9.4 List of postulated initiating events for PWRs and BWRs	SSR 2/1 Req. 16 SSG-2 para 2.6 SRS 23 para 2.2

E.U.R. Chapter 2.1 SAFETY REQUIREMENTS	IAEA Safety and Security Standards
10. APPENDICES (13)	
A. SOURCE TERM AND RELEASE QUANTIFICATION METHODOLOGY FOR DESIGN EXTENSION	SSR 2/1 Req. 20 SSG-2 Chapter 9
B. VERIFICATION PROCESS OF THE EUR ENVIRONMENTAL IMPACT TARGETS	IAEA IEC Series

(12) Frequencies and general acceptance criteria for plant states as defined in E.U.R. Section 2.1.9.2 Table 3 are **in full compliance** with values defined in SSG-2 Table 2 and TECDOC-1791 Table 2, which should represent the IAEA approach. For Anticipated Operational Occurrences in the IAEA documents which correspond to Design Basis Condition 2 in the E.U.R. document, frequency is determined as $< 10^{-2}$. Design Basis Accidents for TECDOC-1791 have frequencies 10^{-2} - 10^{-6} events per year which corresponds to Design Basis Conditions 3 with $10^{-2} > f > 10^{-4}$ plus Design Basis Condition 4 with $10^{-4} > f > 10^{-6}$. E.U.R. document simply divides Design Basis Conditions into two states; Design Basis Condition 3 and Design Basis Condition 4 which both together correspond to Design Basis Accident condition defined in the IAEA documents.

The difference appears in the next level. Complex Sequences of Design Extension Conditions in E.U.R. have frequencies $10^{-4} > f > 10^{-7}$. They correspond to Design Extension Conditions without core melt in the IAEA documents and those have frequencies of $10^{-4} – 10^{-6}$ events per year. The next level however is the same; Severe Accidents of Design Extension Conditions in E.U.R have the same frequencies as corresponding Design Extension Conditions with core melt in the IAEA documents with the values of $f < 10^{-6}$.

The acceptance criteria for Design Extension Conditions are the same in both E.U.R. and IAEA SSR 2/1 Documents. For Design Extension Conditions without core melt (IAEA) or Design Extension Conditions with Complex Sequences (E.U.R.) the acceptance criteria are defined as:

- Limited damage of the fuel
- No consequential damage of the RCS
- Maintaining containment integrity

For Design Extension Conditions with core melt (IAEA) or Design Extension Conditions as Severe Accidents the only acceptance criteria is:

- Maintaining containment integrity

(13) E.U.R. Appendix 10 A on SOURCE TERM AND RELEASE QUANTIFICATION METHODOLOGY FOR DESIGN EXTENSION and E.U.R. Appendix 10 B on VERIFICATION PROCESS OF THE EUR ENVIRONMENTAL IMPACT TARGETS are far more detailed then the Requirements in the IAEA Safety Series. The common basis is the way the values given in Tables of Appendices A and B are calculated. The IAEA SSR 2/1 requires that values for Design Extension Conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments. It is further established that Deterministic assessments could be performed by means of a **best estimate approach.**

3.2 E.U.R. CHAPTER 2.2 PERFORMANCE REQUIREMENTS

E.U.R. Chapter 2.2 contains the design requirements that are not related primarily to safety but rather comprise the plant operational and transient response capabilities, plant type and size, core performance and fuel requirements and for Normal Operation, Anticipated Operational Occurrences and Design Basis Accidents. It also addresses spent fuel and radioactive waste disposal targets and occupational radiation exposure limits. At the end it provides the requirements on unit's reliability and availability.

E.U.R. Chapter 2.2 Performance Requirements are much more detailed then the related IAEA Safety Standards requirements and are **in full compliance** with associated IAEA Requirements from SSR 2/1, SSR 2/2, GSR Part 3 and GSR Part 5 as listed in the table below.

E.U.R Chapter 2.2 PERFORMANCE REQUIREMENTS	IAEA Safety Standards
1. TYPE AND PLANT SIZE	N/A
2. OPERATIONAL CAPABILITIES	SSR 2/2 Req. 8 para 4.31 SSR 2/1 Req. 68
3. CORE PERFORMANCES	SSR 2/1 Req. 45 SSR 2/2 Req. 30
4. FUEL REQUIREMENTS	SSR 2/1 Req. 43 Req. 44 SSR 2/2 Req. 30
5. SPENT FUEL AND RADIOACTIVE WASTE DISPOSAL TARGETS	SSR 2/1 Req. 12 SSR 2/1 Req. 78 SSR 2/2 Req. 21 GSR Part 5 Reqs. 4, 10, 11, 12
6. OCCUPATIONAL RADIATION EXPOSURE	GSR Part 3, SCHEDULE III-3
7. RELIABILITY AND AVAILABILITY OBJECTIVES	SSR 2/2 Req. 9

3.3 E.U.R. CHAPTER 2.3 GRID REQUIREMENTS

E.U.R Chapter 2.3 on Grid requirements was written with the objective to satisfy the needs of the electricity grid network and as such have little or almost no interaction with the IAEA safety standards requirements. It can however be said that the requirements set in this chapter assure that there is **a full compliance** with the Requirement 41 from SSR 2/1 on Interaction between the electrical power grid and the plant.

E.U.R Chapter 2.3 GRID REQUIREMENTS	IAEA Safety Standards
1. INTRODUCTION TO THE GRID REQUIREMENTS	N/A
2. GENERAL CHARACTERISTICS	SSR 2/1 Req. 41
3. OPERATIONS OF A UNIT UNDER NORMAL GRID CONDITIONS	SSR 2/2 Req. 31 8.11
4. OPERATIONS OF A UNIT UNDER DISTURBED GRID CONDITIONS	SSR 2/1 Req. 41

E.U.R Chapter 2.3 GRID REQUIREMENTS	IAEA Safety Standards
5. HOUSE LOAD OPERATION	N/A
6. ISLAND OPERATION	N/A
7. CONTRIBUTION OF A UNIT TO GRID RESTORATION	N/A
8. CONNECTION BETWEEN A UNIT AND THE GRID	SSR 2/1 Req. 41

3.4 E.U.R. CHAPTER 2.4 DESIGN BASIS

E.U.R Chapter 2.4 on Design Basis presents the general design criteria for all buildings, structures, systems and components at a nuclear power plant. External Hazards are divided into two categories; Design Basis External Hazards and Rare and Severe External Hazards. The IAEA Safety Requirements NS-R-3 Rev 1 from 2016 on Site Evaluation for Nuclear Installations does not make this distinction but rather divides them not on their severity but in accordance to their origin.

The IAEA Safety Requirements NS-R-3 Rev 1 is not entirely applicable to the E.U.R. design assessment process as NS-R-3 deals with the site evaluation and E.U.R. on design criteria related to a Standard Design. Only in a later phase of a nuclear project, when a utility requests the designer to perform a site specific evaluation of external hazards these requirements become applicable.

Plant Lifetime as defined in E.U.R. 2.4.2 and E.U.R. 2.4.3 on Design Philosophy are not defined in any IAEA Safety Requirements document and as such the comparison is marked as N/A.

Appendix A of E.U.R. Chapter 2.4 describes Method of seismic analysis. It prescribes that Soil-Structure Interaction analysis be performed by using either the finite elements methods or impedance methods. Such detailed requirements have not been found in standards used for comparison and therefore it is marked as N/A in the below tables.

Numerical values given in the E.U.R. document are much more detailed then in any IAEA Requirements, but in general it can be concluded that E.U.R. Chapter 2.4 on Design Basis is **in full compliance** with other reviewed standards.

E.U.R Chapter 2.4 DESIGN BASIS	IAEA Safety Standards
1. STANDARD SITE DESIGN CONDITIONS	TITLE
1.1 Introduction (14)	NS-R-3 Chptrs. 3&5 SSR 2/1 Req. 17 Paras 5.17 – 5.22
1.2.1 Seismic Hazards	NS-R-3 paras 3.1-3.7 NS-R-3 paras 3.24-3.28
1.2.2 External air temperature and humidity conditions	NS-R-3 paras 3.9-3.10
1.2.3 High winds, Tornados and generated missiles	NS-R-3 paras 3.12-3.14
1.2.4 Cooling water temperature	NS-R-3 para 3.53

E.U.R Chapter 2.4 DESIGN BASIS	IAEA Safety Standards
1.2.5 Precipitation and External Flooding	NS-R-3; paras 3.18-3.23
1.2.6 Lightning	NS-R-3 para 3.11
1.2.7 Drought	NS-R-3 paras 3.8-3.9
1.2.8 Hazards resulting in heat sink clogging	NS-R-3 para 3.54
1.2.9 Hazard values for Non Safety Classified SSCs	NS-R-3 paras 3.41-3.42
1.2.10 Table 1: Synthesis of the natural hazards values to be considered for a Standard plant	NS-R-3 Chptr. 3
1.3 Man-made External hazards	NS-R-3 para 3.51
1.3.1 Plant proximity hazards	NS-R-3 para 3.51
1.3.2 External explosion	NS-R-3 para 3.48-3.51
1.3.3 Aircraft crash	NS-R-3 para 3.44-3.47 NSS-10
1.4 Sabotage	NSS-13
2. PLANT DESIGN LIFE	N/A
3. DESIGN BASIS PHILOSOPHY	N/A
4. COMPONONTS AND SYSTEMS CRITICAL TO PLANT PERFORMANCE	SSG-30
5. DESIGN LOADS AND CONDITIONS	SSG-30 para 4.3(c)
6. SEISMIC DESIGN	SSG-30 para 4.5 NS-R-3 paras 3.1-3.4 NS-R-3 paras 3.38 – 3.43
7. DESIGN OF PIPEWORK SYSTEMS	SSR 2/1 Req. 47 para 6.13
8. EQUIPMENT QUALIFICATION	SSR 2/2 Req. 13 paras 4.48-4.49
A. METHOD OF SEISMIC ANALYSIS	N/A

(14) In the area of External Hazards the E.U.R. distinguish between Design Basis External Hazards and Rare and Severe External Hazards. The IAEA Safety Requirements NS-R-3 Rev 1 from 2016 on Site Evaluation for Nuclear Installations does not make this distinction but rather divides the external hazards as:

- Earthquakes and surface faulting
- Meteorological events
- Flooding
- Geotechnical hazards
- External human induced events and
- Other important considerations

An important addition in the NS-R-3 Rev 1 is paragraph 5.1A which specifies the requirement to periodically review site specific hazards when necessary but at least every 10 years. This requirement is implicitly taken into consideration in the E.U.R. Chapter dealing with the Periodic Safety Review.

The IAEA Safety Requirements NS-R-3 Rev 1 is not entirely applicable to the E.U.R. design assessment process as NS-R-3 deals with the site evaluation and E.U.R. on design criteria related to a Standard Design. Only in a later phase of a nuclear project, when a utility requests the designer to perform a site specific evaluation of external hazards these requirements become applicable.

3.5 E.U.R. CHAPTER 2.5 CODES AND STANDARDS

The E.U.R. Chapter 2.5 on Codes and Standards describes which laws, regulations, codes and standards are to be used for the design of future Light Water Reactors and how they are to be applied.

In EUR 2.5.4.1 Table 1 examples of rules for Safety Classes 1 and 2 for:

- Civil structures
- Mechanical equipment
- Electrical equipment

are presented and the IAEA Safety Guides GS-G are identified.

In EUR 2.5.5.1 Table 2 examples of rules for Safety Class 3 for:

- Structures and mechanical equipment
- Electrical equipment

are presented and the IAEA Safety Guides GS-G are identified.

In EUR 2.5.6.1 Table 3 examples of rules for Non-Safety Class items for:

- Structures
- Mechanical equipment
- Electrical equipment

are presented and the IAEA Safety Guides GS-G are identified.

Levels of rules as presented in this chapter (and in Figure 1 within this chapter i.e. EUR 2.5.7.Fig. 1) are **in full compliance** with the relevant parts of IAEA GSR Part 1 on Governmental, Legal and Regulatory Framework for Safety as well as WENRA requirements and EU directives.

3.6 E.U.R. CHAPTER 2.6 MATERIAL-RELATED REQUIREMENTS

This Chapter of the E.U.R. provides the general requirements for materials to be used in nuclear Structures, Systems and Components of a new design.

In the SSR 2/1 the first layer of defence-in-depth requires: *…… To meet these objectives, careful attention is paid to the selection of appropriate design codes and materials, and to the quality control of the manufacture of components and construction of the plant, as well as to its commissioning…….*

In SSR 2/1 Requirement 12 on Features to facilitate radioactive waste management and decommissioning, it states in 4.20 (a):

The choice of materials, so that amounts of radioactive waste will be minimized to the extent practicable and decontamination will be facilitated.

Further, SSR 2/1 Requirement 47: Design of reactor coolant systems stipulates:

The components of the reactor coolant systems for the nuclear power plant shall be designed and constructed so that the risk of faults due to inadequate quality of materials, inadequate design standards, insufficient capability for inspection or inadequate quality of manufacture is minimized.

SSR 2/1 Requirement 74 on Fire protection stipulates in para 6.54:

Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the plant, in particular in locations such as the containment and the control room.

SSR 2/1 Requirement 81 on Design for radiation protection stipulates in para 6.70:

Materials used in the manufacture of structures, systems and components shall be selected to minimize activation of the material as far as is reasonably practicable.

IAEA SSR 2/2 requires in its Requirement 31 paras 8.15 – 8.17:

8.15. The operating organization shall establish suitable arrangements to procure, receive, control, store and issue materials (including supplies), spare parts and components.

8.16. The operating organization shall be responsible for using these arrangements for the procurement of materials (including supplies), spare parts and components and for ensuring that their characteristics are consistent with applicable safety standards and with the plant design.

8.17. The operating organization shall ensure that storage conditions are adequate and that materials (including supplies), spare parts and components are available and are in proper condition for use.

Requirements set out in Chapter 2.6 of E.U.R. are much more detailed then Requirements in the IAEA Safety Standards and are **in full compliance** with Requirements stipulated in SSR 2/1 and SSR 2/2.

3.7 E.U.R. CHAPTER 2.7 FUNCTIONAL REQUIREMENTS: COMPONENTS

The functional requirements for individual components described in this chapter are of general nature and do not consider possible application i.e. introduction of these components into specific systems at nuclear power plants.

Functional requirements in E.U.R. Chapter 2.7 deal with: INTRODUCTION, REACTOR COOLANT SYSTEM COMPONENTS, VALVES, VALVE ACTUATORS, PUMPS, HEAT EXCHANGERS, TANKS, BOLTED JOINTS and THREADED FASTENERS, PIPEWORK AND FITTINGS, FILTERS AND ION EXCHANGERS, FUEL, ELECTRICAL EQUIPMENT, DIESEL GENERATORS AND OTHER POWER SUPPLIES, HVAC EQUIPMENT, HANDLING EQUIPMENT IN THE NUCLEAR ISLAND, HANDLING EQUIPMENT IN THE POWER GENERATION PLANT.

The IAEA Safety Standards do not address such level of detail in Safety Requirements and not even in their Safety Guides. For example in the IAEA Safety Guide NS-G-1.9 on the Design of Reactor Coolant System and Associated Systems in Nuclear Power Plants it is clearly stated in para 1.6 that the scope of the Safety Guide “*does not extend to the detailed design of specific components, for example pumps or heat exchangers”.*

E.U.R. are much more detailed in functional requirements for components to be used in the design of nuclear power plants and are **in full compliance** with other reviewed safety requirements, including the IAEA Safety Guides.

Functional Requirements on the entire systems and processes is addressed in the next E.U.R. Chapter, Chapter 2.8.

3.8 E.U.R. CHAPTER 2.8 FUNCTIONAL REQUIREMENTS: SYSTEMS AND PROCESSES

This chapter deals with the functional requirements for the main systems and processes in Light Water Reactors designs. They relate to the functional requirements for design basis as well as to design extension conditions.

Most of these requirements have already been addressed in the E.U.R. Chapter 2.1, where a detailed correlation to SSR 2/1 has been made. In this Chapter some functional requirements for systems have been laid down in more detail but in essence, the Requirements for the SSR 2/1 have been met in all cases. Some additional guidance as found in SSG or NS-G documents have also been consulted and it is concluded that all Requirements in Chapter 2.8 are **in full compliance** with other benchmarked standards.

E.U.R Chapter 2.8 FUNCTIONAL REQUIREMENTS: SYSTEMS AND PROCESSES	IAEA Safety Standards
1. INTRODUCTION	Fulfilment of Requirements as in E.U.R. Ch. 2.1 and Ch. 2.2
2. REQUIREMENTS ON FUNCTIONS	SSR 2/1 - Req. 4 paras 4.1 and 4.2
3. SYSTEM FUNCTIONAL REQUIREMENTS	TITLE
3.1 Introduction	SSR 2/1 Req. 14
3.2 Core and fuel systems	SSR 2/1 Reqs. 43-46 NS-G-1.12
3.3 Reactor coolant system	SSR 2/1 Reqs. 47-48 NS-G-1.9
3.4 Engineered safety features	SSR 2/1 Reqs. 51-53
3.5 Reactor auxiliary systems	SSR 2/1 Reqs. 69-76
3.6 Fuel Storage and Handling Systems	SSR 2/1 Req. 80 NS-G-1.4
3.7 Electrical power supplies	SSR 2/1 Req. 68 SSG-34
3.8 Radioactive waste processing systems	SSR 2/1 Reqs. 78-79

E.U.R Chapter 2.8 FUNCTIONAL REQUIREMENTS: SYSTEMS AND PROCESSES	IAEA Safety Standards
3.9 Heating, ventilation and air conditioning systems (HVAC)	SSR 2/1 Reqs. 70-73
3.10 Plant cooling water systems	SSR 2/1 Req. 77
3.11 Process radiation monitoring system	SSR 2/1 Reqs. 5, 81 and 82
3.12 Process instrumentation and control function	SSR 2/1 Reqs. 59-64 SSG-39
3.13 Protection System	SSR 2/1 Req. 61 SSG-39
3.14 Non-process instrumentation and control functions	SSR 2/1 para2.2 Reqs. 55,81,82
4. SYSTEMS ASSOCIATION REQUIREMENTS	SSR 2/1 Reqs. 14,21,24,27,52, 58,62,64 Paras 5.8(2), 5.40

3.9 E.U.R. CHAPTER 2.9 CONTAINMENT SYSTEM

This chapter deals with the importance of the containment system as the last barrier of the defence-in-depth strategy for fission products both for design basis accidents and for design extension conditions. The containment system requirements in this chapter are set in such a way as to fulfil all the design requirements from Chapter 2.1 on Safety requirements and Chapter 2.4 on Design basis. Full compliance of both Chapters with the reviewed requirements of the IAEA, WENRA and EU has already been established in the benchmarking analysis of those chapters and as a consequence it can be summarized that Chapter 2.9 is also **in full compliance** with the benchmarked requirements.

E.U.R Chapter 2.9 CONTAINMENT SYSEM	IAEA Safety Standards
1. CONTAINMENT SYSTEM SAFETY FUNCTIONS	SSR 2/1 Reqs.54-58
1.1 Introduction	SSR 2/1 Req. 54
1.2 Figure 1 – Containment system functions	SSR 2/1 Req. 54
1.3 Fission product confinement and control	SSR 2/1 Req. 54 NS-G-1.10 paras 2.3-2.14
1.4 Protection against external events	SSR 2/1 Req. 54 NS-G-1.10 para 2.15 paras 3.3-3.7

E.U.R Chapter 2.9 CONTAINMENT SYSEM	IAEA Safety Standards
1.5 Biological shielding	SSR 2/1 Req. 54 NS-G-1.10 para 2.16
2. CONTAINMENT SYSTEM PERFORMANCE	TITLE
2.1 Containment system general arrangements	NS-G-1.10 paras 4.6, 4.147 Annex I; I-2
2.2 Fission product confinement and control	SSR 2/1 Req. 55 NS-G-1.10 paras 2.3-2.14 paras 5.15-5.31
2.3 Protection against external hazards	SSR 2/1 Req. 54 NS-G-1.10 paras 3.3-3.7
2.4 Biological shielding	SSR 2/1 Req. 54 NS-G-1.10 para 2.16
2.5 Containment system accessibility(9.1)	SSR 2/1 Req. 57 NS-G-1.10 paras 4.25,4.26
3. STRUCTURAL DESIGN AND VERIFICATIONS	TITLE
3.1 Primary containment (9.2)	SSR 2/1 Req. 20 para 5.30 NS-G-1.10 paras 4.57, 4.120, 4.151, Appendix A8, Annex I-2, I-5
3.2 Secondary containment	NS-G-1.10 paras 4.33, 4.34 4.80,4.120,4.147,4.149,5.25, Annex I-20
4. EQUIPMENT AND SYSTEM DESIGN REQUIREMENTS	TITLE
4.1 Primary containment equipment and systems	NS-G-1.10 paras 4.235-4.238
4.2 Secondary containment equipment and systems	NS-G-1.10 paras 4.235-4.238

(9.1) SSR 2/1 para 6.26 requires that "*Containment openings for the movement of equipment or material through the containment shall be designed to be closed quickly and reliably in the event that isolation of the containment is required".* Wording "quickly and reliably" is not found in the E.U.R. Chapter 2.9 but section 2.9.4.1.7. (D1) specifies desired times for the containment isolation and reads "*Closure times for Primary containment isolation valves could generally be approximately 2 seconds per cm of nominal valve diameter but not less than 15 seconds and no more than 60 seconds unless specifically justified".* It is considered that this requirement is **in full compliance** with para 6.26 of SSR 2/1.

(9.2) SSR 2/1 para 5.30 addressing the design extension conditions stipulates: *"In particular, the containment and its safety features shall be able to withstand extreme scenarios that include, among other things, melting of the reactor core. These scenarios shall be selected by using engineering judgement and input from probabilistic safety assessments".* The equivalent E.U.R. requirement 2.9.3.1.8.1C reads: "*Primary containment design shall be such that the primary containment can withstand any of the severe accident addressed in DEC, without operator action during the first 12 hours from the beginning of Severe Accident conditions".* It is considered that both requirements are **in full compliance** with each other, whereas E.U.R. specifies in addition the conditions for possible operator actions.

3.10 E.U.R. CHAPTER 2.10 INSTRUMENTATION & CONTROL AND HUMAN-MACHINE INTERFACE

The E.U.R. Chapter 2.10 addresses Instrumentation and Control (I&C) Systems and Human-Machine Interface. It addresses the overall I&C architectural requirements as well as the individual I&C system requirements. This Chapter of E.U.R. is primarily based on the IEC 61513 requirements which were not within the scope of this benchmarking exercise. However, several requirements from SSR 2/1 are also applicable and have been taken into account as well as the SSG-39 Safety Guide on Design of Instrumentation and Control Systems for Nuclear Power Plants.

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
1 General introduction on Instrumentation and Control and HMI	TITLE HMI in SSG-39, paras 8.47-8.67.
1.1 Purpose of Instrumentation and Control	SSG-39, paras 3.7-3.16.
1.2 Purpose of Life Cycle	SSG-39, paras 2.10-2.23.
2 Scope	IEC 61513 (1)
3 Normative references	List of IEC standards
4 Terms and definitions	EUR Volume 1 Clause 3 of IEC 61513
5 Symbols and abbreviations	EUR Volume 1
6 Overall I&C life cycle	SSG-39, pars 2.10-2.23.

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
6.1 General	IEC 61 513
6.2 Deriving the I&C requirements from the plant design base	SSG-39, paras 3.7-3.16.
6.2.1 General	SSR-39, paras 1.18 – 1.27
6.2.2 Review of the functional, performance and independence requirements	SSG-39, paras 4.14 – 4.24
6.2.2.1 Defence in depth	INSAG-12, para 3.2.1.
6.2.2.2 Functional and performance requirements	SSG-39, paras 2.1-2.9.
6.2.2.3 Level of automation	SSG-39, paras 8.68- 8.75.
6.2.2.4 Task analysis	SSG-39, paras 8.76 – 8.86
6.2.2.5 Variables to be displayed	SSG-39, para 3.15
6.2.2.6 Priority principle between automatic and manual actions	SSG-39, paras 3.13-3.14
6.2.3 Review of the safety categorisation requirements	SSG-39, para 3.13 IEC 61226
6.2.4 Review of plant constrains	As per EUR Chapters 2.1, 2.4 and 2.8
6.3 Output documentation of I&C requirements specification phase	SSG-39, para 2.90.
6.4 Design of the overall I&C architecture and assignment of the I&C functions	TITLE
6.4.1 General	SSG-39, paras 3.1- 3.6.
6.4.2 Design of the I&C architecture	TITLE
6.4.2.1 General	SSG-39, paras 4.1- 4.10.
6.4.2.2 General requirements	SSG-39, paras 6.1-6.5
6.4.2.2.1 Independence	SSG-39, paras 4.14- 4.24.
6.4.2.2.2 Single Failure Criterion	SSG-39, paras 6.10- 6.19.
6.4.2.2.3 Classification	SSG-39, paras 5.1- 5.13.
6.4.2.2.4 Interfaces and Priority Management	SSG-39, paras 4.13-4.24
6.4.2.3 Human machine interfaces	TITLE
6.4.2.3.1 General	SSG-39, Chapter 8
6.4.2.3.2 Main Control Room	SSG-39, paras 8.1- 8.12. SSR 2/1, paras 6.39 – 6.40A GSR Part 7, para 6.25
6.4.2.3.3 Emergency Control Room (10.2)	SSG-39, paras 8.13- 8.18. SSR 2/1, para 6.41 GSR Part 7, para 6.25

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
6.4.2.3.4 Emergency Centre	SSR 2/1, para 6.42 GSR Part 7, para 6.25 INSAG-12, para 4.8.2.
6.4.2.3.5 Technical Support Centre	SSR 2/1, para 6.42 footnote (23) GSR Part 7, para 6.25 SSG-39, paras 8.36- 8.46. SSG-39, para 7.123
6.4.2.3.6 I&C Service Centre	SSG-39, paras 4.11- 4.12
6.4.2.3.7 Maintenance management room	SSG-39, paras 4.11- 4.12
6.4.2.3.8 Oral communication	SSG-39, paras 8.36 – 8.46
6.4.2.3.9 HMI design general targets	SSG-39, paras 8.47 – 8.67
6.4.2.3.10 Design of alarms	SSG-39, para 7.87
6.4.2.3.11 Operating instructions	SSG-39, para 2.90 INSAG-12, paras 4.5.6 - 4.5.7
6.4.2.3.12 Recording, archiving and reports	SSG-39, para 8.94
6.4.2.3.13 Operating aid	SSG-39, paras 8.76 – 8.86
6.4.2.3.14 Diagnostics	SSG-39, para 2.90
6.4.2.3.15 Technology of HMI	SSG-39, paras 8.47 – 8.67
6.4.2.3.16 Human factor techniques and HMI ergonomics	SSG-39, paras 8.47- 8.67.
6.4.2.4. Data communication	SSG-39, paras 7.79 – 7.100
6.4.2.5 Tools	TITLE
6.4.2.5.1 General requirements	SSG-39, paras 7.148 – 7.164
6.4.2.5.2 Software Tools	SSG-39, paras 7.148 – 7.164
6.4.2.6 Defence against CCF	SSG-39, paras 4.25- 4.35.
6.4.3 Assignment of functions to I&C systems	SSG-39, paras 9.6 – 9.15
6.4.4 Required analysis	TITLE
6.4.4.1 General	SSG-39, paras 9.64 – 9.95
6.4.4.2 Assessment of reliability and defences against CCF	SSG-39, paras 4.25- 4.35.
6.5 Overall planning	TITLE
6.5.1 General	SSG-39, paras 2.24- 2.28.
6.5.2 Overall quality assurance programs	SSG-39, paras 2.1- 2.9.
6.5.3 Overall security plan	GS-G-3.5, 5.182.-5.183.
6.5.4 Overall I&C integration and commissioning	SSG-39, paras 2.124- 2.127.
6.5.5 Overall operation plan	SSG-39, paras 2.152- 2.156.

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
6.5.6 Overall maintenance plan	SSG-39, paras 2.152- 2.156.
6.5.7 Planning of training	SSG-39, paras 2.24- 2.28.
6.6 Output documentation of overall planning	TITLE
6.6.1 General	SSG-39, paras 2.88 – 2.91
6.6.2 Architectural design documentation	SSG-39, paras 2.88-2.91.
6.6.3 Functional assignment documentation	SSG-39, paras 2.88 – 2.91
7 System life cycle (10.3)	TITLE
7.1 General	SSG-39, paras 2.10-2.37. IEC 61513
7.2 Requirements	TITLE
7.2.1 General	SSG-39, paras 2.92 – 2.107
7.2.2 System requirements specification	TITLE
7.2.2.1 General	SSG-39, paras 2.92 – 2.107
7.2.2.2 Functions	IEC 61513
7.2.2.3 Design constraints	TITLE
7.2.2.3.1 General	SSG-39, paras 4.1 – 4.2 SSG-39, paras 9.12 – 9.61
7.2.2.3.2 System architecture	SSG-39, Chapter 4
7.2.2.3.2.1 Implementation	SSG-39, paras 2.118 – 2.123
7.2.2.3.2.2 Redundancy	SSG-39, paras 6.20 – 6.21
7.2.2.3.2.3 Independence Physical Separation	SSG-39, paras 6.30 – 6.37
7.2.2.3.2.4 Single Failure Criterion	SSG-39, paras 6.10- 6.19.
7.2.2.3.2.5 Power supply and grounding	SSG-39, paras 7.60- 7.65.
7.2.2.3.3 Internal behaviour of the system	SSG-39, para 7.143
7.2.2.3.4 Self-supervision and tolerance to failures	SSG-39, paras 2.134 and 6.9
7.2.2.3.5 Testability	SSG-39, paras 6.159- 6.191.
7.2.2.3.6 Maintainability	SSG-39, paras 6.192-6.197
7.2.2.4 Boundaries and interfaces with other systems and tools	SSG-39, para 9.76
7.2.2.5 Interfaces with users	SSG-39, para 7.68
7.2.2.6 Environmental conditions	SSG-39, para 6.96 – 6.99
7.2.2.7 Qualification	SSG-39, 6.77 – 6.134
7.2.3 System specification	TITLE
7.2.3.1 General	EUR Section 2.10.7.3
7.2.3.2 Selection of pre-existing components	SSG-39, paras 2.118 – 2.123

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
7.2.3.3 System architecture	SSG-39, 4.1 – 4.14
7.2.3.4 Software specification	IEC 60880 IEC 62138
7.2.3.5 Assignment of the application functions in the system	IEC 61513
7.2.4 System detailed design and implementation	TITLE
7.2.4.1 General	SSG-39, 7.11-7.18.
7.2.4.2 Required analysis	TITLE
7.2.4.2.1 Functional validation of the application functions requirements specification	SSG-39, paras 2.92 – 2.107
7.2.4.2.2 Reliability assessment	SSG-39, paras 6.6 – 6.9
7.2.5 System integration	SSG-39, 2.124-2.127.
7.2.6 System validation	SSG-39, 2.128- 2.142.
7.2.7 System installation	SSG-39, 2.143-2.151.
7.2.8 System design modification	SSG-39, 2.157- 2.167.
7.3 System planning	TITLE
7.3.1 General	SSG-39, paras 2.24 – 2.28
7.3.2 System quality assurance plan	TITLE
7.3.2.1 General	SSG-39, paras 2.1-2.9
7.3.2.2 System verification plan	SSG-39, paras 2.66-2.74
7.3.2.3 System configuration management plan	SSG-39, paras 2.38-2.55.
7.3.2.4 Fault resolution procedures	SSG-39, paras 2.96; 2.134; 2.172; 9.35; 9.46
7.3.3 System security plan	GS-G-3.5, paras 5.182.-5.183 GSR Part 7, para 6.17
7.3.4 System integration plan	SSG-39, paras 2.143- 2.151.
7.3.5 System validation plan	SSG-39, paras 2.128- 2.142.
7.3.6 System installation plan	SSG-39, paras 2.143- 2.151.
7.3.7 System operation plan	SSG-39, paras 2.140- 2.142.
7.3.8 System maintenance plan	SSG-39, paras 6.192- 6.192.
7.4 Output documentation of system planning	TITLE
7.4.1 General	SSG-39, paras 2.88 – 2.91
7.5 System qualification	TITLE
7.5.1 General	SSG-39, paras 6.77- 6.90.
7.5.2 Generic and application-specific qualification	SSG-39, paras 2.108 – 2.117
7.5.3 Qualification plan	TITLE

E.U.R Chapter 2.10 INSTRUMENTAION & CONTROL AND HUMAN-MACHINE INTERFACE	IAEA Safety Standards
7.5.3.1 General	SSG-39, paras 6.77- 6.95.
7.5.3.2 Functional and environmental qualification	TITLE
7.5.3.2.1 General	SSG-39, paras 6.96- 6.107.
7.5.3.2.1.1 Suitability and correctness	SSG-39, paras 6.91- 6.95.
7.5.3.2.1.2 Environmental qualification	SSG-39, paras 6.96- 6.107.
7.5.3.2.1.3 Qualification for the effects of internal and external hazards	SSG-39, paras 6.108- 6.112.
7.5.3.2.1.4 Electromagnetic qualification	SSG-39, paras 6.113- 6.134.
7.5.3.2.2 Equipment Qualification methods	SSG-39, paras 6.77–6.134
7.5.3.2.2.2 Additional qualification of interconnected systems	SSG-39, paras 4.1 – 4.10
7.5.3.2.2.3 Analysis	NS-G-1.6, paras 4.1 – 4.10
7.5.3.2.2.4 Reliability demonstration	SSG-39, para 6.82
7.5.3.2.2.5 Operating experience	SSG-39, para 6.82
7.5.3.3 Software evaluation and assessment	SSG-39, paras 9.64-9.95
7.5.4 Additional qualification of interconnected systems	SSG-39, paras 4.3
7.5.5 Maintaining qualification	SSG-39, paras 6.78 – 6.90
7.5.6 Documentation	SSG-39, paras 2.88- 2.91.

(10.1) In this column IAEA Safety Standards are listed, mostly SSG-39 which is the most relevant IAEA Safety Guide. However on few places the relevant IEC Standards are also cited for completeness reasons as they also appear in the E.U.R. Chapter 10 document.

(10.2) Emergency Control Room is named Supplementary Control Room in the IAEA Safety Standards (apart from SSG-39 also defined in SSG2/1 on Design and GSR Part 7 on Emergency Preparedness and Response) but the functions of both are the same.

(10.3) Requirement 63 of SSR-2/1 (Rev. 1) states:

"If a system important to safety at the nuclear power plant is dependent upon computer based equipment, appropriate standards and practices for the development and testing of computer hardware and software shall be established and implemented throughout the service life of the system, and in particular throughout the software development cycle. The entire development shall be subject to a quality management system."

E.U.R. Sections 2.10.6.5 on Overall planning, Sections 2.10.7.3 on System planning, 2.10.7.4 on Output documentation of system planning and 2.10.7.5 on System qualification address the same aspects as those which are normally found in the applicable Management System but not in the same integrated manner.

Nevertheless all requirements from SSR-2/1 Req. 63, para 6.37 (a) – (f) are being covered by the E.U.R. sections 2.10.6.5 and 2.10.6.6 as well as 2.10.7.3 – 2.10.7.5 and therefore it can be concluded that **full compliance** has been established also in this subject area.

3.11 E.U.R. CHAPTER 2.11 LAYOUT

This chapter specifies the requirements to be taken into account when establishing the layout of the overall site, layout of the individual buildings and the routing of services within those buildings. It specifies criteria to be used when developing layouts.

All stages of plant life are covered, from construction, through operation to replacement.

This chapter covers:

- Guidelines to enhance safety taking into account the potential hazards, including redundancy, segregation and fire barriers,
- Definition of environmental zones for radiation and contamination control and personnel access and escape routes,
- Access around the plant and equipment for installation, operation and maintenance,
- Separation for the routing of services.

This Chapter also takes discusses security, laboratory requirements and lifting equipment

E.U.R Chapter 2.11 LAYOUT	IAEA Safety Standards
1. GENERALARRANGEMENTS AND SITE ASPECTS	TITLE
1.1 Overall site and building layout considerations	SSR 2/1 Reqs. 33, 40, SSR 2/2 para 3.2(d) SSR 2/1 Reqs. 21, 24
1.2 Access	SSR 2/1 Req. 36
2. SAFETY AND HAZARD ASPECTS	TITLE
2.1 Introduction	NS-R-3
2.2 Physical separation	SSR 2/1 Reqs. 21 and 24
2.3 Penetrations through barriers	SSR 2/1 Req. 65
2.4 Measures against hazards	SSR 2/1 Reqs. 17 and 74
2.5 Environmental zones	SSR 2/1 Req. 16 para 5.15 SSR 2/1 Req. 20 para 5.29(b) SSR 2/1 Req. 30 paras 5.48, 5.49, 5.50 NSS-13
3. EQUIPMENT ARRANGEMENT RULES	N/A
4. ROUTING OF SERVICES	N/A
5. SITE SUPPORT SYSTEMS	SSR 2/1 Reqs. 69-76
6. SECURITY	NSS-13
7. DECONTAMINATION FACILITIES	SSR 2/1 Req. 80 para 6.67(h)

E.U.R Chapter 2.11 LAYOUT	IAEA Safety Standards
	SSR 2/1 Req. 81 para 6.76
8. LABORATORIES	SSR 2/2 Req. 29 para 7.16
9. LIFTING EQUIPMENT	SSR 2/1 Req. 76

Chapter 2.11 of E.U.R. on the Layout contain much more details then other compared requirements but general principles could be matched with other sources as indicated in the above table. It can be therefore concluded that E.U.R. Chapter 2.11 is **in full compliance** with compared other international standards.

3.12 E.U.R. CHAPTER 2.12 DESIGH PROCESSES AND DOCUMENTATION

This Chapter provides the requirements for the **process** to be carried out in designing nuclear power plant. It includes all activities from initial development of the design to handing over the plant to the owner.

The design must be fully documented and must follow the general principles of the project management as stipulated in the IAE GSR Part 2 “Leadership and Management for Safety” and therefore in addition to other design international standards (SSR 2/1 and SSR 2/2), this IAEA Safety Requirements document was mainly used as basis for benchmarking this Chapter of the E.U.R.s. It is important to confirm that the design process is being carried out as a single integrated process which includes sub-processes for different technical disciplines.

E.U.R Chapter 12 DESIGN PROCESSES AND DOCUMENTATION	IAEA Safety Standards
1. INTRODUCTION	TITLE
1.1.1 Human factors	SSR 2/1 Req. 32
1.1.2 Balance in design	GSR Part 2 Req. 4 INSAG-4
1.1.3 Experience feedback	SSR 2/1 Req. 3 para 3.6(b)
1.1.4 Design for constructability, operability and maintainability	SSR 2/1 Req. 3 SSR 2/1 Req. 29
2. DESIGN DEVELOPMENT PLAN	TITLE
2.1 Design manual	SSR 2/1 Req. 2
2.2 Human – Machine Interface (HMI) design process	SSR 2/1 Req. 32 GSR Part 2 Req. 9

E.U.R Chapter 12 DESIGN PROCESSES AND DOCUMENTATION	IAEA Safety Standards
2.3 Involvement of construction and operation personnel	SSR 2/1 Req. 11 GSR Part 2 Req. 5
2.4 Schedule	GSR Part 2 Req. 10
3. TECHNOLOGY BASE	SSR 2/1 Req. 3 para 3.6(b) SSR 2/1 Req. 9 SSR 2/1 Req. 18
4. DESIGN LIFE OF EQUIPMENT	SSR 2/1 Req. 6 SSR 2/1 Req. 29 SSR 2/1 Req. 30 SSR 2/1 Req. 31
5. PLANT SIMPLIFICATION	SSR 2/2 Req. 4 para 3.12 SSR 2/2 Req. 8 para 4.28 SSR 2/2 Req. 11 para 4.41 SSR 2/2 Req. 26 para 7.6 SSR 2/2 Req. 27 para 7.9
6. EQUIPMENT STANDARDIZATION	N/A
7. PROJECT DATA CONSISTENCY AND DOCUMENTATION INFORMATION	GSR Part 2 Req. 8 GSRPart2 Req. 10 SSR 2/1 Req. 2 SSR 2/1 Req. 3 SSR 2/1 Req. 14 para 5.3

E.U.R Chapter 12 DESIGN PROCESSES AND DOCUMENTATION	IAEA Safety Standards
8. SPECIFIC DESIGN PROCESS REQUIREMENTS	GSR Part 2 Req. 1 para 2.2(b) SSR 2/1 Req. 2 para 3.3 SSR 2/1 Req. 9 para 4.15 SSR 2/1 Req. 15 para 5.4 SSR 2/1 Req. 18 SSR 2/1 Req. 63
9. QUALITY ASSURANCE PROGRAMME AND CONFIGURATION MANAGEMENT PROGRAMME	SSR 2/2 Req. 9 para 4.36 SSR 2/2 Req. 10 SSR 2/1 Req. 3 para 3.6(d)
10. DESIGN INTEGRATION	GSR Part 2 Req. 6
11. INTERDISCIPLINARY DESIGN REVIEWS	GSRPart2 Req. 10 para 4.31
12. INFORMATION MANAGEMENT SYSTEM	N/A
13. DESIGN-CONSTRUCTION INTEGRATION	See EUR Chapter 2.13
14. TABLE 1	SSR 2/1 para 2.17 INSAG-19

E.U.R. Chapter 2.12 is in **full compliance** with the benchmarked international standards.

3.13 E.U.R. CHAPTER 2.13 CONSTRUCTABILITY AND COMMISSIONING

E.U.R Chapter 2.13 is addressing the requirements set at the design stage for easier and better constructability and commissioning of new nuclear power plants. It addresses also the modern construction techniques such as modularization and pre-fabrication which are however not included in any IAEA requirements on design and construction. Constructability takes also into account site preparation, hence IAEA Safety Requirements NS-R-3 has also been considered.

The general constructability requirements for a LWR plant are focused on how to build and commission the plant, with the main objectives that concern quality, time period and minimization of the total capital investment costs. As the international safety standards do not specify time period or cost concerns, this part of the E.U.R. Chapter 2.13 was found to be irrelevant. Quality concerns are well documented in the E.U.R. Chapter 2.15.

E.U.R Chapter 13: CONSTRUCTABILITY AND COMMISSIONING	IAEA Safety Standards
1. INTRODUCTION	SSR 2/1 Req. 11 SSR 2/1 Req. 7 para 4.11(b)
2. OWNER'S TARGETS	N/A
3. PROJECT IMPLEMENTATION MODELS	SSR 2/1 Req. 11
4. CRITERIA FOR AN EFFECTIVE APPROACH TO CONSTRUCTABILITY	N/A
5. INSTALLATION ASPECTS AND CONSTRUCTION SOLUTIONS	SSR 2/2 Req. 13 para 4.48
6. COMMISSIONING	SSR 2/2 Req. 25 SSG-28 Sec. 2.19
6.1 Commissioning principles	SSR 2/1 Req. 25 paras 6.1 – 6.7, 6.11, 6.12, 6.14, 6.15
6.2 Start-up test programme	SSR 2/2 Req. 15 para 4.52 SSR 2/2 Req. 30 para 7.23 SSR 2/2 Req. 25 para 6.9
6.3 Commissioning organization	SSR 2/2 Req. 25 paras 6.10, 6.13

E.U.R. Chapter 2.13 is in **full compliance** with benchmarked international safety standards for construction and commissioning of new nuclear power plants.

However there are some outstanding provisions form the IAEA Safety Requirements that are not explicitly covered by E.U.R Chapter 13. These are:

The IAEA NS-R-3 on Site Evaluation of Nuclear Installations requires in para 2.29 that "*Before construction of the nuclear installation is started, it shall be confirmed that there will be no insurmountable difficulties in establishing an emergency plan for the external zone before the start of operation of the installation".*

The IAEA SSR 2/2 on Safety of Nuclear Power Plants: Commissioning and Operation stipulates in Requirement 25, para 6.8 the "*All functions of the operating organization shall be performed at the appropriate stage during commissioning. These functions shall include discharging responsibilities for management, training of personnel, the radiation protection programme, waste management, management of records, fire safety, physical protection and the emergency plan".*

Such requirements have not been found in E.U.R. Chapter 2.13.

3.14 E.U.R. CHAPTER 2.14 OPERATION, MAINTENANCE & PROCEDURES

The E.U.R. Chapter 2.14 on Operation, Maintenance and Procedures addresses more operational than design aspects and therefore its provisions have been compared more with the SSR 2/2 Requirements but partly also with the SSR 2/1 Requirements in the segment of benchmarking with the IAEA Safety Standards.

E.U.R Chapter 14 OPERATION, MAINTENANCE & PROCEDURES	IAEA Safety Standards
1. INTERFACE BETWEEN OPERATORS AND EQUIPMENT	TITLE
1.1 General design	SSR 2/1 Req. 32 SSR 2/1 Req. 37
1.2 Main Control Room And I&C	See 2.10
1.3 Operating area environment	SSR 2/1 Req. 32 paras 5.60, 5.61 SSR 2/1 Req. 73
1.4 Human-Machine interface in maintenance	SSR 2/1 Req. 7 para 4.12(d) SSR 2/1 Req. 32 paras 5.54, 5.55
2. PROVISIONS TO ENHANCE OPERABILITY AND MAINTABILITY(14.1)	SSR 2/1 Req. 27 SSR 2/1 Req. 82 para 6.84 SSR 2/2 Req. 14 para 4.50 SSR 2/2 Req. 25 para 6.13 SSR 2/2 Req. 31 para 8.1
3. PROVISIONS FOR EMERGENCY ARRANGEMENTS	GSR Part 7 Req. 24 para 6.25
4. SUPPORT FACILITIES REQUIREMENTS	TITLE
4.1 Access to radioactive areas	GSR Part 3 Req. 24
4.2 Plant services	SSR 2/1 Req. 27 SSR 2/1 Req. 37

E.U.R Chapter 14 OPERATION, MAINTENANCE & PROCEDURES	IAEA Safety Standards
4.3 Repair shops	SSR 2/1 Req. 23 para 5.38 SSR 2/1 Req. 29 para 5.45
4.4 Spare parts supply, storage and control	SSR 2/2 Req. 31 paras 8.15-8.17
5. PROVISIONS FOR REPLACEMNET OF MAJOR COMPONENTS	SSR 2/2 Req. 14 SSR 2/2 Req. 16 para 4.53 SSR 2/1 Req. 31
6. BASIS FOR PLANT LIFE MANAGEMENT	SSR 2/2 Req. 14 SSR 2/2 Req. 16
7. PERIODIC TESTING AND INSPECTION	SSR 2/1 Req. 80 para 6.66(c) SSR 2/2 Req. 31
8. CONSUMABLES FOR OPERATION AND MAINTNANCE	SSR 2/2 Req. 29 para 7.17
9. OCCUPATIONAL RADIATION EXPOSURE	GSR Part 3, SCHEDULE III-1
10. PLANT ARRANGEMET AND ACCESS	TITLE
10.1 Plant arrangements	See Ch. 2.11
10.2 Access within the plant	See Ch. 2.11
10.3 Access during emergency conditions	GSR Part 7 Req. 8 para 5.27 GSR Part 7 Req. 9 para 5.34 SSR 2/1 Req. 8
11. PROCEDURES GUIDELINES	TITLE
11.1 General	SSR 2/1 Req. 32 para 5.55
11.2 Normal procedures	SSR 2/2 Req. 26 para 7.2
11.3 AOO and DBA procedures	SSR 2/2 Req. 26 para 7.3
11.4 SAMGs	SSR 2/2 Req. 19

SSR 2/1 Requirement 28 stipulates that *"the design shall establish a set of operational limits and conditions for safe operation of the nuclear power plant".* Further in para 5.44 it lists which operational limits and conditions shall be included:

"5.44. The requirements and operational limits and conditions established in the design for the nuclear power plant shall include (Requirement 6 of IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation):

(a) Safety limits;

(b) Limiting settings for safety systems;

(c) Limits and conditions for normal operation;

(d) Control system constraints and procedural constraints on process variables and other important parameters;

(e) Requirements for surveillance, maintenance, testing and inspection of the plant to ensure that structures, systems and components function as intended in the design, to comply with the requirement for optimization by keeping radiation risks as low as reasonably achievable;

(f) Specified operational configurations, including operational restrictions in the event of the unavailability of safety systems or safety related systems;

(g) Action statements, including completion times for actions in response to deviations from the operational limits and conditions".

Such explicit list of operational limits and conditions to be considered in design has not been found in the E.U.R. document.

Nevertheless, the E.U.R Chapter 2.14 is considered to be in **full compliance** with benchmarked international standards as implicitly it covers requirements for operation and maintenance to be considered in the design stage.

3.15 E.U.R. CHAPTER 2.15 QUALITY ASSURANCE

E.U.R. Chapter 2.15 outlines the main principles for Quality Assurance programmes that shall be implemented for the design and construction of Light Water Reactors in Europe. It is a Quality Assurance programme but at the same time its requirements have been written in a way to be in accordance also with Leadership and Management for Safety issues as stipulated in the IAEA GSR Part 2 Requirements and EN ISO 9001 Quality management systems Requirements.

E.U.R. Chapter 2.15 QUALITY ASSURANCE	IAEA Safety Standards
1. BASIC REQUIREMENTS	SSR 2/1 Req. 3*
1.1 General requirements of QAP	SSR 2/1 Req. 2
2. INTERFACES	SSR 2/1 Req. 3
3. CONSTRUCTION VERIFICATION 4. HANDLING OF NON-CONFORMACES	SSR 2/1 Req. 2 SSR 2/1 Req. 3
5. RECORDS	SSR 2/1 Req. 3

* E.U.R Chapter 2.15 on Quality Assurance covers the Requirements 1, 2 and 3 from the IAEA SSR 2/1 Chapter 3. Management of Safety in Design in sufficient detail so that for all purposes we can conclude that E.U.R. Chapter 2.15 is **in full compliance** with the Chapter 3 of SSR 2/1. However there are some differences that should be pointed out.

SSR 2/1 Chapter 3 addresses the *Management system,* whereas E.U.R. Chapter 2.15 addresses the *Quality Assurance Programme.* These two concepts are similar but not identical and the IAEA Safety Standards are making a clear distinction between the two already in their early versions issued even before the Fukushima Daiichi revisions. E.U.R Chapter 2.15 recognizes this development and clearly states that "the overall QAP of the project, as well as management processes and QAPs of Contractors, and review organisations, shall be in accordance with the requirements of the nuclear specific safety standard IAEA GSR Part 2: Leadership and Management for Safety (2016)" (2.15.1E). It further states, that the Quality Assurance programme is part of the Management system (2.15.1A) and further, on few places emphasises, for example, that "where potential contradiction between ISO 9001 (2015) – being a general quality standard for products and services - and IAEA GSR Part 2 (2016) – being nuclear safety standards - exists in the implementation phase, the nuclear specific features of the IAEA GSR Part 2 (2016) standard shall be mandatory QAP input for nuclear related Structures, Systems and Components (SSCs)" (2.15.1F).

GSR Part 2 on Leadership and Management for Safety clearly supports the Requirements 1, 2 and 3 from SSR 2/1 which deal with the Management of Safety in Design.

Requirement 1 from SSR 2/1 which defines the 'Responsibility in the management of safety in plant design", determining that an applicant for licence to construct and/or operate a nuclear power plant shall be responsible for ensuring that the design submitted to the regulatory body meets all applicable safety requirements. This responsibility is only implicit in the E.U.R. Chapter 2.15. It is stated in 2.15.1A that "the award of contracts for the project does not change the safety and quality responsibilities of the Owner", which can be interpreted to cover the same undivided responsibility.

Provisions from SSR 2/1 Requirement 3 on establishing a formal system for ensuring the continuing safety of the plant design throughout the lifetime of the nuclear power plant are addressed to a certain extend in E.U.R. Chapter 2.12.1.1.4. It however does not call for establishment of a "formally designated entity responsible for the safety of the plant design within the operating organization's management system" - (SSR 2/1 Req. 3 para 3.5). In describing the responsibilities of such designated entity in SSR 2/1 Req. 3 para 3.6 all provisions are covered by the E.U.R. except the one that relates to “maintaining of safety culture is included in the formal system for ensuring the continuing safety of the plant design” – SSR 2/1 Req. 3; 3.6(b). However, even though safety culture is not mentioned in E.U.R Chapter 2.15, if we compare the safety culture definition from INSAG-4 “*Safety Culture is that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance”*, with the requirements stipulated in E.U.R. Chapter 2.15, we can find all the elements being adequately covered.

It has been stated above, that E.U.R. Chapter 2.15 in in full compliance with SSR 2/1 Chapter 3. It should however be pointed out that this benchmarking exercise covers also other IAEA Safety Requirements. E.U.R. in the introduction to Chapter 2.15 claims that Quality Assurance programmes shall be in accordance with requirements of the IAEA GSR Part 2. Whereas it is true that they are in accordance, it does not mean that they also cover all aspects presented in all 14 Requirements of GSR Part 2. They still remain to be Quality Assurance programs and not Leadership and Management for Safety programs as required by the IAEA Safety Standards.

3.16 E.U.R. CHAPTER 2.16 DECOMISSIONING

In most countries today the initial decommissioning plan is a licensing requirement already in the design stage. In the initial decommissioning plan a feasibility study has to be performed to demonstrate that plant can be safely decommissioned and dismantled using the current technology even though the technological advances might in decades to come bring more favourable solutions.

Decommissioning is facilitated if planning and preparatory work is undertaken already at the design phase of the project and then carries out throughout the lifetime of the plant. Decommissioning and dismantling might take several possible routes from immediate dismantling to deferred dismantling or entombment. These decisions may be but do not have to be taken at the design stage; feasibility study however has to accommodate all possible options.

This E.U.R. Chapter deals only with provisions that need to be taken into account at the design stage to facilitate future decommissioning and dismantling. Therefore the IAEA Safety Requirement GSR Part 6 and WS-R-5 apply only in this limited scope.

E.U.R Chapter 16 DECOMISSIONING	IAEA Safety Standards
1. INTRODUCTION	SSR 2/1 Req. 6 SSR 2/1 Req. 80 para 6.67(g)
2. DECOMMISSIONING PLANS	GSR Part 6 Chapter 7 SSR 2/1 Req. 12 para 4.20 WS-G-2.1 paras 4.1 – 4.4
2.1 Initial decommissioning plan(16.1)	SSR 2/1 Req. 12 GSR Part 6 Req. 10
2.2 Table of content of the decommissioning plans	GSR Part 6 Reqs. 10, 11
3. DESIGN FOR DECOMMISSIONING	GSR Part 6 Chapter 2 SSR 2/1 Req. 12 para 4.20

4. DOCUMENTATION FOR DECOMMISSIONING	SSR 2/1 Req. 3 para 3.6(h)

(16.1) E.U.R. 2.16.2.1D and 2.16.4A address the decommissioning costs but only as the responsibility of the designer to provide estimates. GSR Part 6 Req. 9 on Financing of decommissioning is much more precise in determining responsibilities in respect of financial provisions for decommissioning and states:

Responsibilities in respect of financial provisions for decommissioning shall be set out in national legislation. These provisions shall include establishing a mechanism to provide adequate financial resources and to ensure that they are available when necessary, for ensuring safe decommissioning".

To place requirements on the national legislation is out of scope of the E.U.R. and therefore it can be established that E.U.R. Chapter 2.16 on Decommissioning is **in full compliance** with the benchmarked international standards.

3.17 E.U.R. CHAPTER 2.17 PSA METHODOLOGY

This Chapter of E.U.R. has been written on the basis of the IAEA PSA guidelines for conduction Level-1 and Level-2 probabilistic assessment (IAEA SSG-3 and SSG -4) and on the basis of the EU guidelines for Harmonization of European Practices in PSA Level-1 methodology and data collection.

The Chapter makes reference to the INSAG-3 document in 2.17.1.2.A.A1. There is nothing wrong with it as all principles still apply, however it should be pointed out that INSAG-12 has superseded in many aspects the old INSAG-3 document. The name of the Group has also changed in the meantime from International Nuclear Safety Advisory Group to be the International Nuclear SAfety Group.

The Chapter defines the objectives of the PSA in plant design, scope of the PSA and its general format and methodology and assumptions to be used when performing PSA.

From the table below it is evident that E.U.R. Chapter 2.17 takes full account of both above mentioned international guidance documents and it can be therefore concluded that E.U.R Chapter 2.17 is **in full compliance** with the benchmarked international standards.

E.U.R. Chapter 2.17 PSA METHODOLOGY	IAEA Safety Standards
1 OBJECTIVES OF THE PSA	SSG-3 para 3.1-3.2 INSAG-6 para 2.2, 2.6
1.1 Support to design	SSG 2/1 Req. 6 para 4.7 SSR 2/1 Req. 10 SSR 2/1 Req. 16 para 5.5 SSR 2/1 Req. 20 SSR 2/1 Req. 42 para 5.76 SSG-30 para 10.73 INSAG-6 para 2.7

E.U.R. Chapter 2.17 PSA METHODOLOGY	IAEA Safety Standards
1.2 Verification of the design	SSR 2/1 Req. 2 para 3.4 SSR 2/1 Req. 3 para 3.6(b) INSAG-6 para 2.5 ISNAG-12
1.3 Assistance to plant operation	SSR 2/2 Req. 8 para 4.32 SSR 2/2 Req. 12 para 4.46 SSR 2/2 Req. 31 para 8.5(a) SSR 2/2 Req. 31 para 8.6, 8.13 SSG-25 Safety factor 6
2 SCOPE OF THE PSA	SSG-3 para 2.2-2.4
2.1 Scope for the design	SSR 2/1 Req. 10 para 4.17 INSAG-6 para 2.7.3
2.2 Definition of Core Damage	NS-G-1.10 para 3.27
2.3 Radioactive sources	INSAG-12
2.4 Plant operation modes	SSG-3 Ch. 9
2.5 Type of initiating events	SSG-3 Ch. 7 and Ch. 8
2.6 Types of quantification	SSG-3 paras 5.140-5.150 NS-G-1.10 paras 4.53-4.58
3 PLANT ANALYSIS	TITLE
3.1 Plant model structure	SSG-3 paras 6.2 – 6.5
3.2 Selection of initiating events	SSG-3 Ch. 7 and Ch. 8
3.3 Event sequence delineation	SSG-3 paras 5.57, 9.26
3.3.1 Event tree construction	SSG-3 paras 5.6, 5.57-5.60, 5.87
3.3.2 Success criteria	SSG-30 paras 5.46-5.56, 9.16, 9.22-9.25
3.3.3 Mission time	SSG-3 paras 5.49, 5.135, 9.28, 9.53
3.4 System modelling	SSG-3 para 9.31, Annex III
3.5 Human reliability analysis	SSG-3 para 5.96 – 5.113 INSAG-6 para 2.7.4
3.6 Treatment of dependencies	SSG-3 paras 5.86-5.91
3.7 Reliability data	SSG-3 para 9.46
3.7.1 Initiating event frequencies	SSG-3 paras 5.121-5.139
3.7.2 Component failure, test, repair and maintenance data	SSG-3 paras 5.69, 5.137, 9.46, 9.49
3.7.3 Common-Cause Failure data	SSG-3 para 5.95-5.107 INSAG-6 para 2.7.5
3.7.4 Human error probabilities	SSG-3 paras 5.95-5.107

E.U.R. Chapter 2.17 PSA METHODOLOGY	**IAEA Safety Standards**
3.8 Accident sequence quantification	SSG-3 paras 5.40-5.68
4 CONTAINMENT ANALYSIS	SSR 2/1 Req.20 para 5.30 SSG-3 para 2.29 NS-G-1.10
4.1 Core damage sequence binning	SSG-3 paras 5.62-5.68 NS-G-1.10 para 3.27
4.2 Containment system analysis	SSG-3 paras 9.28-9.30 NS-G-1.10 Ch. 2
4.3 Containment isolation	SSG-3 para 9.29(c) NS-G-1.10 paras 4.169, 4.170, 4.179, 4.181
4.4 Containment bypass accident	SSG-3 para 5.28 NS-G-1.1- paras 4.145-4.146 TECDOC-1791 Appendix 4, para 8
4.5 In-plant sequence assessment	SSG-3 paras 9.26, 9.28
4.6 Containment event tree analysis	SSG-3 paras 2.29, 5.30,
4.7 In-Containment source term definition	SSG-3 para 9.28 NS-G-1.10 paras 4.121-4.123
5 INTERNAL HAZARDS	SSG-3 Ch. 7; Annex 1
5.1 Fires	SSG-3 paras 7.12 – 7.65
5.2 Flooding	SSG-3 paras 7.66 – 7.92
6 EXTERNAL HAZARDS	SSG-3 Ch. 8; Annex 1
6.1 Earthquakes	SSG-3 para 8.8
6.2 External flooding	SSG-3 paras 8.10 - 8.11
6.3 Temperatures	SSG-3 paras 8.12, 8.23
6.4 High winds and Tornadoes	SSG-3 para 8.9
6.5 Precipitation	SSG-3 para 8.10 (d)
6.6 Drought	SSG-3 paras 8.12, 8.23
6.7 Lightning	SSG-3 paras 8.12, 8.23
6.8 Aircraft impact	SSG-3 para 8.14 (d)
6.9 Hazards from industrial installations	SSG-3 paras 8.14 (a), (b), (c)
6.10 Electromagnetic interference	SSG-3 para 8.14 (h)
7 UNCERTAINTY, SENSITIVITY AND IMPORTANCE ANALYSIS	SSG-3 paras 5.151–5.160
7.1 Uncertainty and sensitivity analysis	SSG-30 paras 9.58-9.60
7.2 Importance analysis	SSG-30 paras 9,58-9.60, 10.19-10.22

E.U.R. Chapter 2.17 PSA METHODOLOGY	IAEA Safety Standards
8 DOCUMENTATION AND QUALITY ASSURANCE OF THE STUDY	TITLE
8.1 Quality assurance	SSG-3 paras 3.13, 3.14
8.2 Documentation	SSG-3 paras 9.61-9.71

3.18 E.U.R. CHAPTER 2.18 PERFORMANCE ASSESSMENT METHODOLOGY

E.U.R. Chapter 2.18 discusses the integration of reliability, availability and maintainability requirements into design. It focuses on plant performance, production and availability targets. Even though it is recognized in the IAEA Safety standards that plant availability and safety go hand in hand i.e. very often depend on the same management principles, these aspects of plant operation are not covered in the IAEA Safety Standards. Therefore being more detailed in all these aspects then the IAEA Safety Standards, it can be concluded that E.U.R. Chapter 2.18 is **in full compliance** with benchmarked international standards.

E.U.R Chapter 2.18 PERFORMANCE ASSESSMENT METHODOLOGY	IAEA Safety Standards
1 INTRODUCTION	SSR 2/2, Req. 16 para 5.15 SSR 2/2 Req. 28 para 5.44(f)
2 PAM GOALS AND OBJECTIVES	SSR 2/2, Req. 32
3 AVAILABILITY PERFORMANCE REQUIREMENTS	SSR 2/2, Req. 9
3.1 Production availability requirements	SSR 2/2, Req. 9
3.2 Impact of classified equipment	SSR 2/1 Req. 14
4 PRELIMINARY TASKS PRIOR TO THE PAM ANALYSIS PROCESS	SSR 2/1 Req. 14
4.1 Experience review	SSR 2/1 Req. 3 para 3.6(b),(c) SSR 2/1 Req. 6 para 4.6 SSR 2/1 Req. 9 para 4.16 SSR 2/1 Req. 11 para 4.19 SSR 2/2 Req. 24
4.2 Reliability and availability performance	SSR 2/1 Req. 16 para 5.15
4.3 Failure mechanisms	SSR 2/1 Req. 17, 5.15A SSR 2/1 Req. 31
4.4 Specific system design features	SSR 2/1, para 2.13 (2) TECDOC-1791, 4.2. (2)
4.5 Start-up test programme	SSR 2/2 Req. 25
4.6 Components reliability database	SSG-3 paras 5.77, 9.46

E.U.R Chapter 2.18 **PERFORMANCE ASSESSMENT METHODOLOGY**	**IAEA Safety Standards**
	SSG-39 paras 6.6 – 6.76
5 AVAILABILITY ANALYSIS PROCESS	SSR 2/1 Req. 23 SSR 2/1 Req. 62
5.1 Analysis requirements	
5.2 Development of main functions level models	NS-G-1.3 para 2.40
5.3 Modelling techniques	SSG-25 paras 5.55-5.60 INSAG-12 para 4.2.1
5.4 Unavailability allocation	
5.5 Content analysis	
5.6 Reference outage programmes	SSR 2/2 Req. 32
6 RECOMMENDED RELIABILITY ACTIVITIES	SSR 2/1 Req. 29, 30
6.1 Reliability activity requirements	SSR 2/1 Req. 30 SSG-39 paras 6.6 – 6.76
6.2 Maintenance activity requirement	SSR 2/1 Req. 29 SSG-39 paras 6.192 – 6.197
7 DOCUMENTATION	SSR 2/1 Req. 14
7.1 Monitoring of PAM design assumptions	SSR 2/2, Req. 6
7.2 Design organisation interface	SSR 2/1 Req. 2
8 HISTORICAL LWR UNAVAILABILITY	N/A
8.1 Table 1 - Historical PWR unavailability	N/A
8.2 Table 2 - Historical BWR unavailability	N/A
9 MAIN SYSTEMS CONTRIBUTING TO CURRENT LWR UNPLANNED TRIPS	N/A
9.1 Table 3 - current PWR unplanned trips	N/A
9.2 Table 4 - current BWR unplanned trips	N/A
Appendix A - METHODOLOGY FOR ASSESSING OVERALL AVAILABILITY	N/A

3.19 E.U.R. CHAPTER 2.19 COST ASSESSMENT INFORMATION REQUIREMENTS

The E.U.R Chapter 2.19 deals with cost information which is essential for bid comparisons and confirmation that the project economic goals are accomplished, especially when comparison with other energy sources is required.

The IAEA Safety Standards however do not address these issues in any way and therefore comparison of this Chapter with the IAEA Safety Requirements does not yield any results.

E.U.R Chapter 2.19 COST ASSESSMENT INFORMATION REQUIREMENTS	IAEA Safety Standards
1. INTRODUSCTION	N/A
2. LEVELISED COSTING METHODOLOGY	N/A
3. COST INFORMATION	N/A
4. OTHER DATA REQUIREMENTS	N/A
5. BASES OF INFORMATION	N/A

3.20 E.U.R. CHAPTER 2.20 ENVIRONMENTAL IMPACT

E.U.R. Chapter on Environmental impact assessment (EIA) covers three phases of plant life:

- Construction phase

- Operational phase

- Decommissioning phase

The Chapter is based mostly on the EU Directives in the field of Environmental impact assessment. This chapter is much more detailed than EIA addressed in the IAEA Safety Standards and being based on the EU Directives it is found to be in **full compliance** with the benchmarked international standards.

E.U.R Chapter 2.20 ENVIRONMENTAL IMPACT	IAEA Safety Standards
1. OBJECTIVES OF ENVIRONMENTAL PROTECTION	SF-1 Ch. 2 SF-1, Principle 7 para 3.28
2. ASSESSMENT OF ENVIRONMENTAL IMPACT	GSR Part 4 paras 1.9, 2.6(d) NS-R-3 para 2.3
3. ENVIRONMENTAL PROTECTION REQUIREMENTS OF CONSTRUCTION PAHASE	NS-R-3 paras 2.3, 2.23, 2.24
3.1 Land use	SSG-18 paras 3.38 – 3.39
3.2 Dredging requirements	
3.3 Hydrological alteration	SSG-18 Section 5 INSAG-12 para 137
3.4 Water use	
3.5 Noise	
4. ENVIRONMENTAL PROTECTION REQUIREMENTS OF OPERATIONL PHASE	SSR 2/2 Req. 1 para 3.3 SSR 2/2 Req. 2 para 3.5
4.1 Land use	SSG-18 paras 3.38 – 3.39 INSAG-12 para 139
4.2 Hydrological alteration	SSG-18 Section 5 INSAG-12 para 139

E.U.R Chapter 2.20 ENVIRONMENTAL IMPACT	IAEA Safety Standards
4.3 Water use	INSAG-12 para 139
4.4 Noise	INSAG-12 para 139
4.5 Thermal discharge	
4.6 Release of hazardous or toxic chemicals	GSR Part 7 Req. 4 SSR 2/2 Req. 29 para 7.17 NS-R-3 paras 3.48-3.50
4.7 Radiological impact on human health	SSR 2/1 Req. 82 INSAG-12 para 4.1.2
4.7.1 Monitoring of gaseous and liquid releases	NS-G-1.10 paras 4.84, 4.85
4.7.2 Radiological impacts	SSR 2/1 Req. 82 INSAG-12 para 4.1.2
4.8 Radiological impact on flora and fauna	SSR 2/1 Req. 82 INSAG-12 para 4.1.2
5. ENVIRONMENTAL PROTECTION REQUIREMENTS OF DECOMMISSIONING PHASE	GSR Part 6, Req. 6 para 3.4 WS-G-2.1 paras 2.17 and 5.11 INSAG-12 para 4.7
5.1 Land use	WS-G-2.1 para 7.13
5.2 Contamination of soil and groundwater	WS-G-2.1 para 7.13
5.3 Noise	
5.4 Radiological impact to human health	WS-G-2.1 para 2.15
5.5 Radiological impact of flora and fauna	WS-G-2.1 para 2.15
6. FIGURE AND TABLES	N/A

GENERAL CONCLUSIONS, OUTSTANDING ISSUES

After going through the entire Set of chapters of E.U.R. Volume 1 and 2 we have looked also through the IAEA SSR 2/1 Rev 1 standards to determine if any of the 82 Requirements have not been addressed by the E.U.R. including all individual paragraphs. Only one Requirement was found to be missing and that is the Requirement 35 which reads as follows:

"Requirement 35: Nuclear power plants used for cogeneration of heat and power, heat generation or desalination

Nuclear power plants coupled with heat utilization units (such as for district heating) and/or water desalination units shall be designed to prevent processes that transport radionuclides from the nuclear plant to the desalination unit or the district heating unit under conditions of operational states and in accident conditions".

It is not considered to be a shortcoming in the E.U.R. document and it can be concluded from the above tables which benchmark the E.U.R. Chapter 2.1 with the IAEA Safety Standards that there is **a full compliance** of the E.U.R. Chapter 2.1 with other international standards.

Perhaps the largest difference between the E.U.R. requirements and international safety standards can be found in E.U.R. Chapter 15. It is the chapter that covers Quality Assurance. As pointed out already in the benchmarking of this chapter above, it is the concept that differs and not the un-fulfilment of certain safety requirements from other international standards. The IAEA Safety Standards have moved long ago from the concept of Quality Assurance and have introduced the concept of Management Systems already in their early publications such as GS-R-3 “Management System for Facilities and Activities” and GS-G-3.1 “Application of the Management System for Facilities and Activities”. Already in these documents, safety culture plays an important role. Safety Culture was first introduced in INSAG-4 document already in 1991 after incorporation of lessons learned from the Chernobyl accident and this concept was in many ways supposed to broaden the aspects of quality assurance. After the incorporation of lessons learned from the Fukushima Daiichi accident the GS-R-3 Requirements and associated Safety Guides were superseded by GSR Part 2 on “Leadership and Management for Safety”.

4 BENCHMARKING OF VOLUME 4 OF E.U.R. AGAINST THE IAEA SAFETY STANDARDS

4.1 E.U.R. VOLUME 4

Volume 4 of E.U.R. establishes requirements for the Power Generation Plant of the Light Water Reactor Nuclear Power Plants to be built in Europe. The Volume presents the objectives of these requirements, as well as a general description of Power Generation Plant.

Chapter 4.1 is an introductory chapter.

Chapter 4.2 deals with the Main Turbine Generator Systems and covers the Main turbine, Main generator and Main condensed, together with the associated auxiliary and support systems. Short description of each system is presented in terms of scope, interfaces and overall functions.

Chapter 4.3 presents the generic requirements relating to steam, condensate and feed water systems of the Power Generation Plant. Again, short description is given for individual systems in order to clarify the scope and application of the requirements.

Chapter 4.4 deals with the electric power system for the Power Generation Plant. Aspects like transmission, electrical protection, lightening protection or earthing are covered in this chapter by specific requirements.

Chapter 4.5 presents requirements for the Circulating water system. It covers main and auxiliary circulating water system, chemical injection system, acid feeding system and make-up water treatment plant.

Chapter 4.6 is the last chapter of Volume 4 and presents requirements for the auxiliary systems. All together 11 auxiliary systems are described (see table below) with main characteristics that clarify the application of the requirements.

E.U.R Chapter 4	**IDENTIFICATION IN E.U.R.**
4.1 INTRODUCTIN TO VOLUME 4	T
4.1.1 Objectives	T
4.1.2 REFERENCES TO VOLUME 2	T
4.1.3 INTERFACES BETWEEN THE POWER GENERATION PLANT (PGP) AND THEN NUCLEAR ISLAND (NI) OR GRID	T
4.2. MAIN TURBINE GENERATOR SYSTEM	T
4.2.1 SYSTEM DESCRIPTION	T
4.2.2 SYSTEM FUNCTIONS	T
4.2.3 COMMON SYSTEM AND COMPONENT REQUIREMENTS	T
4.2.4 MAIN TURBINE	T

E.U.R Chapter 4	IDENTIFICATION IN E.U.R.
4.2.5 MAIN GENERATOR SYSTEM	T
4.2.6 GENERATOR AUXILIARY SYSTEMS	T
4.2.7 MAIN CONDENSER	T
4.2.8 TURBINE GENERATOR CONTROL SYSTEM	T
4.2.9 MONITORING AND DIAGNOSTIC FUNCTIONS	T
4.3 STEAM, CONDENSATE AND FEEDWATER SYSTEMS	T
4.3.1 INTRODUCTION	T
4.3.2 MAIN STEAM SUPPLY AND EXTRACTION SYSTEM	T
4.3.3 FEEDWATER AND CONDENSATE SYSTEM	T
4.4 ELECTRIC POWER SYSTEMS	T
4.4.1 ELECTRICAL POWER TRANSMISSION AND AUXILIARIES NORMAL SUPPLY	T
4.4.2 AUXILIARIES STENDBY SUPPLY	T
4.4.3 ELECTRICAL PROTECTION SYSTEM	T
4.4.4 LIGHTNING AND SMALL POWER SYSTEMS	T
4.4.5 LIGHTNING PROTECTION SYSTEMS	T
4.4.6 PLANT EARTHING SYSTEMS	T
4.4.7 FIGURE – AUXILARIES POWER SUPPLY	T
4.5 CIRCULATING WATER SYSTEM	T
4.5.1 MAIN CIRCULATING WATER SYSTEM (MCWS)	T
4.5.2 AUXILIARY CIRCULATING WATER SYSTEM (ACWS)	T
4.5.3 CHEMICAL INJECTION SYSTEM (CIS)	T
4.5.4 ACID FEEDING SYSTEM (ACFS)	T
4.5.5 MAKE-UP WATER TREATMENT PLANT	T
4.6 AUXILIARY SYSTEMS	T
4.6.1 TURBINE BUILDING COMPONENT COOLING WATER SYSTEM (TBCCWS)	T
4.6.2 PGP HEATING, VENTILATION AND AIR CONDITIONING SYSTEM (HVAC)	T
4.6.3 AUXILIARY STEAM SYSTEM	T
4.6.4 FIRE PROTECTION SYSTEM	T
4.6.5 NON RADIOACTIVE LIQUID WASTE SYSTEM	T
4.6.6 RAW WATER SYSTEM	T
4.6.7 WATER TREATMENT SYSTEM	T
4.6.8 DEMINERALIZED WATER STORAGE AND TRANSFER SYSTEM	T

E.U.R Chapter 4	IDENTIFICATION IN E.U.R.
4.6.9 STEAM GENERATOR BLOWDOWN SYSTEM (PWR ONLY)	T
4.6.10 RECOMBINER SYSTEM (BWR ONLY)	T
4.6.11 COMPRESSED AIR AND GAS SYSTEM	T

CONCLUSIONS

All requirements in Volume 4 are marked as T and as such are not to be benchmarked against international standards. During the kick-off meeting in Luxemburg on 17th October 2017 with the DG-ENER and EUR representatives it was decided that only requirements marked as N (nuclear) or C (common) will fall under the benchmarking exercise. Nevertheless, the entire Volume 4 has been screened to verify that specific requirements to individual systems described in Volume 4 are not covered by the international safety standards, at least not in such details as in the E.U.R documents. Therefore the general conclusion on Volume 4 of the E.U.R. would be that **international safety standards are not applicable to Volume 4 of the E.U.R.**

5 BENCHMARKING E.U.R. AGAINST WENRA SAFETY REFERENCE LEVELS FOR EXSISTING REACTORS (REPORT OF 24TH SEPTEMBER 2014)

The principal aim of the WENRA Safety Reference Levels for Existing Reactors document (2014) is to harmonize approaches to safety in WENRA Member States. The emphasis of the document is on nuclear safety and does not include nuclear security and only partially radiation protection. Considering that nuclear power plant technologies in WENRA Member States consist of PWR, BWR, CANDU and Gas-cooled reactors and that national legislations of Member States might be different, the WENRA Reference Levels do not cover in-depth legal and technical details. Where necessary, they refer to the appropriate IAEA standards. As within this benchmarking study, the E.U.R. are benchmarked also against the IAEA safety standards, it can be concluded that these aspects have already been covered. After the Fukushima Daiichi accident a new Reference Level Issue T on Natural Hazards has been added to the previous edition of 2006.

WENRA RHWG Reference Levels for Existing Reactors address the safety of nuclear power plants in operation whereas E.U.R. have been anticipated mostly as requirements to be fulfilled in the design stage. There is a significant overlap in both requirements and it is therefore feasible to make a benchmarking comparison of both sets of requirements bearing in mind that E.U.R are addressing operational issues that need to be considered in the design stage and do not address operational requirements for a plant that is already in operation. For this reason, a number of WENRA RHWG Reference Levels for Existing Reactors are not applicable to the E.U.R. which is noted as N/A in the comparison tables. All WENRA Issues marked as Safety Management (A – D), Design (E – G, T), Safety Verification (N-P), Emergency Preparedness (R-S) have direct relevance to EUR, whereas Operation (H-M, Q) are of less relevance.

On few occasions, WENRA and E.U.R. refer to safety analysis, defence-in-depth, common cause failures and similar broad topics that have already been addressed in benchmarking the E.U.R. with the IAEA Safety Standards. On such places, the conclusion on compliance may have been made based on the insights gained when benchmarking against the IAEA Safety Standards and no specific cross-reference has been made to the WENRA document in order to avoid duplication of findings. These circumstances have however been kept to a minimum and in the great majority of WENRA SRLs the corresponding E.U.R. requirement has been identified.

WENRA RHWG SRL for Existing Reactors		E.U.R.
01 Issue A:	Safety Policy (1)	Chapter 2.15.1
A1.	Issuing and communication of a safety	Chapter 2.15.1
A2.	Implementation of the safety policy and monitoring safety performance	Chapter 2.15.1
A3.	Evaluation of the safety policy	Chapter 2.15.1
02 Issue B:	Operating Organisation	Chapter 2.14.2.
B1.	Organisational structure (2)	Chapter 2.14.2.1-2.14.2.2, 3
B2.	Management of safety and quality (3)	Chapter 2.15; Chapter 2.14.2.8
B3.	Sufficiency and competency of staff (4)	Chapter 2.14.2.3
03 Issue C:	Management System (5)	Chapter 2.15 (complemented with the reference to the IAEA GSR Part 2)
C1	Objectives	Chapter 2.15.1

WENRA RHWG SRL for Existing Reactors		E.U.R.
C2	General requirements	Chapter 2.15.1.1
C3	Management commitment	GSR Part 2 Req. 3
C4	Resources	GSR Part 2 Req. 9
C5	Process implementation	GSR Part 2 Req. 10
C6	Measurement , assessment and improvement	GSR Part 2 Req. 13
C7	Safety culture	GSR Part 2 Req. 12
04 Issue D:	Training and Authorization of NPP Staff (Jobs with Safety Importance) (6)	Chapter 2.14.2.3.
D1.	Policy	Chapter 2.14.2.3
D2.	Competence and qualification	Chapter 2.14.2.3.
D3.	Training programmes and facilities	Chapter 2.14.2.3.
D4.	Authorization	N/A
05 Issue E:	Design Basis Envelope for Existing Reactors (7)	Chapter 2.1.2.3
E1.	Objective	Chapter 2.1.1.1 and 2.1.3.1.
E2.	Safety strategy	Chapter 2.1.1.4
E3.	Safety functions	Chapter 2.1.1.2
E4.	Establishment of the design basis	Chapter 2.1.2.3.
E5.	Set of design basis events	Chapter 2.1.2.2
E6.	Combination of events	Chapter 2.1.2.2D
E7.	Definition and application of technical acceptance criteria	Chapter 2.1.2.2 C4; 2.1.3.3 B1; 2.1.4.2; 2.1.6.1; 2.1.9.2 Table 3 Chapter 2.1/ Appendix B E7.1 – Chapter 2.1.2.2 E7.2 – Chapter 2.1.7.1.1 and related chaps: Chapter 2.2.3; Chapter 2.6.4.11 Chapter 2.7.11 Chapter 2.8.3.2 E7.3 – Chapter 2.1.7.2.2 and related chaps: Chapter 2.8.2.1.2.3 Chapter 2.8.3.4.1.3.6

WENRA RHWG SRL for Existing Reactors		E.U.R.
		E7.4 – Chapter 2.1.7.2.2 and related chaps: Chapter 2.8.2.1.2.3 Chapter 2.8.3.4.1.3.6 E7.5 – Chapter 2.1A16 and related chaps: Chapter 2.9.4.1.2.3.3 Chapter 2.9.3.1.4.4.2 Chapter 2.9.4.1.1.1
E8.	Demonstration of reasonable conservatism and safety margins	Chapter 2.1.4.2 Chapter 2.1.4.2.A2; 2.1.4.2.3 E8.1 – Chapter 2.1.4.2.3 E8.2 – Chapter 2.1.4.2.2C and Chapter 2.1.6.3.1 E8.3 – Chapter 2.1.4.2.2 E8.4 – Chapter 2.1.9.4 and Chapter 2.1.8.1.1D second bullet E8.5 – Chapter 2.1.4.2B E8.6 – Chapter 2.1.2.2F E8.7 – Chapter 2.1.4.1 E8.7a – Chapter 2.1.4.1 and Chapter 2.1.4.2A1-2 E8.7b – Chapter 2.1.4.1F E8.7c – Chapter 2.1.4.2.3 and Chapter 2.1.8.1.1D E8.7d – Chapter 2.1.4.1G
E9.	Design of safety functions	E9.1 - Chapter 2.1.6.5;2.1.7.4.3B;2.1.7.5I1 fail-safe criterion E9.2 - Chapter 2.1.6.3.1;2.1.4.2.2; single failure criterion E9.3 - Chapter 2.1.6.7.1B 30 minutes rule E9.4 - Chapter 2.1.6.4 reliability of systems E9.5 - Chapter 2.1.6.6 multiple units E9.6 – Chapter 2.8.2.1.1.5A two diverse systems E9.7 – Chapter 2.8.2.1.1.5B adequate margin

WENRA RHWG SRL for Existing Reactors		E.U.R.
		E9.8 – Chapter 2.8.2.1.1.5C and 2.8.2.2.1.1H sub-criticality E9.9 – Chapter 2.1.7.2.5 and Chapter 2.8.3.4.1.3.1 heat removal from the core and 2.8.3.10.4 heat removal from the SFP E9.10 – Chapter 2.9.1.3 Chapter 2.9.3.1.3.4 Chapter 2.9.4.3B1 Chapter 2.9.1.2 Fig. 1 containment system functions E9.11 and E9.12 – Chapter 2.1.7.3.3A Chapter 2.9.3.1.3.4 and 2.9.4.1.7 Chapter 2.9.2.2.1.4 Chapter 2.9.4.1.9 containment penetrations
E10.	Instrumentation and control systems	E10.1 – Chapter 2.1.7.4.1 and Chapter 2.10.1.1 purpose of I&C E10.2 – Chapter 2.1.5.1.7 and Chapter 2.10.7.5.3.2.1.2 environmental qualification E10.3 – Chapter 2.1.7.4.6and Chapter 2.10.6.4.2.3.2 Main Control Room E10.4 – Chapter 2.1.1.4A2 E10.5 – Chapter 2.1.7.4.6C E10.6 –Chapter 2.1.7.4.7 E10.7 – Chapter 2.1.7.4.3B and Chapter 2.8.3.13 protection system E10.8 – Chapter 2.1.7.4.3A and Chapter 2.8.3.13 E10.9 – Chapter 2.1.7.4.3C E10.10 – Chapter 2.8.3.13 E10.11 – Chapter 2.1.7.5 emergency power
E11.	Review of the design basis	N/A
06 Issue F:	Design Extension of Existing Reactors	Chapter 2.1.2.4

WENRA RHWG SRL for Existing Reactors		E.U.R.
F1.	Objective [8]	F1.1 – Chapter 2.1.2.4A objectives F1.2 – Chapter 2.1.2.4A1 DEC-A and DEC-B
F2.	Selection of design extension conditions	F2.1 – Chapter 2.1.2.4A DSA+PSA+EJ F2.2 – Chapter 2.1.2.4.1A and 2.6A,B hazards, CCF, multiple units F2.3 – Chapter 2.1.2.4.2 DEC B
F3.	Safety analysis of design extension conditions	F3.1 – Chapter 2.1.2.4A2, B1, B3 methods; 4.2.3B cliff-edge effects; 4.3B use of PSA a) - Chapter 2.1.2.4A,B and Chapter 2.1.4.2.3B b) – Chapter 2.1.4.1F c) – Chapter 2.1.2.4A,B d) – Chapter 2.1.4.1B e) – Chapter 2.12.10.3.1 f) – Chapter 2.1.4.1F g) – Chapter 2.1.4.3B and Chapter 2.17 h) – Chapter 2.17.3.3.2C and Chapter 2.17.4.5 and Chapter 2.17.7.1AB i) – Chapter 2.17.3.3.3
F4.	Ensuring safety functions in design extension conditions	F4.1 – Chapter 2.1.1.2 Fundamental safety functions F4.2 – Chapter 2.1.5.1.2.3 safety category 3 functions F4.3 – Chapter 2.1.6.7.1.1 mobile equipment F4.4 – Chapter 2.1.6.7.1 common services and supplies F4.5 – Chapter 2.1.6.7 autonomous supplies F4.6 – Chapter 2.1.7.5G reactor and 7.7 SFP sub-criticality F4.7 – Chapter 2.1.7.5 reactor and 7.7.1 SFP removal of heat F4.8 – F4.14 – Chapter 2.9 containment performance in DEC F4.15 – Chapter 2.10.7.2.2.6 qualified instrumentation F4.16 – Chapter 2.10 6.4.2.3.2S habitable control room

WENRA RHWG SRL for Existing Reactors		E.U.R.
		F4.17 – Chapter 2.1.7.5 power supply in DEC F4.18 – Chapter 2.1.6.7.3E,F,G batteries in DEC
F5.	Review of the design extension conditions	N/A
07 Issue G:	Safety Classification of Structures, Systems and Components	Chapter 2.1.5. Chapter 2.4.1.2.1.2
G1.	Objective	G1.1 – Chapter 2.1.5.1.4A
G2.	Classification process (9)	G2.1 – general G2.2 – Chapter 2.1.5.1.4B; 2.1.5.1.5
G3.	Ensuring reliability	G3.1 – Chapter 2.1.5.1.2 G3.2 – Chapter 2.1.5.1.2.1-2.1.5.1.2.3
G4.	Selection of materials and qualification of equipment	G4.1 – Chapter 2.1.5.1.7 G4.2 – Chapter 2.1.5.1.4F;2.1.5.1.7
08 Issue H:	Operational Limits and Conditions (OLCs) (10)	N/A
09 Issue I:	Ageing Management (11)	Chapter 2.1.8.1.3
I1.	Objective	I1.1 - Chapter 2.1.8.1.3A2
I2.	Technical requirements, methods and procedures	I2.1 – Chapter 2.1.8.1.3A1 I2.2 – Chapter 2.1.8.1.3C1 I2.3 – N/A PSR I2.4 – Chapter 2.1.8.1.3A I2.5 – N/A PSR
I3.	Major structures and components	I3.1 – Chapter 2.1.8.1.3A I3.2 – Chapter 2.1.8.1.3C1
10 Issue J:	System for Investigation of Events and Operational Experience Feedback (12)	Chapter 2.12.1.1.3. Chapter 2.1.2.4.1
11 Issue K:	Maintenance, In-Service Inspection and Functional Testing (13)	Chapter 2.7.1.3.5, 2.5 Chapter 2.10.7.2.2.3.5 and 2.10.7.2.2.3.6 Chapter 2.1.8.1.1
12 Issue LM:	Emergency Operating Procedures and Severe Accident Management Guidelines (14)	Chapter 2.10.6.4.2.3.11 Chapter 2.14.11.3 – 2.14.11.4
LM1.	Objectives	LM1.1 – Chapter 2.10.6.4.2.3.11

WENRA RHWG SRL for Existing Reactors		E.U.R.
LM2.	Scope	LM2.1 – LM2.2 – Chapter 2.14.11.3B and D LM2.3 – Chapter 2.14.11.4A LM2.4 – Chapter 2.14.11.3C LM2.5 – Chapter 2.14.11.4B LM2.6 – LM2.7 – Chapter 2.14.11.4C LM3.1 – Chapter 2.14.11.3D LM3.2 – Chapter 2.14.11.3B2 LM3.3 – Chapter 2.14.11.4C LM3.4 – Chapter 2.14.11.4G and G1 LM3.5 – Chapter 2.14.11.4B
LM3.	Format and Content of Procedures and Guidelines	N/A
LM4.	Verification and validation	N/A
LM5.	Review and updating	N/A
LM6.	Training and exercises	N/A
13 Issue N:	Contents and Updating of Safety Analysis Report (SAR) (15)	Not addressed
N1.	Objective	N/A
N2.	Content of the SAR	Not addressed
N3.	Review and update of the SAR	N/A
14 Issue O:	Probabilistic Safety Analysis (PSA) (16)	Chapter 2.17
O1.	Scope and content of PSA	O1.1 – Chapter 2.17.2.1 O1.2 – Chapter 2.17.3.6 O1.3 – Chapter 2.17.7.1 O1.4 – Chapter 2.17.3.7.4C.B; 2.17.3.8.1B; 2.17.3.8.3 O1.5 – Chapter 2.17.3.5
O2.	Quality of PSA	O2 – Chapter 2.17.8
O3.	Use of PSA	O3 – Chapter 2.17.1.1; 2.17.1.2; 2.17.1.3
O4.	Demands and conditions on the use of PSA	O4.1 – Chapter 2.17.8.2C O4.2 – N/A op. issue O4.3 – Chapter 2.17.6.3; 6.4
15 Issue P:	Periodic Safety Review (PSR) (17)	N/A
16 Issue Q:	Plant Modifications (17)	N/A
17 Issue R:	On-site Emergency Preparedness (18)	Chapter 2.14.3.
R1.	Objective	R1.1 – Chapter 2.14.3A

WENRA RHWG SRL for Existing Reactors		E.U.R.
R2.	Emergency Preparedness and Response Plan	N/A
R3.	Organization	N/A
R4.	Facilities and equipment	Chapter 2.14.3A-3D Chapter 2.1.7.4.7 and 2.1.7.4.8
R5.	Training, drills and exercises	N/A
18 Issue S:	Protection against Internal Fires (19)	Chapters 2.1; 2.4; 2.11; 2.17
S1.	Fire safety objectives	2.1.2.6.1
S2.	Basic design principles	S2.1 – Chapter 2.1.5.1.4C and C1 S2.2 – Chapter 2.11.2.4.2A S2.3 – Chapter 2.11.2.4.2C and Chapter 2.4.5.8.2.2C S2.4 – Chapter 2.1.8.3.3E S2.5 – Chapter 2.11.2.4.2F and Chapter 2.11.1.2.3.1
S3.	Fire hazard analysis	S3.1 – Chapter 2.11.2.4.2 A-R S3.2 – Chapter 2.4.5.8.2.2 S3.3 – Chapter 2.4.5.8.2.2 S3.4 – Chapter 2.17.5.1
S4.	Fire protection systems	S4.1 – Chapter 2.4.5.8.2.2 and Chapter 2.4.5.8.2.3 S4.2 – Chapter 2.4.5.8.2.2 and Chapter 2.1.6.7.1A1, 2.1.2.6.4I, S4.3 – Chapter 2.4.5.8.2.2A S4.4 – Chapter 2.1.7.6.5 and Chapter 2.8.4.1.5 S4.5 – Chapter 2.1.7.6.5
S5.	Administrative controls and maintenance	N/A
S6.	Fire fighting organization	N/A
19 Issue T:	Natural Hazards (20)	Chapter 2.1.2.6
T1.	Objective	T1.1 – Chapter 2.1.2.6A-E Chapter 2.4.1.1A-G
T2.	Identification of natural hazards	T2.1 – Chapter 2.4.1.2.10 Chapter 2.1.9.3.1 and 2.1.9.3.2 T2.2 – Chapter 2.1.2.6.2A1

WENRA RHWG SRL for Existing Reactors		E.U.R.
T3.	Site specific natural hazard screening and assessment	T3.1 – Chapter 2.1.2.6.2.2 Chapter 2.1.2.6.3 T3.2 – Chapter 2.1.2.6.3.1D T3.3 – Chapter 2.1.2.6.2.2 and 2.1.2.6.3.1H
T4.	Definition of the design basis events	T4.1 – Chapter 2.1.2.6.3.1A T4.2 – Chapter 2.1.2.6.3.1B,C T4.3 – Chapter 2.1.2.6.3.1A T4.4 – Chapter 2.1.2.6.3.1E
T5.	Protection against design basis events	T5.1 – Chapter 2.1.2.6.4A T5.2 – Chapter 2.1.2.6.4B T5.3a – Chapter 2.1.2.6.4C T5.3b – Chapter 2.1.5.1.3A1, sixth bullet T5.3c – Chapter 2.1.2.6.4A T5.3d – Chapter 2.1.2.5D T5.3e – Chapter 2.1.1.4 T5.3f – Chapter 2.1.6.3.2 Chapter 2.11.1.1.4 and Chapter 2.1.2.6B T5.3g – Chapter 2.11.1.1.4 T5.4h – Chapter 2.1.2.6A T5.4 – Chapter 2.1.2.6.4L T5.5 – Chapter 2.1.2.6.4O T5.6 – operational issue
T6.	Considerations for events more severe than the design basis events	T6.1 – Chapter 2.1.2.6.3A-I T6.2 – Chapter 2.1.9.3.1 T6.3 – Chapter 2.1.2.6.2.1A2; 2.1.2.6.4

(1) EUR do not address directly issuance, implementation and evaluation of safety policy by the utility. It however states in the EUR Chapter 2.15.1E that the overall Quality Assurance Programme (QAP) of the project as well as management processes and QAPs of contractors, and review organizations, shall be in accordance with the requirements of the nuclear specific safety standard IAEA GSR Part 2 »Leadership and Management for Safety«.

IAEA GSR Part 2 addresses Issuing and communicating safety policy, implementation of the safety policy and evaluation of the safety policy in its Requirements 3 and 4 (on the last account of evaluation it does not specify that evaluation shall be more frequent then the periodic safety review as the WENRA Issue A3 requires). It is however not within the scope of this benchmarking to determine the exact degree of compliance between the IAEA GSR Part 2 and WENRA Issue A.

Therefore, the EUR does not explicitly satisfy the WENRA RHWG Issue A on Safety Policy but requiring that the overall QAP shall be in full compliance with the IAEA GSR Part 2 in a way implicitly addresses the issue of safety policy. In short, there is no explicit compliance of EUR Requirements with the

WENRA RHWG Issue A, but there is an implicit way to identify it. As such it can be concluded that EUR are in **full compliance** with **WENRA Issue A.**

(2) EUR Chapter 2.14.2.2 requires that the Designer shall discuss the organization of plant personnel with the Owner and propose a model organization that reflects the design of the plant. It is a general statement which does not explicitly require justification and documentation of such organizational structure as do WENRA Issues B1.1 and B1.3 but it can be presumed that it is implicitly covered.

WENRA Issue B1.2 asks for the assessment of the impact of organizational changes on its impact on safety which is also not addressed in the EUR as it is not applicable in the design stage.

(3) WENRA Issue B2 addresses Management of safety and quality during operation within the operating organization and the EUR Chapters 2.14 and 2.15 which are referenced in the comparison tables refer to the same issues at the design stage in the design organization and therefore are not the same.

(4) EUR addresses obligations of the designer to provide training material and means to the operator and WENRA Issue B3 refers to the obligations of the operating organization. Overall EUR can be considered to be in **full compliance** with **WENRA Issue B.**

(5) For **WERA Issue C** refer to conclusions in the benchmark comparison of E.U.R. Chapter 15 with the IAEA Requirements. The same conclusions apply also here, bearing in mind that WENRA Issue C Reference Levels apply to an operating organization and E.U.R. Chapter 15 to a Quality Assurance programme in the design organization. EUR can be considered to be in **full compliance** with WENRA Issue C.

(6) On the issue of training, **WENRA Issue D** refers to obligations of the operating organization and the EUR Chapter 2.14.2.3 on the obligations of the designer to provide training means and methods to the operating organization. EUR can be considered to be in **full compliance** with WENRA Issue D.

(7) **WENRA Issue E**: Design basis envelope for existing reactors has been benchmarked with the EUR and **full compliance** has been identified in this area.

(8) Design extension conditions in EUR are composed of Complex sequences and Severe accidents.

Complex sequences are those which go beyond those in the design basis conditions in terms of failure of equipment and operator errors and have a potential to lead to significant releases, but do not involve core melt. This corresponds to DEC-A in WENRA Issue F (and IAEA standards)

Severe accidents are event sequences beyond design basis conditions which involve significant core melt and have a potential for large r/a releases. This corresponds to DEC- B in WENRA Issue F (and IAEA standards).

WENRA Issue F: Design basis envelope for existing reactors has been benchmarked with the EUR and **full compliance** has been identified in this area.

(9) WENRA Issue G2.1 states that the classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment. Such explicit statement is not found in the EUR section on safety classification of structure, systems and components but as this approach on using safety assessment methods applies throughout the design process it can be assumed that it applies also in the case of SSC classification and therefore **full compliance** can be established also for the **WENRA Issue G.**

(10) **WENRA Issue H** deals with Operational Limits and Conditions which are operational issues and not considered in such detail during the design stage. There are some basic principles that are established already in the design stage such as EUR requirement in Chapter 2.1.6.1 which specifies that the design shall be such that the plant can be operated safely within the operational limits and conditions (Issue H1.1) or EUR requirement on staffing in Chapter 2.14.2.1B which stipulates that the designer shall provide the operator with his assessment of personnel required to meet the operations and maintenance plan (Issue H8.1). Overall, the **WENRA Issue H can be identifies as N/A** as it relates to

operation of existing nuclear power plants and as such cannot be benchmarked with the design requirements.

(11) **WENRA Issue I** on Aging Management is an operational issue. The design aspects that have to be taken into account in the design stage are addressed in EUR 2.1.8.1.3. There are of course differences in both set of requirements as for example WENRA asks for operating organization to establish an Ageing Management Program (Issue I1.1), whereas EUR stipulates that the Aging Management Program has to be defined by the designer (EUR Chapter 2.1.8.1.3A2). Both can be taken as equivalent and this equivalence has been marked in the Tables. It can be concluded that WENRA Issue I for operation has been adequately addressed in the EUR requirements for design i.e. EUR are in **full compliance** with WENRA Issue I.

(12) Operational Experience Feedback as described in **WENRA Issue J** is an operational issue. EUR address the use of operational experience feedback in the design process in EUR Chapter 2.12.1.1.3. In addition, EUR Chapter 2.1.2.4.1 requires that the operational experience feedback be used to identify the additional safety features for design extension conditions. As in the case of WENRA Issue I, the operational FOE in WENRA document are matched with the FOE requirements in design as stipulated in the EUR document and it can be concluded that they are adequate. EUR are therefore in **full compliance** with WENRA Issue J.

(13) **WENRA Issue K** on Maintenance, In-Service Inspection and Functional Testing is an operational issue. EUR address those issues for the design stage in EUR Chapter 2.7.1.3.5 where it requires the plant designer shall provide the operator with the reliability data needed to establish a reliability program inspection, maintenance and in-service testing of essential equipment. Further in EUR Chapter 2.7.2.5 EUR stipulates that appropriate design factors shall be followed by the designer to ensure the maintainability of the RCS components and In-service inspection capability. Further, EUR Chapter 2.1.8.1.1 gives requirements for design of items important to safety for inspection, on-line monitoring, testing and maintenance. For Instrumentation and Control design requirements address testability in EUR Chapter 2.10.7.2.2.3.5. and for maintainability in EUR Chapter 2.10.7.2.2.3.6. As for other WENRA operational issues it can be concluded that EUR design requirements are adequate for the purpose that they have been developed for. Here again it can be concluded that EUR are in **full compliance** with WENRA Issue K.

(14) **WENRA Issue LM** on EOPs and SAMGs is an operational issue. EUR nevertheless cover the development of EOPs and SAMGs during design in sufficient detail that they can be compared with WENRA issues in operation. For sub-issue LM2.5 which explicitly requires that there be a set of procedures and guidelines suitable to manage accident conditions that simultaneously affect the reactor and spent fuel storage, there is no direct mention of both but EUR Chapter 2.14.11.4B does make reference to all physically identifiable situations which could also be seen as covering the simultaneous effect on the reactor and the spent fuel storage. Format and content of procedures and guidelines as well as verification and validation, review and updating and training and exercises are typical operational issues and cannot be covered to such detail in the design stage. EUR are therefore in **full compliance** with WENRA Issue LM.

(15) EUR Chapter 2.12.14 lists the Safety Analyses Report in Table 1 which provides the minimum list of documentation to be provided by the designer to the operator. Detailed content and format of the safety analyses report are not addressed in the EUR document as they are in the WENRA Issue N2. WENRA Issues N1.1 and N1.2 define that the licensee shall provide SAR and use it for assessing the safety implications of changes and therefore these two issues have been identified as N/A. The same is true for the issue N3.1 that deals with the review and update of the SAR. EUR are in **full compliance** with **WENRA Issue N.**

(16) Requirements on PSA in both documents WENRA and EUR have been found to be equivalent in spite of the fact that the WENRA issues have been written for plants in operation and the EUR for the design stage. It can be concluded that EUR Chapter 2.17 on PSA methodology is in **full compliance** with the **WENRA Issue O**.

(17) **WENRA Issue P** on Periodic Safety Review and **WENRA Issue Q** on Plant Modifications are operational issues and as such outside the scope of this benchmarking.

(18) **WENRA Issue R**: On-site emergency preparedness arrangements are responsibility of the operating organization. EUR document in Chapter 2.14.3 defines in 3A1 - 3A4 the need that in addition to the Main Control Room and Emergency Control Room there should also be separate Emergency Response Facility which comprises of Technical Support Centre, the Operational Support Centre and the Emergency Preparedness Centre (also as per IAEA GSR Part 7). Further EUR in Chapter 2.14.3A5 states that further functionalities of the Emergency Response facilities shall be defined by the operating organization. Therefore the Emergency Plan, Organization, Training, Drills and Exercises are operational issues and not covered by the EUR. It can be concluded that EUR emergency preparedness aspect that need to be addressed in the design are in **full compliance** with the WENRA Issue R.

(19) In safety objectives for protection against internal fires **WENRA Issue S1.1** requires that the licensee shall implement fire protection measures, whereas the same safety objective exist also in the EUR document in Chapter 2.1.2.6.1 where it is the design that shall take due account of internal hazards, such as fires…..It is seen as the same safety objective for the purpose of this benchmarking study. Basic safety objectives, basic design principles, fire hazard analysis and fire protection systems are in **full compliance.** The administrative control and fire-fighting organization are operational issue and outside the scope of this benchmarking.

(20) Site-specific evaluation of hazards is not within the scope of the EUR design assessment process. Nevertheless, general hazard assessment is well documented in the EUR Chapters 2.1 and 2.4. WENRA document prescribes also a number of operational requirements which are not considered in the EUR document. In general it can be concluded that Natural Hazards as covered in the EUR document are **in full compliance** with the **WENRA Issue T**.

CONCLUSIONS

Even though the WENRA RHWG Reference Levels for Existing Reactors address the safety of nuclear power plants in operation and E.U.R. have been anticipated mostly as requirements to be fulfilled in the design stage some comparison could be made and the overall result of this part of Benchmarking is that **E.U.R. are in full compliance with WENRA Safety Reference Levels for Existing Reactors,** wherever comparison could be established.

6 BENCHMARKING EUR AGAINST WENRA RHWG REPORT: SAFETY OF NEW NPP DESIGNS

One of the objectives of WENRA was to harmonize approaches to nuclear safety and radiation protection in their member countries. In addition to the publication of WENRA Reference Levels (for the benchmarking results see Chapter 6 of this report), they published in 2010 a WENRA Statement on safety objectives for new nuclear power plants which included seven safety objectives, which are the basis for further harmonization. These **seven safety objectives** are the following:

- O1. Normal operation, abnormal events and prevention of accidents
- O2. Accidents without core melt
- O3. Accidents with core melt
- O4. Independence between all levels of Defence-in-Depth
- O5. Safety and security interface
- O6. Radiation protection and waste management
- O7. Leadership and management for safety

The above safety objectives are formulated in such a way as to drive design enhancements for new plants with the aim of reaching a higher level of safety than that expected from existing plants. Most notably it is required for new plants to extend the safety demonstration to some situations that are considered as "beyond design basis" for existing plants. These are the design extension conditions (DEC) that need to be considered in the design of new plants and are in the WENRA document termed as "multiple failure conditions" (equivalent to DEC-A in other documents) and "core melt accidents" (DEC-B in other documents).

Based on the above seven safety objectives, WENRA developed **common positions on selected key safety issues** for the design of new nuclear power plants. These are:

- O3.1 Position 1: Defence-in-depth approach for new nuclear power plants
- O3.2 Position 2: Independence of the levels of Defence-in-depth
- O3.3 Position 3: Multiple failure events
- O3.4 Position 4: Provisions to mitigate core melt and radiological consequences
- O3.5 Position 5: Practical elimination
- O3.6 Position 6: External hazards
- O3.7 Position 7: Intentional crash of a commercial airplane

As these common positions were formulated before the Fukushima Daiichi accident, they have been updated following the lessons learned from the accident. Lessons learned from the Fukushima Daiichi were captured in the following subject areas:

- O4.1 External hazards (enhancing position 6)
- O4.2 Reliability of safety functions (enhancing positions 1, 2, 3 and 6)
- O4.3 Accidents with core melt (enhancing positions 2, 4, and 5)
- O4.4 Spent Fuel Pools (enhancing positions 1, 3, 5 and 6)
- O4.5 Safety assessment
- O4.6 Emergency preparedness in design

European Utility Requirements were benchmarked to the above safety objectives and common positions and the results are presented in the below table.

WENRA Safety of new NPP designs		EUR
01	**Introduction**	
02	**WENRA safety objectives for new nuclear power plants** (1)	Chapter 2.1.1.1 General safety objectives Chapter 2.1.1.4A.A Table
	O1. Normal operation, abnormal events and prevention of accidents - Reducing the frequency of abnormal events by enhancing plant capability to stay within normal operation. - Reducing the potential for escalation to accident situations by enhancing plant capabilities to control abnormal events.	Chapter 2.1.1.4C Chapter 2.1.1.4E
	O2. Accidents without core melt - No off-site radiological impact - Reducing as far as reasonably achievable core damage frequency and the release of r/a material - Providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts	Chapter 2.1.3.3A (quotes consistency with WENRA Objective O2) Chapter 2.1.3.5A.A and A.B1 Chapter 2.1.9.3.1 and 2.4.2.4.1 hazards Chapter 2.1.8.3.1D malevolent acts
	O3. Accidents with core melt - Reducing potential r/a releases to the environment from accidents with core melt, also in the long term, by following the qualitative criteria	Chapter 2.1.3.4A1 (quote consistency with WENRA Objective O3)
	O4. Independence between all levels of Defence-in-Depth - enhancing the effectiveness of the independence between all levels of defence-in-depth	Chapter 2.1.1.4D, D1 Chapter 2.1.6.3.2.1 Chapter 2.10.6.4.2.2.1 for I&C
	O5. Safety and security interfaces (2) - Ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.	Chapter 2.1.8.3 Chapter 2.11.6 Chapter 2.14.10.3D1 Chapter 2.14.12.2 Table 4 Chapter 2.10.6.5.3 and 2.10.7.3.3 I&C security

WENRA Safety of new NPP designs		EUR
	O6. Radiation protection and waste management [3] - Reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning, and dismantling activities: o Individual and collective doses for workers o r/a discharges to the environment o quantity and activity of r/a waste	Chapter 2.1.1.1 Chapter 2.1.1.4A.A Table Chapter 2.1.3.2; 2.1.3.3; 2.1.3.4 Chapter 2.1.7.9.1A and 2.1.7.9.2A Chapter 2.16.3B
	O7. Leadership and management for safety [4] - ensuring effective management for safety from the design stage. This implies that the licensee: o establishes effective leadership and management for safety over the entire plant project and…… o ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of new plants demonstrate	Chapter 2.15
03	**Selected key safety issues**	
03.1	Position 1: Defence-in-depth approach for new nuclear power plants [5]	Chapter 2.1.1.4 Chapter 2.10.6.2.2.1 for I&C
	Rationale for an evolution of DiD levels	Chapter 2.1.1.4
	Refined structure of the levels of DiD	Chapter 2.1.1.4A.A Table
03.2	Position 2: Independence of the levels of Defence-in-depth [6]	Chapter 2.1.1.4D;D1 Chapter 2.1.6.3.2.1 Chapter 2.10.6.4.2.2.1 for I&C
	Emergency AC power supply	Chapter 2.1.7.5 Chapter 2.7.13
	Separation of cables	Chapter 2.1.1.4D2

WENRA Safety of new NPP designs		EUR
	Reactor protection system (RPS) and other I&C aspects	Chapter 2.1.7.4.3. – 7.4.5
	Containment	Chapter 2.1.7.3
	Reactor pressure vessel	Chapter 2.7.2.2 and 2.7.2.3B
03.3	Position 3: Multiple failure events (7)	Chapter 2.1.2.4.1 Chapter 2.1.9.2 and 2.1.9.4
03.4	Position 4: Provisions to mitigate core melt and radio-logical consequences (8)	Chapter 2.1.2.4.2 and 2.1.2.4.3 Chapter 2.1.9.2
03.5	Position 5: Practical elimination (9)	Chapter 2.1.2.5
	Demonstration of Practical Elimination via Physical Impossibility	Chapter 2.1.2.5D
	Demonstration of Practical Elimination as extremely unlikely with a high degree of confidence	Chapter 2.1.2.5E
03.6	Position 6: External hazards (10)	Chapter 2.1.2.6.2 Chapter 2.1.9.3 List of Hazards Chapter 2.4.1.2 and 2.4.1.3
03.7	Position 7: Intentional crash of a commercial airplane (11)	Chapter 2.1.8.3.3 Chapter 2.4.1.3.3.2
04	**Lessons Learnt from the Fukushima Dai-ichi accident**	
04.1	External hazards	Enhancement of Position 6
04.2	Reliability of safety functions	Enhancement of Positions 1, 2, 3 and 6
04.3	Accidents with core melt	Enhancement of Positions 2, 4, and 5
04.4	Spent Fuel Pools	Enhancement of Positions 1, 3, 5 and 6
04.5	Safety assessment (12)	Chapter 2.11.1.1.4 Chapter 2.1.6.6
04.6	Emergency preparedness in design (13)	Chapter 2.14.3 Chapter 2.1.7.4.6 – 2.1.7.4.8
Annex 1	**WENRA Statement on Safety Objectives for New Nuclear Power**	Covered above in O2

(1) Seven safety objectives are described in slightly more detail in the Annex 1 of the WENRA Report on Safety of new NPPs design and therefore the benchmarking is performed using the description presented in Annex 1.

(2) Security aspects are addressed on numerous places in the EUR document. Safety and security interface is implicitly included in all those requirements but explicit mention of safety and security interface is mentioned only in 2.14.12.2 Table 4 where “Rapid Operator Response and Security Interface” is listed among LWR operation issues to be addressed in design.

(3) EUR document specifies that the radiation protection in the design stage is implemented by observing ICRP recommendations during operation, maintenance and decommissioning processes (EUR Chapter 1.2. 5.).

(4) As already stated under benchmarking of EUR with the IAEA Safety Standards the EUR document does not address Leadership and Management for safety as defined in the IAEA GSR Part 2 standard or the WENRA Safety Objective O7, but rather relies on the overall quality assurance programme as described in the EUR Chapter 2.15. The same conclusions as in the benchmarking of EUR with the IAEA Safety Standards apply also here.

(5) Refined structure of the levels of defence-in-depth is included in the IAEA Safety Standards, WENRA Safety of new NPP designs and EUR document. The table below shows that all standards have the same breakdown of the level 3 of defence-in-depth but they all use different terms to describe the basic concept which is equivalent in all three documents. The table demonstrates **full compliance** of EUR with Position O3.1 of the WENRA document:

Level of defence-in-depth	E.U.R.	IAEA	WENRA Safety of new NPP designs
Level 3a	Design Basis Accidents	Design Basis Accidents	Postulated single initiating events
Level 3b	DEC – Complex sequences	DEC-A without core melt	Postulated multiple failure events
Level 4	DEC – Severe accidents	DEC-B with core melt	Postulated core melt accidents

(6) Independence of the levels of Defence-in-Depth as covered in the EUR document is in **full compliance** with principles described in Position 03.2 of the WENRA document.

(7) Multiple failure events in Position 3 in the WENRA document are equivalent to Complex Sequences in Chapter 2.1.2.4.1 of the EUR document. Both documents recognize the need for inclusion of this part of the design extension conditions at the design stage and the recognition that the safety features used in this part of DEC shall be, as far as reasonable practicable, independent of safety systems (EUR 2.1.2.4.1E). Further both documents recognize the need for inclusion of the spent fuel pool in addition to NPP operational modes (EUR 2.1.2.4.1A1). The lists of examples of multiple failure scenarios in the WENRA document (O3.3) and those for DEC complex sequences in EUR document (Chapter 2.1.9.4) are almost identical. The EUR document includes in addition also the Containment bypass accidents without core melt that include: Main steam line break with consequential steam generator tube rupture (MSLB+SGTR) and Interfacing system LOCAs outside containment boundary. Therefore, EUR document is in **full compliance** with principles described in Position O3.3 of the WENRA document.

(8) Accidents with core melt in Position 4 in the WENRA document are equivalent to Severe Accidents in Chapter 2.1.2.4.2 of the EUR document. Both documents require that all sequences with a potential for early and large releases be practically eliminated (2.1.2.4.2A2) and that for those sequences that have not been practically eliminated, only limited protective measures in area and time are needed – EUR 2.1.9.2 Table 3 (no permanent relocation or emergency evacuation apart from the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption – and sufficient time to implement these actions). Similar to the case of multiple failure events i.e. complex sequences also in the case of accidents with core melt, the safety features for DEC used in severe accidents (accidents with core melt) shall be, as far as reasonably practicable, independent from safety systems and safety features for DEC used in complex sequences (multiple failure events). Exception can be the use of alternate power source (EUR 2.1.2.4.2E). Requirements on the containment performance in the case of accidents with core melt

as stipulated in Position 4 of the WENRA document (reliability, redundancy, strength, heat removal, venting, filtration, penetrations etc.) are adequately dealt in the EUR Chapter 2.9 on Containment. Both documents also require that deterministic and probabilistic safety analysis be used. WENRA interpretation of limited protective measures in case of accidents with core melt is covered in full by the EUR in Chapter 2.1.3.4A1. The EUR document is in **full compliance** with principles described in Position O3.4 of the WENRA document.

(9) All accident sequences that have a potential for early or large release shall be practically eliminated are the requirement in both documents. Both documents follow the definition of practical elimination from the IAEA Safety Standards as well as definition of early and large releases. Practical elimination of sequences practically means that the mitigation of their consequences does not need to be included in the design. Both documents also follow the IAEA explanation that practical elimination is achieved if it is physically impossible for the accident sequence to occur or it is extremely unlikely to arise and that it is preferred that practical elimination is demonstrated by the physical impossibility. Also, both documents require deterministic and probabilistic analysis to be used together with the engineering judgement. EUR document lists the phenomena to be practically eliminated (2.1.2.5B) which is in compliance with the listed phenomena in the WENRA Position 5. EUR Chapter 2.1.2.5 on Practical elimination is in **full compliance** with principles described in Position 03.5 of the WENRA document.

(10) EUR Chapter 2.1.2.6A1 lists among other safety documents also the WENRA Report on Safety of new NPP designs for detailed consideration of external hazards. Both documents address natural and man-made external hazards (EUR 2.1.2.6.2A1). Both documents distinguish between Design Basis External Hazards (DBEH – EUR 2.1.2.6.2.1A1) and Rare and Severe External Hazards (RSEH – EUR 2.1.2.6.2.1A2). The safety demonstration as specified in the WENRA document and involves four stages: identification, screening, determination of hazard parameters and analysis is adequately covered in EUR 2.4.1.2 and 2.4.1.3. The lists of natural and man-made hazards in the EUR document have been taken from the WENRA document and referenced accordingly. External Hazards in EUR provide far more details on specific values to be considered than the WENRA document and as such are in **full compliance** with the Position O3.6 of the WENRA document.

(11) The intentional aircraft crash is defined in EUR 2.1.8.3.3 and the load/time curves in EUR 2.4.1.3.3.2. In both documents the expectation is that he intentional crash of a commercial plane should not lead into a core melt. The sensitive parts of the installation are identified in WENRA document and in EUR 2.1.8.3.3C1 identically. Also, the direct and indirect effects of the airplane crash are identically identified in WENRA and EUR 2.1.8.3.3D and 2.1.8.3.3.E sections. EUR document is in **full compliance** with the WENRA Position O3.7 on Intentional crash of a commercial plane.

(12) Safety Assessment enhancements following the lessons learned from the Fukushima Daiichi accident (O4.5) relate primarily to the Periodic Safety Review which is the operating organization responsibility and therefore outside the scope of this benchmarking study. The inclusion of interaction between different units on a multi-unit site into the safety assessment and identification of hazards on the multi-unit side which are also new requirements based on lessons learned from the Fukushima Daiichi accident are **fully and adequately covered** in EUR Chapters 2.1.6.6 and 2.11.1.1.4.

(13) Emergency preparedness in design is fully covered in EUR Chapters 2.14.3 and 2.1.7.4.6 – 2.1.7.4.8 where all necessary arrangements in design for the Main Control Room, Emergency Control Room, Technical Support Centre, Operational Support Centre and Emergency Preparedness Centre are identified in EUR Chapter 2.14.3A1 – A4 (also fully in line with the IAEA GSR Part 7 Requirements). Protection against rare and severe external hazards is also addressed (EUR Chapter 2.14.3B). Reliability and functionality of the on-site and off-site communication systems is also fully addressed in EUR Chapter 2.14.3D as required by O4.6 in the WENRA document. Emergency preparedness in design (O4.6) is **fully and adequately** covered in EUR Chapters 2.14.3 and 2.1.7.4.6 – 2.1.7.4.8.

CONCLUSIONS

All WENRA positions including defence-in-depth, independence of the levels of defence-in-depth, multiple failure events, provisions to mitigate core melt and radiological consequences, need for the practical elimination, treatment of external hazards and the issue of the intentional aircraft crash of a commercial airplane are fully covered in the EUR document and therefore it can be concluded that EUR document is in **full compliance** with the WENRA Report on Safety of new NPP designs.

7 BENCHMARKING E.U.R. AGAINST THE COUNCIL DIRECTIVE 2014/87/EURATOM

The EU has the most advanced legally binding and enforceable framework for nuclear safety which is of utmost importance for the industry development and for protection of people and the environment. The Council Directive 2014/87/Euratom of 8 July 2014 which brings the amendment to the Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations (commonly referred to as the **amended Nuclear Safety Directive)** brings these standards to even a higher level. The original Council Directive 2009/71/Euratom imposes obligations of the Member States to establish and maintain a national framework for nuclear safety and reflects the provisions of the main international instruments in the field of nuclear safety, namely the Convention on Nuclear Safety, as well as the IAEA Safety Fundamentals. After the Fukushima Daiichi accident “stress tests” were performed community-wide but in addition, the European Council also called on the Commission to review, as appropriate, the existing legal and regulatory framework for the safety of nuclear installations and propose any improvements that may be necessary. The European Council also stressed that the highest nuclear safety standards should be implemented and continuously improved in the European Union. As a consequence, amended Nuclear Safety Directive has been prepared and entered into force in July 2014.

Under its *Section 1* on *General obligations*, it stresses that a strong, competent regulatory authority with effective independence in regulatory decision making is a fundamental requirement of the nuclear safety regulatory framework. The regulatory body should be able to carry its duties impartially, transparently and free from undue influence in its regulatory decision making. Therefore the initial provisions set in Directive 2009/71/Euratom on establishing the effective regulatory framework and on functional separation of competent regulatory authorities has been strengthened in the amended Nuclear safety Directive by adding in:

- Article 4(1) the following; “Member States shall establish and maintain a national, legislative, regulatory and organizational framework *(“national framework”)* for the nuclear safety of nuclear installations. The national framework shall provide in particular for:

    a. The allocation of responsibilities and coordination between relevant state bodies;

    b. National nuclear safety requirements, covering all stages of the lifecycle of nuclear installations;

    c. A system of licensing and prohibition of operation of nuclear installations without a licence;

    d. A system of regulatory control of nuclear safety performed by the competent regulatory authority;

    e. Effective and proportionate enforcement actions, including, where appropriate, corrective actions or suspension of operation and modification or revocation of a licence.”

As for the effective independence of the regulatory authority, the Article 5 has been amended to read as follows:

- “Member States shall ensure the *effective independence* from undue influence of the competent regulatory authority in its regulatory decision making. For this purpose, Member States shall ensure that the national framework requires that the competent regulatory authority:

    a. Is functionally separate from any other body or organization concerned with the promotion or utilization of nuclear energy, and does not seek or take instructions from any such body or organization when carrying out its regulatory tasks;

b. Takes regulatory decisions founded on robust and transparent nuclear safety-related requirements;
c. Is given dedicated and appropriate budget allocations to allow for the delivery of its regulatory tasks as defined in the national framework and is responsible for the implementation of the allocated budget;
d. Employs an appropriate number of staff with qualifications, experience and expertise necessary to fulfil its obligations. It may use external scientific and technical resources and expertise in support of its regulatory functions;
e. Establishes procedures for the prevention and resolution of any conflicts of interest;
f. Provides nuclear safety-related information without clearance from any other body or organization, provided that this does not jeopardise other overriding interests, such as security, recognized in relevant legislation or international instruments."

Further it is stipulated that the competent regulatory authority is given the *legal powers* necessary to fulfil its obligations under the national framework. To assure this, the competent regulatory authority shall be entrusted

- "with the following main regulatory tasks, to:
  a. Propose, define or participate in the definition of national nuclear safety requirements;
  b. Require that the licence holder complies and demonstrates compliance with national nuclear safety requirements and the terms of the relevant licence;
  c. Verify such compliance through regulatory assessments and inspections;
  d. Propose or carry out effective and proportionate enforcement actions."

Further, the amended Nuclear Safety Directive re-emphasizes that the prime responsibility for nuclear safety rests with the licence holder and that this responsibility cannot be delegated. It includes also the responsibility for the activities of contractors and sub-contractors whose activities might affect the nuclear safety.

Section 1 on General obligations also covers amendments that relate to Expertise and skills in nuclear safety as well as to Transparency.

Changes to Section 1 have been described in detail above as they relate mostly to the regulatory body issues and as such are outside the scope of the European Utility Requirements and therefore also of this benchmarking study. They are however an important amendment to the previous directive and as such needs to be documented.

After the Article 8 that covers Transparency a whole new *Section 2* is inserted that describes the *Specific obligations.* It starts with the article that defines the *nuclear safety objective for nuclear installations.* It requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective to prevent accidents and, should an accident occur, mitigate its consequences and avoid early radioactive releases that would require off-site emergency measures but with insufficient time to implement them or large radioactive releases that would require protective measures that could not be limited in area or time.

In order to achieve this nuclear safety objective, the defence-in-depth is applied to ensure that the impact of extreme external hazards is minimised, abnormal operation and failures are prevented, abnormal operation is controlled and failures are detected, accidents within the design basis are controlled, severe conditions are controlled, including prevention of accidents progression and mitigation of the consequences of severe accidents.

Article on *Initial assessment and periodic safety review* stipulates the need for the initial installation specific assessment that results in a nuclear safety demonstration with respect to national regulatory requirements. Subsequently a periodic safety review to re-assess the safety of the nuclear installation is required at least every 10 years. Such re-assessment should take into account ageing issues, operational experience, most recent research results and developments in international standards.

In addition, it introduces the requirement for a European system of peer reviews, with specific safety issues to be reviewed every 6 years.

An important part of the amendment refers to on-site emergency preparedness and response. Member States must ensure that the national framework requires that an organizational structure for on-site emergency preparedness and response is established with clear allocation of responsibilities and coordination between licence holders and the competent authorities and organizations for all phases of an emergency.

Member States were required to bring into force the laws, regulations and administrative provisions necessary to comply with the above by 15 August 2017.

Amended NSD - COUNCIL DIRECTIVE 2014/87/EURATOM		**European Utility Requirements**
Chapter 1 OBJECTIVES, SCOPE AND DEFINITIONS		
Article 1	Objectives (1)	Chapter 1.1.2
Article 2	Scope (2)	Chapter 1.1.4
Article 3	Definitions (3)	Vol. 1 Appendix B
Chapter 2 OBLIGATIONS		
SECTION 1: General obligations		
Article 4	Legislative, regulatory and organizational framework (4)	Chapter 1.1.7
Article 5	Competent regulatory authority	N/A (see comment under (4))
Article 6	Licence holder	
(a)	The prime responsibility for nuclear safety (5)	N/A
(b)	When applying for a licence, the applicant is required to submit a demonstration of nuclear safety	N/A (see comment under (5))
(c)	Licence holders are to regularly assess, verify, and continuously improve, nuclear safety of their nuclear installations (6)	Chapter 2.18.5.1A Chapter 2.1.1.4A
(d)	Licence holder establishes and implement management system (7)	Chapter 2.15
(e)	Licence holders provide for appropriate on-site emergency procedures (EOPs) and arrangements, including severe accident management guidelines (SAMGs) (8)	Chapter 2.10.6.4.2.3.11 Chapter 2.14.3 Chapter 2.14.11.3 – 2.14.11.4
(f)	Licence holders provide for and maintain financial and human resources with appropriate qualifications and competences (9)	Chapter 2.14.2.3
Article 7	Expertise and skills in nuclear safety (9)	Chapter 2.14.2.3
Article 8	Transparency	Chapter 1.2.2 last bullet

Amended NSD - COUNCIL DIRECTIVE 2014/87/EURATOM		European Utility Requirements
SECTION 2 Specific obligations		
Article 8a	Nuclear safety objective for nuclear installations	
8a 1 (a) and (b)	Avoiding early r/a releases and large r/a releases (10)	Chapter 2.1.2.5 Chapter 2.1.3.5 Chapter 2.1.5.1.7.1.2
8a 2 (a) and (b)	Application of objectives under 8(a)-1	N/A
Article 8b	Implementation of the nuclear safety objective for nuclear installations	
8b 1 (a)	Impact of extreme external natural and unintended man-made hazards are minimized (11)	Chapter 2.1.2.6.2 – 2.1.2.6.4 Chapter 2.1.9.3
8b 1 (b)	Abnormal operation and failures are prevented (12)	Chapter 2.1.1.4A.A
8b 1 (c)	Abnormal operation is controlled and failures are detected (13)	Chapter 2.1.1.4A.A
8b 1 (d)	Accident within the design basis are controlled (14)	Chapter 2.1.1.4A.A
8b 1 (e)	Control of severe accidents (15)	Chapter 2.1.1.4A.A
8b 1 (f)	Organizational structure for emergency preparedness and response (16)	Chapter 2.1.1.4A.A
8b 2 (a)	Management system (17)	Chapter 2.15
8b 2 (b)	Operating experience (18)	Chapter 2.1.8.2
8b 2 (c)	Obligation of the licence holder to report events to the regulatory body	N/A
8b 2 (d)	Arrangements for education and training	Chapter 2.14.2.3
Article 8c	Initial assessment and periodic safety review	
8c (a)	Initial assessment in accordance with Article 8a	Chapter 2.1.2.5 Chapter 2.1.3.5 Chapter 2.1.5.1.7.1.2
8c (b)	Periodic safety review	N/A
Article 8d	On-site emergency preparedness and response (19)	Chapter 2.14.3 Chapter 2.1.7.4.8 Chapter 2.1.1.4A.A
8d 1.	Organizational structure for on-site EPR	N/A
8d 2.	Consistency and continuity between on-site EPR and other EPR arrangements	N/A
Chapter 2a PEER REVIEWS AND REPORTING		N/A
Article 8e	Peer reviews	N/A

Amended NSD - COUNCIL DIRECTIVE 2014/87/EURATOM		European Utility Requirements
Article 9	Reporting to the Commission	N/A
Article 10	MS without nuclear installations	N/A

(1) Amended NSD and EUR have the same overall objective and that is to contribute to the enhancement of nuclear safety in the European Member States. However, the particular objectives i.e. the contribution towards the enhancement of nuclear safety in Europe are different. The amended NSD puts obligations on Member States (i.e. Governments) to establish a framework for improved nuclear safety to protect workers and the general public against the dangers arising from ionizing radiations from nuclear installations. The European Utility Requirements on the other hand aim to harmonize the requirements to which the Light Water Reactor (PER and BWR) nuclear power plants to be built in Europe will be designed, built, commissioned, operated and maintained and is therefore targeted mostly towards design organizations.

(2) Scope of the amended NSD is the same as the scope of the EUR and both are applicable to civilian nuclear installations. EUR however emphasizes its applicability to GEN III NPPs. EUR are applicable to PWR and BWR reactors and it is emphasized that other types of reactors do not have sufficient operating experience to be built, licenced and operated in Europe in the near future.

(3) Amended NSD has a very limited number of definitions in its Article 3. EUR on the other hand has a very extensive list of definitions which are used in the document and are collected in the Appendix B of Volume 1. No definitions in both documents are in contradiction.

(4) Amended NSD requires in its Article 4 that a sound regulatory regime be established in each Member State. Such requirement is beyond the scope of the EURs; however EUR Section 1.1.7 does address the necessary interfaces among different stakeholders including national regulators.

(5) The fact that the prime responsibility for the nuclear safety of a nuclear installation rests with the licence holder and cannot be delegated is a universal requirement and as such not explicitly mentioned in the EUR document as it is an obvious requirement. The same applies to the need for a safety demonstration in the form of a Safety Analysis Report when applying for the licence. Safety demonstration is being requested throughout the EUR Chapter 2.1 for particular items (like practical elimination, hazard assessment, etc.) and it is presumed that the need for the Safety Analysis Report (Preliminary and Final) is set as a requirement in other documents dealing with licensing.

(6) Article 6(c) of the amended NSD requires license holders to assure that defence-in-depth is assured and that accidents are prevented and consequences mitigated should they occur. EUR puts the same requirements on the design and therefore they can be seen as equivalent.

(7) Again, Article 6(d) of the amended NSD puts requirement for management system on the licence holder. The same requirement is placed also on the “owner/operator” that is the company that owns the plant but in terms of the overall quality assurance programme. The difference between the management system and the overall quality assurance programme is discussed already in the benchmarking of EUR with the IAEA Safety Standards and the same conclusions apply also here.

(8) Like the rest of article 6 of the amended NSD also this part addresses licence holder obligations. EOPs and SAMGs are however adequately addressed also in EUR document.

(9) EUR Chapter 2.14.2.3 requires the designer to provide the operator with adequate and systematic training material. The same covers also the Article 7 of the amended NSD.

(10) Amended NSD requires Member States to assure the avoidance of early and large radioactive releases. This issue is extremely well covered in the EUR document where it is placed already on the design of nuclear installations.

(11) Extreme external natural and unintended man-made hazards are fully covered in the design requirements in the EUR document.

(12) Prevention of abnormal operation and failures is Level 1 of defence-in-depth and as such fully covered in the EUR document.

(13) Control of abnormal operation and detection of failures is Level 2 of defence-in-depth and as such fully covered in the EUR document.

(14) Control of accidents within the design basis is Level 3a of defence-in-depth and as such fully covered in the EUR document.

(15) Control of severe accidents is Level 3b and Level 4 of defence-in-depth and as such fully covered in the EUR document.

(16) On-site and off-site emergency preparedness and response is Level 5 of defence-in-depth and as such fully covered in the EUR document.

(17) The same observation as under footnote (7)

(18) Importance of the documented feedback of operating experience is recognized under human factors already at the design stage and adequately addressed in the EUR.

(19) EUR describes the provisions made during the design for the on-site emergency response facilities and it is outside the scope of the EPR to discuss organizational structure of the EPR organization and its interface with other emergency preparedness and response arrangements. EUR document addresses hardware requirements for the on-site EPR arrangements adequately and in full.

CONCLUSIONS:

The amended Nuclear Safety Directive brings the nuclear safety standards to a higher level by setting the objective to reduce the risk of accidents and avoid large radioactive releases. The compliance to this objective by the European Utility Requirements (EUR) has been benchmarked in this study. The second objective of the amended Nuclear Safety Directive calling for periodic safety reviews every 6 years is out of the scope of the EUR document and therefore also out of the scope of this benchmarking study.

Initial paragraphs of the amended Nuclear safety Directive relate mostly to the regulatory body issues and as such are outside the scope of the European Utility Requirements and therefore also of this benchmarking study. The same applies for the organizational structure of the EPR organization and its interface with other emergency preparedness and response arrangements.

For the aspects that relate in both the amended Nuclear Safety Directive and E.U.R. a **full compliance** has been established during this benchmarking study.

8 BENCHMARKING E.U.R. AGAINST THE PROVISIONS OF THE VIENNA DECLARATION ON NUCLEAR SAFETY

Following the efforts and initiatives to strengthen the review process under the Convention on Nuclear Safety (CNS) and efforts and initiatives that took place nationally, regionally and internationally after the Fukushima Daiichi accident, the Contracting Parties met at the Diplomatic Conference to amend the CNS in February 2015. The initiative for the amendment of CNS came from the Swiss Confederation with particular emphasis on the amendment of Article 18 of the CNS. The initiative was presented in order to bring the nuclear safety standards in other regions to the same high level as prescribed to the EU countries through the amended NSD. It would assure that high safety standards are applied worldwide and that these are not undermined by the use of cheaper or outdated technology.

The Diplomatic Conference of the Convention on Nuclear Safety at the end adopted the so-called Vienna Declaration on the 9th of February 2015. It has only 3 Articles. The first one addresses new nuclear power plants, the second one the existing nuclear power plants and the third the obligation to follow the IAEA Safety Standards.

For this benchmarking exercise only the first article is of interest as it addresses new builds. It stipulates that "*new nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions*".

E.U.R. document addresses these issues throughout the document and in particular in Chapter 2.1.1.1D on General safety objectives and Chapter 2.1.3.4 on Safety objectives and off-site release targets for accidents with core melt. It can be therefore concluded that E.U.R. are **in full compliance** with the Vienna Declaration on Nuclear Safety.

9 COMPARISON OF SELECTED, MOST IMPORTANT ISSUES

The work on the **European Utility Requirements** started in 1991 when major electricity producers joint effort to harmonize national efforts in the development, design and licensing of Light Water Reactor plants. The first Revision A was published in 1994, Revision B in 1995, Revision C in 2001, Revision D in 2012 and the last Revision E in 2016. The last Revision E, whose development started only 2 years after the publication of the previous Revision D, was initiated in order to capture the lessons learned from the Fukushima Daiichi accident. The aim of the European Utility Requirements was to promote harmonization of requirements for Generation III Nuclear Power Plants (NPPs) across Europe (and worldwide).

The aim was to harmonize the requirements to which Light Water Reactor NPP to be built in Europe will be designed, built, commissioned, operated and maintained. At the same time, the objective of Revision E was to incorporate all international requirements that have been developed based on lessons learned from the Fukushima Daiichi NPP accident.

At the same time, international arena adopted several standards that will influence future designs of nuclear power plants. The main international standards against which the European Utility Requirements were benchmarked, with special emphasis on <u>areas that were newly introduced</u> are the following:

An **amended Nuclear Safety Directive** which was adopted by the European Council in 2004 as an amendment to the original 2009 Directive. The main amendments included:

- a new high level nuclear safety objectives
- provisions regarding the implementation of defence-in-depth expectations regarding the considerations of external hazards, both natural and man-made of magnitude greater than those considered in the design basis
- expectations regarding continuous improvement of nuclear safety including periodic safety reviews with a frequency of at least every 10 years

In particular, the Article 8a para. 1 of the amended Nuclear Safety Directive requires Member States to “ensure that the national safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and , should they occur, mitigating its consequences and avoiding:

- <u>early radioactive releases</u> that would require off-site emergency measures but with insufficient time to implement them;
- <u>large radioactive releases</u> that would require protective measures that could not be limited in area or time.”

Similarly, the **Vienna Declaration on Nuclear Safety,** which was adopted in February 2015 on a diplomatic conference as a compromise after being unable to reach a consensus on the revision of the Nuclear Safety Convention, requires in its Principle 1 that “new nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.”

In order to incorporate lessons learned from the Fukushima Daiichi accident, **WENRA** has issued two documents to reflect those:

- **Report on Safety Reference Levels for Existing Reactors** and
- **Report on Safety of new NPP designs**

The objective of the development of Safety Reference Levels was to increase harmonization within WENRA countries on safety requirements issued by the regulatory bodies and their implementation in existing nuclear power plants. Initially they identified 18 areas where harmonization was considered as necessary. After the Fukushima Daiichi accident an additional, 19th issue has been included to cover natural hazards (extreme weather conditions, external flooding, and seismic events). Major changes after the Fukushima Daiichi accident have also been in the area of Design Extension Conditions DEC – Issue F, where clear differentiation between DEC without core melt (DEC A) and DEC with core melt (DEC B) has been introduced plus addressing the DEC aspects for the spent fuel pool and multi-unit sites. NPP autonomy for a justified time has also been introduced.

The intention of the document on Safety of new NPP design was to develop common positions on selected key safety issues and safety expectations for the design of new nuclear power plants.

The new design requirements in the **IAEA Safety Standards** have been mainly reflected in the revision of SSR 2/1 Rev 1 "Safety of Nuclear Power Plants; Design". The major changes in SSR-2/1 Rev 1 as compared to the earlier version, relate to the following thematic fields:

- Prevention of severe accidents by strengthening the design basis for the plant;
- Prevention of unacceptable radiological consequences of a severe accident for the public and the environment;
- Mitigation of the consequences of a severe accident to avoid or to minimize radioactive contamination off the site.

The newly introduced concepts in SSR 2/1 have not always been interpreted in the same way and hence the IAEA has produced the TCDOC-1791 with the aim of clarifying the novelty of the concepts introduced. Even though TECDOCs were not meant to be included in this benchmark study as they represent a lower level documents which do not go through the approval process by Member States or the IAEA Board of Governors, this particular TECDOC-1791 was viewed as very important in clarifying certain issues from SSR 2/1 Rev. 1. It has proved to be a very useful source of information for benchmarking the European Utility Requirements.

Certain concepts have been introduced already in the first edition of SSR 2/1 (Rev. 0) in the year 2012, like the inclusion of the design extension conditions into the design envelope or the requirement for the strengthened independence of different levels of defence-in-depth, which were the major improvement compared to the IAEA Safety Standard NS-R-1, the predecessor of SSR 2/1, published in the year 2000. After the Fukushima Daiichi accident a revision of SSR 2/1 took place and was published in 2016 as SSR 2/1 Rev. 1. It reinforced the concepts already introduced in the SSR 2/1 Rev. 0 including the requirement that the design should address also the necessary provisions for the mitigation of severe accidents and the practical elimination of event sequences that could lead to early or large releases. In addition, the lessons learned from the Fukushima Daiichi accident led to the reinforcement of certain requirements that relate to the robustness of the design against external natural hazards exceeding those derived from the site hazard assessments and inclusion of non-permanent equipment.

The overall benchmarking of European Utility Requirements against the above international standards has resulted in the conclusion that the European Utility Requirements are in full compliance with the international standards. Detailed comparison of each individual requirement is presented in the main body of this report, however it is worthwhile to highlight the most important elements which were introduced or highlighted in all the above standards after the Fukushima Daiichi accident. These elements have been reflected in the latest revisions of the above mentioned international standards. This has been decided in order to see the comparison and compliance in the most important (or at least the newest) safety issues. In General Conclusions it has therefore been decided to highlight the comparison of European Utility Requirements against the international standards in the following key areas:

1. **Plant States considered in the design of nuclear power plants – SSR 2/1 Req. 13**
2. **Application of defence-in-depth and independence of levels of defence-in-depth – SSR 2/1 Req. 7**
3. **The concept of practical elimination**
4. **Early and large releases**
5. **Postulated initiating events – SSR 2/1 Req. 16**
6. **Internal and external hazards – SSR 2/1 Req. 17**
7. **Aircraft crash**
8. **Design extension conditions – SSR 2/1 Req. 20**
9. **Safety analysis of the plant design – SSR 2/1 Req. 42**
10. **Containment – SSR 2/1 Reqs. 54 - 58**
11. **Emergency power supply – SSR 2/1 Req. 68**
12. **Use of non-permanent equipment for accident management**
13. **Integrated management system vs. quality management system**

The conclusions on the selected 13 important issues are summarized below:

1. Plant States considered in the design of nuclear power plants

Plant states that are considered in the design of nuclear power plants are defined identically in all international standards but with different titles/names and it is therefore prudent to show the equivalence among different definitions.

In E.U.R., IAEA SSR 2/1 and in WENRA Safety of new NPP designs the initial states are identical:

- Normal operation
- Anticipated operational occurrences
- For design basis accidents and design extension conditions the below table shows different names used for the corresponding conditions:

E.U.R.	IAEA	WENRA Safety of new NPP designs
Design Basis Accidents	Design Basis Accidents	Postulated single initiating events
DEC – Complex sequences	DEC-A without core melt	Postulated multiple failure events
DEC – Severe accidents	DEC-B with core melt	Postulated core melt accidents

For the purpose of classifying the postulated initiating events, the plant conditions are categorized in the following way:

- Normal Operation as – Design Basis Condition 1 = DBC1
- Anticipated operational occurrences as – Design Basis Condition 2 = DBC2
- Design basis accidents with frequency of 10^{-2} – 10^{-4} as – Design Basis Condition 3 = DBC3
- Design basis accidents with frequency of 10^{-4} – 10^{-6} as – Design Basis Condition 4 = DBC4

Even though the names used are different, the concepts are the same in all international standards and requirements.

It is therefore evident that E.U.R. are **in full compliance** with the international standards on the issue of the definition of plant states in the design of nuclear power plants.

2. Application of defence-in-depth and independence of levels of defence-in-depth

Refined structure of the levels of defence-in-depth is included in the IAEA Safety Standards, WENRA Safety of new NPP designs and EUR document. The table below shows that all standards have the same breakdown of the level 3 of defence-in-depth but they all use different terms to describe the basic concept which is equivalent in all three documents:

Level of defence-in-depth	E.U.R.	IAEA	WENRA Safety of new NPP designs
Level 3a	Design Basis Accidents	Design Basis Accidents	Postulated single initiating events
Level 3b	DEC – Complex sequences	DEC-A without core melt	Postulated multiple failure events
Level 4	DEC – Severe accidents	DEC-B with core melt	Postulated core melt accidents

The IAEA SSR 2/1 Rev 1 in para 4.13A requires that "the levels of defence-in-depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of the other level. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems." Similar requirement can be found in WENRA Objective 4 "Independence between all levels of defence-in-depth". IAEA TECDOC-1791 devotes the entire Chapter 6 to the independence of levels of defence-in-depth.

E.U.R. in Chapter 2.1.1.4D addresses the independence of levels of defence-in-depth to the same extent as the above international requirements. In 2.1.1.4D1-D4 defines why the term "as far as reasonably practicable" is being used, i.e. describes the limitations for the full independence of levels of defence-in-depth. In 2.1.1.4D2 it explains that separation of cables is already a requirement and additional separation on the basis of defence-in-depth. In 2.1.1.4D3 it requires that the Reactor Protection System be adequately independent from other I&C systems and must be functionally separated from them; however there are cases where it is used in different levels of defence-in-depth.

Independence of the levels of Defence-in-Depth as covered in the E.U.R. document is in **full compliance** with principles described in the international standards.

3. The concept of practical elimination

The concept of practical elimination is not new but has gained more importance in revisions of international standards after the Fukushima Daiichi accident. The term "practical elimination" was first introduced in the IAEA publication INSAG-12 in 1999, where the objective for future plants was set as the practical elimination of accident sequences that could lead to large early radioactive releases. This concept was further elaborated in the IAEA Safety Guide NS-G-1.10 on Containment where the definition of practical elimination was first introduced as "the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise". This definition is still being adopted by all standards and requirements. The concept of practical elimination has been also adopted in the IAEA SSR 2/1Rev 1. in paras 2.11, 2.13(4), 4.3, 5.27, 5.31 and 6.68. They all refer to practical elimination of sequences that could lead to an early or large radioactive release. The same requirement is written in WENRA's Safety objectives for new nuclear power plants in Objective O3. Accidents with core melt.

European Utility Requirements have the same requirement in 2.1.2.5A that sequences that have the potential to cause a large release or early release shall be practically eliminated. Further in 2.1.2.5B it gives a list of sequences to be practically eliminated. This list is practically identical with the list of sequences to be practically eliminated given in the Appendix 4 of the IAEA TECDOC-1791.

In addition, E.U.R. give explanation on what constitutes "physically impossible" – in 2.1.2.5D and what constitutes "extremely unlikely" – in 2.1.2.5E.

More on the inclusion of the concept of practical elimination in international safety standards can be found in the presentation of L. Ammirabile: JRC analysis of key concepts of Article 8a: Practical achievements of the nuclear safety objective, presented on the Workshop on Implementation of Articles 8a-c of the Nuclear Safety Directive, 8- 9 March 2017, Luxembourg.

It is therefore clear that E.U.R. are **in full compliance** with the international standards on the issue of practical elimination.

4. Early and large releases

Article 8a of the amended Nuclear Safety Directive requires Member States to "ensure that the national safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and , should they occur, mitigating its consequences and avoiding:

- <u>early radioactive releases</u> that would require off-site emergency measures but with insufficient time to implement them;
- <u>large radioactive releases</u> that would require protective measures that could not be limited in area or time."

Definitions of large and early radioactive release are given in SSR 2/1 Rev 1 as:

- A <u>large radioactive release</u> is a radioactive release for which off-site protective actions that are limited in terms of lengths of time and areas of application would be insufficient for the protection of people and of the environment.
- An <u>early radioactive release</u> is a radioactive release for which off-site protective actions would be necessary but would be unlikely to be fully effective in due time.

WENRA Safety Objectives for new nuclear power reactors requires in its objective O3. Accidents with core melt that:

- Reducing potential radioactive releases to the environment from accidents with core melt, also in the long term, by following the quantitative criteria below:
  - Accidents with core melt which would lead to early or large releases have to be practically eliminated;
  - For accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.

European Utility Requirements address the issues of early and large radioactive releases in Chapters 2.1.1.1D, 2.1.2.5, 2.1.2.6.4G, 2.1.4.1F, 2.1.7.3.4D, 2.1.7.7.1A with the same definitions and requirements as amended NSD, SSR 2/1 Rev 1. and WENRA documents and as such are **in full compliance** with the international standards.

More details on early and large releases is conveniently summarized in the presentation of B. Farrar on JRC analysis of key concepts of Article 8a: Early/large release presented on the Workshop on Implementation of Articles 8a – c of the Nuclear safety Directive, 8 – 9 March 2017, Luxembourg. It provides much more details on WENRA and E.U.R. Criteria for Limited Impact for Severe Accidents as described in E.U.R. Chapter 2.1.B5. In this article a **full compliance** of E.U.R. criteria for limited impact for severe accidents with the WENRA requirements is also demonstrated.

5. Postulated initiating events

SSR 2/1 Req. 16 on Postulated initiating events and EUR Chapter 2.1.2.2 A with the same title have identical text and call for identification of a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design. As such it can be concluded, that there is a **full compliance** in this thematic area.

SSR 2/1 para 5.5 calls for the use of engineering judgement and a combination of deterministic and probabilistic safety assessments in identification of postulated initiating events (PIEs). This requirement is captured in EUR Chapter 2.1.2.2A.

SSR 2/1 para 5.6 requires that PIEs include all foreseeable failures of SSCs as well as operator errors and possible failures arising from internal and external hazards in all modes of plant operation. The same requirement is captured in EUR Chapter 2.1.2.2E.

SSR 2/1 para 5.7 requires that an analysis of the PIEs be made to establish the preventive and protective measures necessary for the assurance of required safety functions. Exactly the same wording can be found as a requirement in EUR Chapter 2.1.2.2G.

SSR 2/1 para 5.8 lists the expected behaviour of the plant in any PIE in order of their priority. Exactly the same wording can be found as a requirement in EUT Chapter 2.1.2.2H.

SSR 2/1 para 5.9 requires grouping of PIEs when used for developing the performance requirements for the items important to safety. Exactly the same wording can be found as a requirement in EUR Chapter 2.1.2.2I.

SSR 2/1 para 5.10 requires a technical justification for exclusion from the design of any PIE that has been identified. Exactly the same wording can be found as a requirement in EUR Chapter 2.1.2.2J.

SSR 2/1 para 5.11 call for automatic safety actions for the actuation of safety systems when prompt and reliable action would be necessary. Such requirement can be found on several places in the EUR document. EUR Chapter 2.1.1.4C requires that the design shall provide for supplementing the control of plant by means of automatic actuation of safety systems, EUR Chapter 2.1.4.2A requires that DSA shall provide demonstration that the management of DEC is possible by the automatic actuation of safety systems, EUR Chapter 2.1.5.1C1 dealing with classification of SSCs requires that an automatic actuation may be required for those functions which require prompt actions as established based on the sequence of events following a PIEs, EUR Chapter 2.1.6.3.1C which requires that an automated action be provided where prompt and reliable operator action is required following any internal or external hazard.

SSR 2/1 para 5.12 deals with operator actions in response to PIE which do not need a prompt response. This issue is adequately dealt with in the EUR Chapter 2.1.6.7.1 on Autonomy objectives in respect of operators and plant personnel.

In this Chapter EUR is more specific and requires that for the first 30 minutes from the first significant signal, the release targets if AOO and accident conditions shall be met without operator action in the control room.

In addition it requires that the containment be designed in such a way that it would sustain any of the severe accidents considered in DEC without operator action during the first 12 hours from the beginning of the severe accident. They also add that the designer should aim at extended this period up to 24 hours.

SSR 2/1 para 5.13 requires that capability to monitor the status of the plant be available. Similar requirement can be found in EUR Chapter 2.1.1.2C where it is stated that means of monitoring the status of the plant shall be provided for ensuring that required fundamental safety functions are performed.

SSR 2/1 para 5.14 requires equipment and procedures to be available for keeping control over the plant and for mitigating any harmful consequences. Similar requirement can be found in EUR Chapter 2.1.7.4.1 on provision of instrumentation.

SSR 2/1 para 5.15 requires any equipment for which manual action is required to be placed in the most suitable location to ensure its availability at the time of need and to allow safe access to it under the environmental conditions anticipated. The same is elaborated in EUR Chapter 2.1.6.7.1 E.

In addition, EUR Chapter 2.1.9.4 provides the list of PIEs for NO, AOO, DBC3, DBC4 and DEC-complex sequences for PWR and BWR reactors.

6. Internal and external hazards

IAEA SSR 2/1 Req. 17 stipulates that all foreseeable internal and external hazards, …….. shall be identified …..and evaluated. Hazards shall be considered in designing the layout of the plant and in determining the PIEs and generated loadings for use in the design of relevant items important to safety for the plant. Exactly the same wording is used in the EUR Chapters 2.1.2.6 A and C and therefore it can be concluded that there is a **full compliance** in this respect.

Paragraphs SSR 2/1 Req. 17 para 5.15A – 5.22 elaborate in some more detail specific requirements for internal (SSR 2/1 Req. 17 para 5.16) and external hazards (SSR 2/1 Req. 17 paras 5.17-5.22).

SSR 2/1 Req. 17 para 5.15A calls for the items important to safety to withstand the effects of hazards or to be protected. The equivalent requirement with more specifics can be found in EUR Chapter 2.18.4.3 dealing with failure mechanisms.

SSR 2/1 Req. 17 para 5.15B addresses hazards at multiple unit plant sites. The exact same wording is in EUR Chapter 2.1.2.6 B requiring that for multiple unit sites, the design shall take due account of the potential for specific hazards to give rise to impact on several or even all units on the site simultaneously.

SSR 2/1 Req. 17 para 5.16 addresses Internal hazards and provides the example list of internal hazards and call for appropriate features for prevention and mitigation to be provided. Exact same wording is in EUR Chapter 2.1.2.6.1 on consideration of internal hazards.

SSR 2/1 Req. 17 paras 5.17 – 5.22 address External hazards. Para 2.17 requires that causation and likelihood shall be considered in postulating potential hazards. The same requirement is in EUR Chapter 2.1.2.6 C. Further in the same para it is required that in the short term, the safety of the plant shall not be dependent on the availability of off-site souses such as electricity supply and firefighting services and that the maximum delay time by which off-site services need to be available be determined. The same requirements are in EUR Chapter 2.1.2.6 D.

SSR 2/1 Req. 17 para 5.19 requires that features shall be provided to minimize any interaction between buildings containing items important to safety and any other plant structures as a result of external events considered in the design. Similar requirements are addressed in EUR Chapter 2.1.6.3.2.1 on Independence and EUR Chapter 2.1.6.3.2.2 on Functional isolation.

SSR 2/1 Req. 17 paras 5.21 and 5.21A require design to provide sufficient safety margins to avoid cliff-edge effects and early or large radioactive release. Very similar requirements are in EUR Chapter 2.1.2.6.3.1 D.

EUR Chapter 2.1.2.6.2.1 on Identification of external hazards distinguishes between Design Basis External Hazards (DBEH) and Rare and Severe External Hazards (RSEH). These are defined in 2.1.2.6.2.1 A1 and A2 as:

- DBEH are those for which the plant is designed
- RSEH as those more challenging or less frequent external hazards

This distinction goes beyond requirements of SSR 2/1. However, the IAEA TECDOC-1791 uses the same distinction labelling them Design Basis External Events (DBEE) and Beyond Design Basis External Hazards (BDBEE), having the same meaning and requirements.

Additionally, EUR provide the entire list of internal (EUR Chapter 2.1.9.3.1) and external (EUR Chapter 2.1.9.3.2) hazards to be considers in the design, which again is not considered in SSR 2/1.

7. Aircraft crash

EUR Requirements on Aircraft crash (EUR Chapter 2.1.2.6.2.1.1) are divided into two categories:

- Accidental aircraft crash
- Intentional aircraft crash as a result of human malevolent action

The designer shall identify the buildings which have to be protected. As a minimum the following shall be included (EUR Chapter 2.1.8.3.3 C1):

- The rector building
- The fuel handling building and spent fuel pool
- The building housing steam/feedwater lines from the containment up to and including at least the steam and feedwater isolation valves
- Auxiliary buildings

Further intentional aircraft crash requirements are given under the Security section in EUR Chapter 2.1.8.3.3:

- Event should be considered in the design but with Best Estimate design rules and radiological consequences associated with complex sequences.
- Intentional aircraft crash shall not lead to core melt. Releases of radioactive material should meet the safety objectives and radiological release targets for complex sequences.

Load/time curves for the accidental aircraft crash are given in EUR Chapter 2.4.1.3.3.1 for accidental aircraft crash:

- For light aircraft (EUR 2.4.1.3.3.1.1 Figure 4)
- For military aircraft (EUR 2.4.1.3.3.1.2 Figure 5).

And for Intentional aircraft crash in EUR Chapter 2.4.1.3.3.2 Figure 6.

Protection shall be based on PSA methodology described in EUR Chapter 2.17.6.8. PSA should consider three types of aircraft crashes (EUR Chapter 2.17.6.8.A1):

- Light aircraft
- Military aircraft such as fighters
- Commercial aircrafts such as large passenger carriers.

IAEA Safety Requirements NS-R-3 Rev 1 on Site Evaluation for Nuclear Installations address the aircraft crash in paras 3.44 – 3.47 and requires the site specific evaluation of such hazards, including impact, fire and explosion. If assessment shows that hazards are unacceptable, the site shall be deemed unsuitable.

IAEA Nuclear Security Series in the IAEA NSS-10 call for inclusion of the aircraft crash into the Design Basis Threat (DBT). The IAEA NSS-4 on Engineering Safety Aspects of the Protection of NPPs against Sabotage classifies the aircraft crash as threat type 2 (TT-2) but does not include any recommendations on this issue.

WENRA document of Safety of New NPP Design addresses intentional crash of a commercial aircraft on an NPP in Position 7 (O3.7) and is given as an example of safety and security interface described in Objective O5. WENRA Positon O3.7 has very similar requirements as the EUR document. The main elements are as follows:

- The event of a commercial aircraft crash should be considered in the design and should be characterized by load/time curves.
- The event is considered as a very significant example of safety and security interface.
- The general expectation is that a crash should not lead to core melt.
- In particular the following shall be ensured:
  - Reactivity control
  - Residual heat removal
  - Confinement of radioactive material
- Direct and indirect effects of the airplane crash shall be considered.

It cab ne concluded that EUR provide more detailed requirements for the event of an airplane crash and are in **full compliance** with the international standards.

8. Design extension conditions

SSR 2/1 Req. 20 deals with the Design Extension Conditions (DEC). It specifies that a set of DEC shall be derived on the basis of engineering judgement, DSA and PSA for the purpose of further improving the safety of the NPP by enhancing plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than DBA or that involve additional failures. These DEC shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences. EUR Chapter 2.1.2.4 A and B have exactly the same wording as SSR 2/1 Req. 20 and as such it can be concluded that there is a **full compliance** in this issue. Even though the basic concepts in both requirements (plus in WENRA Safety of new design) are the same, the titles given to individual parts of DEC are different in those three requirements documents. The table below depicts the names used by EUR, IAEA and WENRA to describe different stages in DEC.

E.U.R.	IAEA	WENRA Safety of new NPP designs
DEC – Complex sequences	DEC-A without core melt	Postulated multiple failure events
DEC – Severe accidents	DEC-B with core melt	Postulated core melt accidents

SSR 2/1 Req. 20 para 5.27 requires that the safety analysis for DEC be performed but allows (in the footnote) that Best Estimate (BE) approach be used. BE approach can be used also for the analysis of the effectiveness of provisions to ensure the functionality of the containment. This is reflected in EUR Chapter 2.1.4.2.3 B on DSA where it is stated that safety analysis for DEC shall relay on BE approach, giving details for complex sequences and severe accidents. Corresponding use of BE approach for containment functionality is addressed in EUR Chapter 2.9.2.2.1.2 C1 on Primary containment performance. In addition, definition of DEC in the EUR document reads: "DEC are considered in the design process of the facility in accordance with BE methodology".

The same para 5.27 further defines the main technical objective of considering DEC is to provide assurance that the design of the plant is such as to prevent accident conditions that are not considered in design basis accidents or to mitigate their consequence. For this an additional safety features are necessary or the extension of capabilities of existing safety systems. This is adequately addressed in EUR Chapter 2.1.2.4 B, B1, B2, B3.

SSR 2/1 Req. 20 para 5.28 defines that the design specifications for safety features be derived from DEC. Similar requirement is in EUR Chapter 2.1.2.4 B2 and B3.

SSR 2/1 Req. 20 para 5.29 makes requirements for safety features to be used in DEC. They are required to:

- Be independent, to the extent practicable, of those used in more frequent accidents. In this respect, EUR is even more precise; In EUR Chapter 2.1.2.4.1.E it require that safety features for DEC used in complex sequences be, as far as reasonably practicable, independent of safety systems and in EUR Chapter 2.1.2.4.2 E that safety features for DEC used in severe accidents be as far as reasonably practicable, independent of safety systems and of safety features used in complex sequences.
- Be capable of performing in the environmental conditions pertaining to DEC. Such requirement can be found in EUR Chapter 2.1.5.1.7.1 A on Requirements on environmental and hazard conditions resistance and EUR Chapter 2.1.2.4.3.3 A on Reference Source Term.
- Shall have reliability commensurate with the functions that they are required to fulfil. This is addressed in EUR Chapter 2.1.6.3.1.1B where it is stated that safety features for DEC should have sufficient redundancy of active components to reach adequate reliability and in EUR Chapter 2.1.6.4 A addressing reliability of items important to safety .

SSR 2/1 Req. 20 para 5.30 requires the containment and its safety features be able to withstand extreme scenarios including core melt and that these scenarios shall be selected using engineering judgement and inputs from PSA. EUR Chapter 2.9.3.1.7.1 addresses this point in stating that the containment shall be designed for mitigating the consequences of severe accidents. In addition, in EUR Chapter 2.9.3.1.8.1 C it is even more precisely stated that the primary containment design shall be such that it can withstand any severe accident in DEC without operator action for the first 12 hours.

SSR 2/1 Req. 20 para 5.31 requires that the design be such that the possibility of conditions arising that could lead to an early radioactive release or large radioactive release be practically eliminated. The same requirement can be found in EUR Chapter 2.1.1.1E on general safety objectives where it is stated that for accident sequences leading to core melt, that have not been practically eliminated, design provisions shall be taken so that:

- only limited protection measures in area and time are needed for the public
- sufficient time is available to implement these measures.

SSR 2/1 Req. 20 para 5.31A requires that design shall be such that in case of DEC, protective actions limited in terms of length of time and areas of application shall be sufficient. EUR Chapter 2.1.3.1 A1 states that for severe accidents, targets are set to avoid the need of significant off-site protective actions consistently with WENRA's safety objective O3. Criteria for limited impact of severe accidents would lead to limited social consequences.

9. Safety analysis of the plant design

SSR 2/1 Req. 42 addresses Safety analysis of the plant design. It requires a safety analysis of the design for the NPP to be conducted in which DSA and PSA shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed. EUR Chapter 2.1.4.1 A has <u>exactly the same wording</u> as SSR 2/1 Req. 42 and as such **full compliance** can be attributed to this issue.

SSR 2/1 req. 42 para 5.71 requires that based on safety analysis, the design basis for items important to safety be established. The same wording can be found in EUR Chapter 2.1.4.1 C. Further, para 5.71 requires that design shall comply with authorized limits on discharges and dose limits in all operational states. EUR Chapter 2.1.9.2 Table 3 presents frequencies and general acceptance criteria for all plant states.

SSR 2/1 Req. 42 para 5.72 requires that safety analysis shall provide assurance that defence-in-depth has been implemented in the design of the plant. The same requirement can be found in EUR Chapter 2.1.4.1 E.

SSR 2/1 Req. 42 para 5.73 requires that safety analysis shall provide assurance on safety margins, avoidance of cliff-edge effects and early and large radioactive releases. The same wording as in para 5.73 can be found in EUR Chapter 2.1.4.1 F.

SSR 2/1 Req. 42 para 5.74 requires that the applicability of the analytical assumptions, methods and degree of conservatism used in the design of the plant shall be updated and verified for the current or as built design. The same wording is in EUR Chapter 2.1.4.1 G.

SSR 2/1 Req. 42 para 5.74 provides the requirements for Deterministic Safety Analysis in 6 points (a) to (f). Exactly the same wording can be found in EUR Chapter 2.1.4.2 A.

EUR Chapter 2.1.4.2 on Deterministic Safety Analysis gives requirements in its subchapters 2.1.4.2.1, 2.1.4.2.2 and 2.1.4.2.3 which go beyond the SSR 2/1 Req.42.

- In 2.1.4.2.1 on general principles it defines acceptance criteria, safety objectives and targets for all plant states. It further defines “safe state” following AOO, DBA and Complex sequences and “severe accident safe state” and requires that safe state shall be reached within 24 hours following AOO and DBA, within 24 hours but in any case before 72 hours for complex sequences and within 7 days following severe accidents. Such numerical targets go far beyond requirements in IAEA Safety Standards.
- In 2.1.4.2.2, rules used in performing deterministic safety analysis are prescribed, such as use of single failure criteria and requirement to postulate LOOP with each initiating event when analysing AOO and DBA. This again goes beyond requirements in the IAEA Safety Standards but can be found in lower level documents such as Safety Report Series # 23 “Accident analysis for NPPS”.
- In 2.1.4.2.3 defines deterministic safety analysis methodologies. For AOO and DBA three options are available: full conservative analysis with conservative codes and conservative initial and boundary (I&B) conditions, best estimate (BE) codes and conservative I&B conditions and BE with evaluation of uncertainties (BEPU). For DEC it is suggested to use only BE approach due to large uncertainties. IAEA Safety Guide SSG-2 on Deterministic Safety Analysis supports such analysis methodologies. In addition to three listed methods; conservative, BE and BEPU, the SSG-2 also introduces the so-called Option 4 or Extended BEPU that included BEPU plus the treatment of the availability of safety systems in a probabilistic manner.

SSR 2/1 Req. 42 para 5.76 deals with probabilistic safety analysis. EUR Chapter 2.1.4.3 addresses the same and in EUR Chapter 2.1.4.3 A uses the same wording as requirements in SSR 2/1 Req. 43 para 5.76 (a) – (c). EUR however address PSA in a separate Chapter 2.17 on PSA Methodology, elements of which are equivalent to those in the IAEA safety Guides SSR-3 and SSR-4.

10. Containment SSR 2/1 Reqs. 54 - 58

SSR 2/1 Requirements 54 – 58 set requirements for the Containment structure and containment system. EUR document devotes the entire chapter 2.9 to the Containment system and addresses several important aspects also in Chapter 2.1 on Safety requirements. SSR 2/1 does not make distinction on primary and secondary containment as does EUR document but treats the containment as a single entity which needs to fulfil certain safety functions as required in SSR 2/1 Req. 54 and EUR Chapter 2.1.7.3.1.A.

SSR 2/1 Req. 54 requires that the containment shall assure the following safety functions:

- Confinement of radioactive substances in operational states and in accident conditions
- Protection of the reactor against natural external events and human induced events
- Radiation shielding in operational states and accident conditions.

The exact same wording can be found in EUR Chapter 2.1.7.3.1 A and therefore there is a **full compliance** on this issue.

SSR 2/1 Req. 55 requires that containment be such as to ensure that any radioactive release is ALARA, is below authorized limits on discharges in operational states and is below acceptable limits in accident conditions. EUR Chapter 2.1.7.3.2 A has the same requirement with the only difference that it requires discharges in operational states to be below EUR discharge targets and in accidental conditions to be below EUR radiological release targets. The requirements are seen as equivalent and it is concluded that there is a **full compliance** in this issue.

SSR 2/1 Req. 55 para 6.20 sets requirement on the containment leak rate testing. EUR Chapter 2.9.2.2.2.1.2 A also requires periodic integrated leak rate tests (ILRT) of the containment.

SSR 2/1 Req. 55 para 6.21 requires that the number of penetrations through the containment be minimized. This is reflected in EUR Chapter 2.9.4.1.9 A. The entire EUR Chapter 2.9.4.1.9 is devoted to requirements on containment penetrations and in 2.9.4.1.9 M sets requirements for protecting the seal gaskets against missiles and jet pipes as also required in SSR 2/1 Req.55 para 6.21.

SSR 2/1 Req. 56 deals with the isolation of the containment. It requires that all penetrations be automatically and reliably sealable in in the event of accidents. Exactly the same wording is in EUR Chapter 2.1.7.3.3 and therefore it is concluded that there is a **full compliance** in this issue.

SSR 2/1 Req. 56 para 6.22, para6.23 and para 6.24 address requirements for the containment penetration isolation or check valves. EUR Chapter 2.9.4.1.7 provides requirements for the containment isolation system and in 2.9.4.1.7 B specifies that an acceptable configuration of isolation devices shall be defined. In 2.9.4.1.7 B1 it specifies that the IAEA SSR 2/1 paras 6.22, 6.23 and 6.24 are considered as acceptable configuration of isolation devices.

SSR 2/1 Req. 57 deals with the access to the containment and specifies that the access by operating personnel to the containment shall be through airlocks equipped with doors that are interlocked to ensure that at least one of the doors is closed during reactor power operation <u>and in accident conditions.</u> EUR Chapter 2.9.2.5.1 A-B has exactly the same wording except that the end of the sentence refers to “power operation modes” i.e. there is no specific reference to accident conditions. Power operation modes are defined in the EUR as modes where the reactor is critical. We therefore conclude that there is a **full compliance** in this issue.

SSR 2/1 Req. 57 para 6.25 requires provisions for ensuring protection and safety for operating personnel entering containment for surveillance purposes. Such explicit requirement was not found in EUR Chapter 2.9 specifying requirements for the Containment system. In general, protection and safety of operating personnel is in most cases dealt with in separate documents.

SSR 2/1 Req. 57 para 6.26 requires that containment openings for the movement of equipment shall be designed to close quickly and reliably in the event that isolation of the containment is required. This is not explicitly mentioned in EUR event though the EUR Chapter 2.9.4.1.7 on Containment isolation system requires quick and reliable closer of isolation valves following the isolation signal.

SSR 2/1 Req. 58 addresses the control of containment conditions. It requires that pressure and temperature be controlled, that build-up of fission products or other gaseous, liquid or solid substances be controlled. Exactly the same wording is in EUR Chapter 2.1.7.3.4 A and therefore it is concluded that there is a **full compliance** in this issue.

SSR 2/1 Req. 58 para 6.27 deals with the requirements that affect separate compartments within the containment. Exactly the same wording is in EUR Chapter 2.1.7.3.4 B.

SSR 2/1 Req. 58 para 6.28 deals with the capability to remove the heat from the containment. Exactly the same wording is in EUR Chapter 2.1.7.3.4 C.

SSR 2/1 Req. 58 para 6.28A requires provisions to be made to prevent the loss of structural integrity of the containment in all plant states. Exactly the same wording is in EUR Chapter 2.1.7.3.4 D.

SSR 2/1 Req. 58 para 6.28B requires the design to include features to enable the safe use of non-permanent equipment for restoring the capability to remove heat from the containment. Exactly the same wording is in EUR Chapter 2.1.7.3.4 E.

SSR 2/1 Req. 58 para 6.29 prescribes the design features to control fission products, hydrogen, oxygen and other substances that might be released into the containment. Exactly the same wording is in EUR Chapter 2.1.7.3.4 F.

SSR 2/1 Req. 58 para 6.30 covers coverings, thermal insulation and coatings for components and structures within the containment system. Such requirements are not found in the EUR Chapters 2.1 on Safety requirements or 2.9 on Containment system. However the entire EUR Chapter 2.6 deals with Material related requirements and specifies requirements on thermal insulation (2.6.4.5) and on coatings (2.6.4.6 B) but does not specifically address the containment system.

11. Emergency power supply

SSR 2/1 Req. 68 addresses requirements for the Emergency power supply i.e. the design for withstanding the loss of off-site power. It requires the design to include an <u>emergency power supply</u> capable of supplying necessary power in <u>AOOs and DBAs</u>, in the event of a loss of off-site power. The design shall include an <u>alternate power source</u> to supply the necessary power <u>in DEC</u>. Exactly the same wording is in EUR Chapter 2.1.7.5 A and therefore it can be concluded that there is a **full compliance** in this issue.

SSR 2/1 Req. 68 para 6.43 defines that for both, the emergency power supply and for alternate power supply the design specifications shall include the requirements for capability, availability, duration of the required power supply, capacity and continuity. Exactly the same wording is in EUR Chapter 2.1.7.5 B.

SSR 2/1 Req. 68 para 6.44 requires that the combined means to provide emergency power shall have a reliability and type that are consistent with all the requirements of the safety systems to be supplied with power, and their functional capability shall be testable. Exactly the same wording is in EUR Chapter 2.1.7.5 C.

SSR 2/1 Req. 68 para 6.44A requires that the alternate power supply be capable of supplying necessary power to preserve the integrity of the RCS and to prevent significant damage to the core and to spent fuel in the event of LOOP combined with failure of the emergency power supply. The same wording is in EUR Chapter 2.1.7.5 E with the difference that EUR uses “prevent core damage” without the word significant and at the end uses the expression “station black out” instead of “LOOP combined with failure of the emergency power supply”, which is the same.

SSR 2/1 Req. 68 para 6.44B requires that equipment that is necessary to mitigate the consequences of melting the core shall be capable of being supplied by any of the available power sources. Exactly the same wording is in EUR Chapter 2.1.7.5 J.

SSR 2/1 Req. 68 para 6.44C calls for independence and physical separation of the alternate power source from the emergency power supply. Exactly the same wording is in EUR Chapter 2.1.7.5 F. In addition EUR specifies that the connection time of the alternate power source shall be consistent with the depletion time of the batteries.

SSR 2/1 Req. 68 para 6.44D requires that continuity of power for monitoring of the key plant parameters and for completion of short term actions necessary for safety shall be maintained in the event of loss of the AC power sources. Exactly the same wording is in EUR Chapter 2.1.7.5 I.

SSR 2/1 Req. 68 para 6.45 specifies the design basis for any diesel engine or other prime mover that provides emergency power supply to items important to safety. Exactly the same wording is in EUR Chapter 2.1.7.5 D. Prime mover is defined in SSR 2/1 as a component (such as a motor, solenoid operator or pneumatic operator) that converts energy into action when commanded by an actuation device.

SSR 2/1 Req. 68 para 6.45A requires that the design shall also include features to enable the safe use of non-permanent equipment to restore the necessary electrical power supply. Exactly the same wording is in EUR Chapter 2.1.7.5 K.

12. Use of non-permanent equipment for accident management

Use of non-permanent equipment is addressed in the IAEA SSR 2/1 for the following conditions:

- to restore the capability to remove heat from the containment (SSR 2/1 para 6.28B)
- to restore the necessary electrical power supply (SSR 2/1 para 6.45A)
- to ensure sufficient water inventory for the long term cooling of spent fuel (SSR 2/1 para 6.68)

In all cases it is defined that non-permanent equipment need not necessarily be stored on-site. Further, TECDOC-1791 devotes the entire Chapter 10 to the use of non-permanent equipment.

E.U.R. in Chapter 2.1.6.8 gives requirements on the use of on-site and off-site permanent equipment and specifies the conditions for their use in greater detail. It can therefore be concluded that E.U.R. are **in full compliance** with the international standards on the issue of the use of non-permanent equipment.

13. Integrated Management System vs. Quality Management System

SSR 2/1 Chapter 3 addresses the *Management system,* whereas E.U.R. Chapter 2.15 addresses the *Quality Assurance Programme.* These two concepts are similar but not identical and the IAEA Safety Standards were making a clear distinction between the two already in their early versions issued before the Fukushima Daiichi revisions.

Today, the state-of-the-art in this field is the IAEA Safety Requirements GSR Part 2 on Leadership and Management for Safety.

The concepts from this publication have evolved over time from pure quality control systems (early IAEA QA Standards – Safety Series No. 50-C/SG-Q Codes and Safety Guides Q1 – Q14), i.e. simple checks such as inspections and tests, to quality assurance and quality management systems (best represented by the International Organization for Standardization – ISO Standards).

Further development of these concepts led to a development of new IAEA Safety Requirements GS-R-3 on Management Systems for Facilities and Activities with accompanying Safety Guides GS-G-3.1 on Application of the Managements Systems for Facilities and Activities and GS-G-3.5 on the Management System for Nuclear Installations.

The IAEA Safety Requirements GS-R-3 has been superseded by the new requirements publication GSR Part 2 which introduces the concept of an Integrated Management System. Such system provides a single framework for the arrangements and processes necessary to address all the goals of the organization. These goals include safety, health, environment, security, quality and economic elements, and other considerations such as social responsibility (Req. 6 from GSR Part 2). A key advantage of the integrated management system approach , compared to the quality management system as described in ISO standards is that is explicitly incorporates safety. This is included in every aspect of the organization, including in procurement specifications and evaluations of suppliers and supplier requirements.

New element in GSR Part 2 is the introduction of the concept of “culture for safety” in addition to already established concept of “safety culture”. It is argued that by moving towards “culture for safety” we acknowledge that “safety culture” is not a separate entity that can be installed or removed from an organizational culture. It is rather an outcome of the organizational culture as it influences every aspect of how the organization’s members behave, from how the management system is developed to how defence-in-depth principles are manifested. Therefore the goal of any organization is to create an organizational culture that is working to achieve safety day by day – that is a culture for safety. Another hey principle introduced in GSR Part 2 is the influence of leadership on safety. It states that the safety performance of an organization starts with leadership. Leaders establish values and further align those values throughout the organization. Leaders set expectations and ensure accountability for their safety programmes. Therefore, leaders set standards for safe behaviour which in turn encourages and motivates workers to effectively engage in safe behaviour.

E.U.R Chapter 2.15 recognizes this development and clearly states that “the overall QAP of the project, as well as management processes and QAPs of Contractors, and review organisations, shall be in accordance with the requirements of the nuclear specific safety standard IAEA GSR Part 2: Leadership and Management for Safety (2016)” (2.15.1E). It further states, that the Quality Assurance programme is part of the Management system (2.15.1A) and further, on few places emphasises, for example, that “where potential contradiction between ISO 9001 (2015) – being a general quality standard for products and services - and IAEA GSR Part 2 (2016) – being nuclear safety standards - exists in the implementation phase, the nuclear specific features of the IAEA GSR Part 2 (2016) standard shall be mandatory QAP input for nuclear related Structures, Systems and Components (SSCs)” (2.15.1F).

GSR Part 2 on Leadership and Management for Safety clearly supports the Requirements 1, 2 and 3 from SSR 2/1 which deal with the Management of Safety in Design.

Requirement 1 from SSR 2/1 which defines the ‘Responsibility in the management of safety in plant design”, determining that an applicant for licence to construct and/or operate a nuclear power plant shall be responsible for ensuring that the design submitted to the regulatory body meets all applicable safety requirements. This responsibility is only implicit in the E.U.R. Chapter 2.15. It is stated in 2.15.1A that “the award of contracts for the project does not change the safety and quality responsibilities of the Owner”, which can be interpreted to cover the same undivided responsibility.

10 CONCLUSIONS

Task 1 addresses the benchmarking of the European Utility Requirements (EUR) against international safety standards. The objective of this task was to corroborate the EUR organisation's claims that the EUR Revision E is in line with the requirements of the IAEA, WENRA and the amended Nuclear Safety Directive of the EU. Even though certain differences exist, they are seen as being there in most cases due to different interpretation or different level of depth in addressing certain requirements. On few occasions E.U.R. make a simple statement that if certain requirements are not sufficiently specified the corresponding IAEA safety standards should be applied. This approach has been accepted as being adequate. Detailed comparison of each individual requirement is presented in tabular form in the main body of the report in which also observed differences are described.

A special Chapter at the end of the task 1 report is devoted to the analysis of selected, most important requirements that were introduced in all international standards after the Fukushima accident. It should be pointed out that in these, most recent and most important international standards full compliance has been identified with no significant observations, as obviously the creators of the E.U.R. have also devoted needed attention to these emerging issues.

The overall benchmarking of European Utility Requirements against these international standards has resulted in the conclusion that the European Utility Requirements **are in full compliance** with the international standards.

<u>IAEA Safety Standards used for the benchmarking</u>

The Benchmarking included the review of the following IAEA Safety Standards to establish relevance to E.U.R.:

Fundamental Safety Principles SF-1

Safety Requirements:

Governmental, Legal and Regulatory Framework for Safety

General Safety Requirements, GSR Part 1 (Rev. 1)

Leadership and Management for Safety

General Safety Requirements, GSR Part 2

Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards

General Safety Requirements, GSR Part 3

Safety Assessment for Facilities and Activities

General Safety Requirements, GSR Part 4 (Rev. 1)

Predisposal Management of Radioactive Waste

General Safety Requirements, GSR Part 5

Decommissioning of Facilities

General Safety Requirements, GSR Part 6

Preparedness and Response for a Nuclear or Radiological Emergency

General Safety Requirements, GSR Part 7

Site Evaluation for Nuclear Installations

Specific Safety Requirements, NS-R-3 (Rev. 1)

Safety of Nuclear Power Plants: Design

Specific Safety Requirements, SSR-2/1 (Rev. 1)

Safety of Nuclear Power Plants: Commissioning and Operation

Specific Safety Requirements, SSR-2/2 (Rev. 1)

Other Safety Requirements were found not to be of relevance for this Benchmarking study.

Safety Guides:

Design of Reactor Containment Systems for Nuclear Power Plants

Specific Safety Guides, NS-G-1.10

Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants

Specific Safety Guides, NS-G-1.11

Design of the Reactor Core for Nuclear Power Plants

Specific Safety Guides, NS-G-1.12

External Events Excluding Earthquakes in the Design of Nuclear Power Plants

Specific Safety Guides, NS-G-1.5

Seismic Design and Qualification for Nuclear Power Plants

Specific Safety Guides, NS-G-1.6

Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants

Specific Safety Guides, NS-G-1.7

Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants

Specific Safety Guides, NS-G-1.9

Fire Safety in the Operation of Nuclear Power Plants

Specific Safety Guides, NS-G-2.1

External Human Induced Events in Site Evaluation for Nuclear Power Plants

Specific Safety Guides, NS-G-3.1

Deterministic Safety Analysis for Nuclear Power Plants

Specific Safety Guides, SSG-2

Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants

Specific Safety Guides, SSG-3

Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants

Specific Safety Guides, SSG-4

Safety Classification of Structures, Systems and Components in Nuclear Power Plants

Specific Safety Guides, SSG-30

Design of Electrical Power Systems for Nuclear Power Plants

Specific Safety Guides, SSG-34

Design of Instrumentation and Control Systems for Nuclear Power Plants

Specific Safety Guides, SSG-39

Seismic Hazards in Site Evaluation for Nuclear Installations

Specific Safety Guides, SSG-9

Operating Experience Feedback for Nuclear Installations

Specific Safety Guides, SSG-50

Other Safety Guides were found not to be of relevance for this benchmarking study.

Apart from the above safety fundamentals, safety requirements and safety guides the benchmarking study looked also at a number of other IAEA documents of lower level where it was thought to be of relevance for the study. These documents included TECDOCs and INSAG report.

TASK 2 - Possible application of the Franco German ETC

TENDER SPECIFICATIONS OF TASK 2

EU Tender ENER/D2/2016-677 specified the requirements for Task 2 as Follows:

"The Franco German ETC. This initiative gathered the French and German TSOs to develop more practical design related issues, common understanding and practical description of safety requirements into safety features. According to the industry, had Germany not reversed its nuclear policy at the end of the nineties, it is very likely that EPRs built in France and Germany would have had a very similar design basis of the nuclear island. This work is considered sufficiently detailed to support efficiently licensing detailed review and was integrated in the AFCEN relevant codes.

Deliverable: a feasibility study on the possibility to extend this existing work to other national legal frameworks and other type of reactors. If the study is positive it should indicate in detail the steps to be taken to realize the full study, should it be decided by the Commission."

Under this task the experience from experts involved in this collaboration was solicited and extensive surveys of available literature, published papers and official publications dealing with the issue were conducted. Based on the information gathered, and based on the knowledge of experts involved in the implementation of this task on the current status of international codes and standards, a feasibility study was performed on the way that Franco-German experience can be utilized for possible future cooperation between or among different European countries in designing, licensing or constructing a nuclear power plant of selected reactor types. At the end, steps were detailed that would be necessary to be undertaken if a full study on the feasibility of such cooperation would be performed.

The work undertaken under this Task 2 was performed in several steps:

1. Description of the process which led to the establishment of the "Guidelines" and ETCs. Both positive aspects and challenges are addressed as well as lessons learned.
2. How was the work performed and how this successful output was achieved?
3. How did the TSOs and regulators come up with a set of codes which were acceptable to both sides, in spite of well-established and detailed codes being in existence in France (RCC) and in Germany (KTA rules)?
4. Feasibility study; could this process be replicated today, for any given pair or group of regulators and any given design, and under which boundary conditions it could succeed?
5. Steps to be undertaken if Full Study is to be initiated.

1 DESCRIPTION OF THE PROCESS

In 1989, Framatome and Siemens founded a joint venture called Nuclear Power International (**NPI**), whose primary aim was to develop a standard PWR design (the Evolutionary Power Reactor-EPR). A key goal of the EPR development was to ensure licensability of the design in France and Germany.

For this purpose, the French and German safety authorities (at that time **DSIN** in France and **BMU** in Germany) extended their already existing cooperation to develop *a common safety approach defining safety objectives and general principles* for licensing of future nuclear power plant projects. This was backed by the two governments: In a joint government declaration of 6 June 1989, France and Germany agreed on a close cooperation in the field of nuclear safety.

The German BMU and the French DSIN issued a common declaration and an agreement between the technical support organisations **GRS** (Gesellschaft fuer Anlagen- und Reaktorsicherheit) and **IRSN** (Institut de Radioprotection et de Sûreté Nucléaire - at that time IPSN) was signed for the development of *technical basis*.

In 1990 the **DFD** (Deutsch-Französischer Direktionsausschuss – German-French Directorate) was created with the objective to harmonize the general safety objectives. Its members came from the French **GPR** - "Groupe Permanent Réacteur" and the German **RSK** - "Reaktorsicherheitskommission", which were the advisory bodies to DSIN and BMU respectively. Cooperation between GPR and RSK had an objective to *harmonize the French and German licensing requirements*.

The cooperation is depicted in the below Figure 1 taken from the reference "Roadmap Towards European reactor Design Acceptance – Annex C, European Nuclear Energy Forum, Bratislava – Prague, 2012"

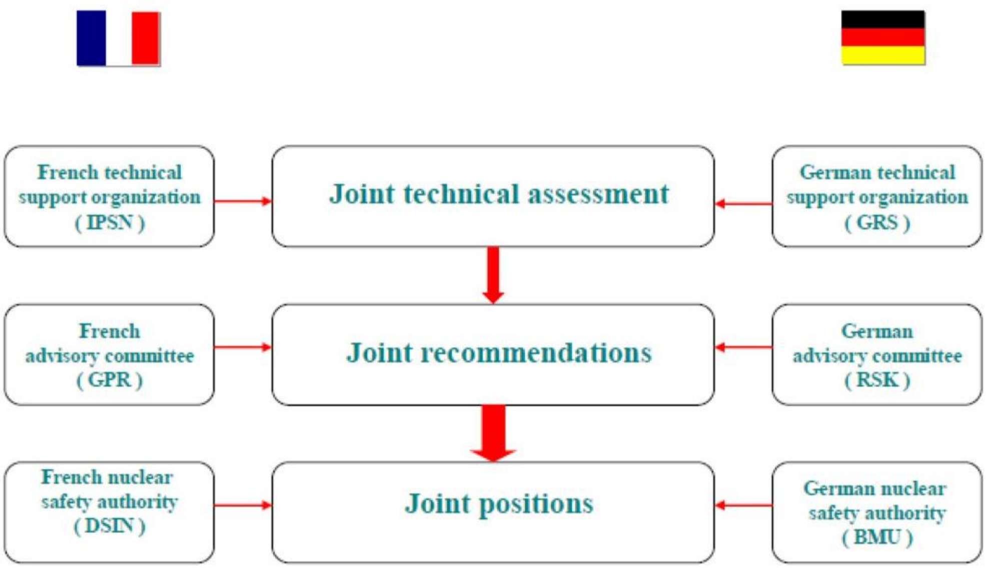

Figure 1: Technical assessment organization (till 1998)

The joint work took into account the experience from both countries. Both, France and Germany had in place a comprehensive set of laws, rules and regulations that had to be taken into account and harmonized where necessary. Figure 2, taken from the same source as Figure 1, presents the hierarchy of laws, rules and regulations which existed in both countries.

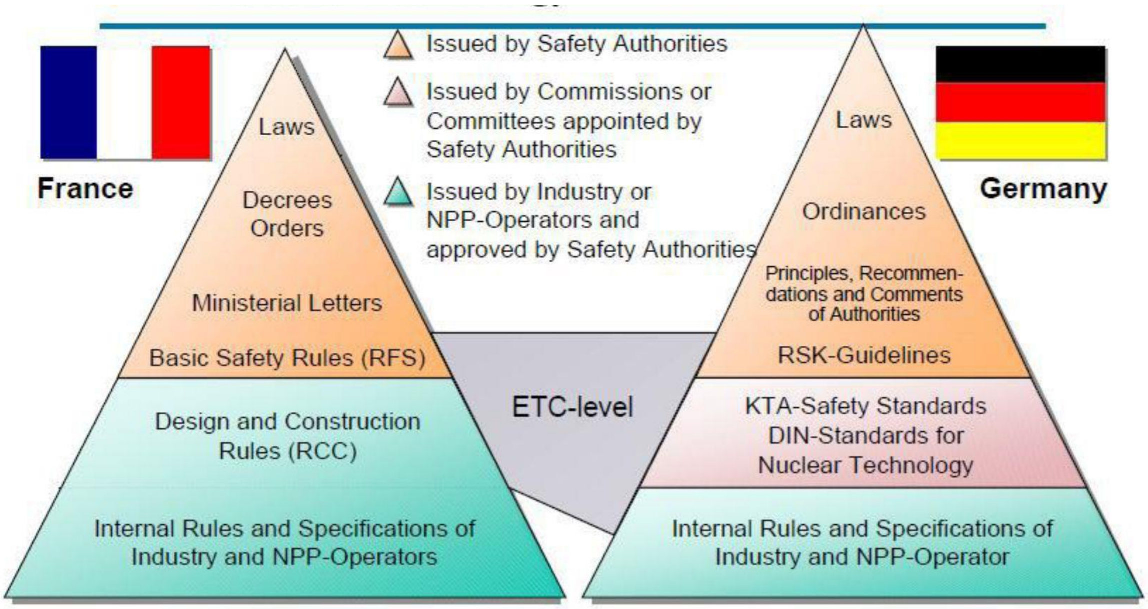

Figure 2: Hierarchy of rules and regulations related to nuclear technology

In late 1991, the German government adopted an energy policy which included the requirement that the consequences of a severe accident at a future NPP must be restricted to the NPP itself (introduced into the German Nuclear Energy Act in 1994 as section 7 para. 2a). This necessitated the development of new requirements for future reactors. The German RSK, in a session of February 1993, established first elements of such requirements which were subsequently fed into the consultations with GPR. As a consequence, three RSK/GPR ad-hoc working groups were formed, namely groups on:

Severe Accidents;

Materials;

Human Factor Engineering.

In 1993 the GPR and RSK issued their joint proposal – **Proposal for a Common Safety Approach for Future Pressurized Water Reactors**, which was important as it established the approach to be used in the future. The adopted philosophy was that the approach to nuclear safety should be evolutionary rather than revolutionary. This was to be applied both to the EPR design targets and in the EPR's conceptual design. The document was published in English. A German translation was endorsed by the RSK on 22 September 1993 and published, with the official date 8 November, in the German Federal Gazette (Bundesanzeiger) Nr. 218 of 20 November, with the German title "Gemeinsame Empfehlung von RSK und GPR für Sicherheitsanforderungen an zukünftige Kernkraftwerke mit Druckwasserreaktoren".

On 20 December 1993, the DFD requested GPR and RSK to continue and to amplify their deliberations on safety requirements for future NPPs, taking into account the solutions proposed by the industry. This was also proposed with a view to the development of technical criteria at the level of detail of the existing set of requirements (*RSK-Leitlinien*) in Germany.

Subsequently, the Common Approach of 1993 was followed up by a number of joint recommendations by GPR and RSK. On the German side, these were published in the Federal Gazette (Bundesanzeiger) in English language in two instalments:

A publication with official date 5 May 1995 in the Federal Gazette Nr. 127 of 11 July 1995 with the results of sessions Nrs. 11 through 16 (all in 1994) of GPR and RSK on the following topics:

- System Design and Use of PSA;
- Integrity of the Primary Circuit;
- External Hazards;
- Radiological Consequences of Reference and Low Pressure Core Melt Accidents;

- Severer Accidents.

A publication with official date 24 July 1997 in the Federal Gazette Nr. 185 of 2 October 1997 with the results of sessions Nrs. 17 through 24 (held in 1995-1997) of GPR and RSK on the following topics:

- System Design Issues;
- Secondary Side Overpressure Protection;
- Radiation Protection during Normal Operation;
- Primary Circuit Integrity;
- ECCS Design;
- Protection against Internal Hazards;
- Severe Accident R&D Needs;
- Design of Systems and Use of Probabilistic Safety Assessments for Future PWR Plants.

In 1993-1995, the safety authorities, with their TSOs and advisory groups, carried out a detailed assessment of the EPR design specification. They requested enhancements and changes. The outcome was an updated design specification (Main Feature File [MFF]) which presented the detailed design requirements for the EPR plant and the rationale for the options chosen.

The basic design was completed in 1997 and issued as the 1997 EPR Basic Design Report (**1997 BDR**). Additional studies followed to fine-tune the basic design concept. This resulted in a "basic design optimization" which was completed in 1999 and issued as **1999 BDR**. Already the first version of BDR proposed a site-independent, standardized Nuclear Island, and contained sufficient information to enable safety authorities to perform the safety assessment of the EPR. At that time, it was decided that the scope and content of the BDR should have such a level of detail that it could be used as an application document for obtaining the construction licence in Germany or France. This meant that the level of detail should be in compliance with the French and German regulations and should be equivalent to the standard part of the Preliminary Safety Analysis Report (PSAR) required in France. In addition the table of contents of the BDR also followed the US NRC Regulatory Guide 1.70 Rev. 3.

Generally, the sources suggest that it was a very difficult task to find a common approach between the French and the German safety approaches and between regulators and TSOs on the one side and industry on the other side. As it seems, the industry side was initially reluctant to embrace the Common Safety Approach, including for example the crash of a military airplane or a steam explosion. Finally, however, a compromise consensus was reached.

The above described harmonization of French and German licensing requirements derived from French and German applicable regulations were prepared in years between 1995 and 2000 by IPSN/GRS and published in October 2000 as **"Technical Guidelines for the Design and Construction of the Next Generation of NPPs with Pressurized Water Reactors"**. These guidelines *present the <u>safety philosophy and approach</u> as well as the <u>general safety requirements</u> to be applied for design and construction of the next generation of PWR nuclear power plants.*

The Technical Guidelines implemented and refined the EPR conceptual safety design features identified in 1993. Several steps were involved in their production:

The first step was to propose a structure allowing the GPR/RSK recommendations to be implemented in such a way that the technical guidelines can be used in both French and German regulatory processes. This was the objective of the work performed in 1997;

The second step was to organize the GPR/RSK recommendations in the adopted structure. This work was performed in the first part of 1998;

> In 1998, IPSN/GRS began to turn the text of the GPR/RSK recommendations into guideline text, rearranging and making some changes in the material (without changing the substances of the original GPR/RSK recommendations). An intermediate version of the Technical Guidelines was

issued by IPSN/GRS at the end of 1998. This document was reviewed by GPR/RSK members and by the EPR design organization, leading to the issue of an updated version in 1999.

The Technical Guidelines cover the following principles and conceptual safety features:

- The General Safety Principles which look at plant transient behaviour, redundancy, diversity and separation in safety systems, human-system interface, protection against internal and external hazards, use of probabilistic safety assessment and radiation protection of workers and the public;
- The Conceptual Safety Features which look at the design of barriers (all four barriers; fuel cladding, fuel matrix, primary system, and containment) and at the safety functions (control of reactivity, cooling and confinement) and safety systems;
- Accident Prevention and Plant Safety Characteristics;
- Control of Reference Transients, Incidents and Accidents;
- Control of Multiple Failures Conditions and Core Melt Accidents;
- Protection Against Hazards; and
- System design Requirements and Effectiveness of the Safety Functions.

The impact of the cooperation on national licensing process in Germany was evident. In Germany, the RSK/GRS Common Safety Approach was published in 1993, with additional recommendations in 1995 and 1997, in the Federal Gazette (see above) and therefore acquired the same status as RSK recommendations in general. This meant that they would have been not legally binding, but in practice of the utmost relevance for any new build licensing process.

In 1998, a new provision was added to the German Nuclear Energy Act (section 7c, "evaluation procedure"), creating the possibility of a kind of pre-licensing: upon request, the Federal regulator BMU could issue an evaluation of safety aspects of a design. This evaluation was to be published in the Federal Gazette. It was not legally binding on subsequent NPP licensing processes (which, in the German federal system, are done by *Länder* authorities) but in practice it would certainly have had an important impact. The explanatory statement of the government draft for the 1998 amendment expressly mentioned the Franco-German EPR as an object of application for the safety evaluation.

However, since the German side dropped out of the process after the elections of November 1998, Germany did not endorse anymore the Technical Guidelines of 2000 and since no new NPPs were built in Germany, these instruments were left unused. The Technical Guidelines have therefore never been officially adopted by the German regulatory authority.

On the French side, the Technical Guidelines were published by ASN in 2004 and used by ASN as a reference for the ATMEA safety options design review in 2010-2012 (see Task 3 of this Report, chapter 2.3).

Further to the Technical Guidelines, a set of detailed *industrial rules* common to the French and German nuclear industry were elaborated in the subsequent time period. The so-called **ETCs (EPR Technical Codes)** were based on the **French RCCs** and the **German KTA**-rules. Six documents under ETCs covered:

- Safety and process,
- Mechanical components,
- Electrical equipment,
- Instrumentation and control,
- Civil works,
- Fire protection,
- Common requirements for handling devices/ventilating.

As opposed to the above-mentioned “Guidelines” which are publicly available, the ETCs are not publicly available. Figure 3, taken from the same source as Figure 1 and Figure 2, schematically presents derivation of ETCs form French and German legislative framework.

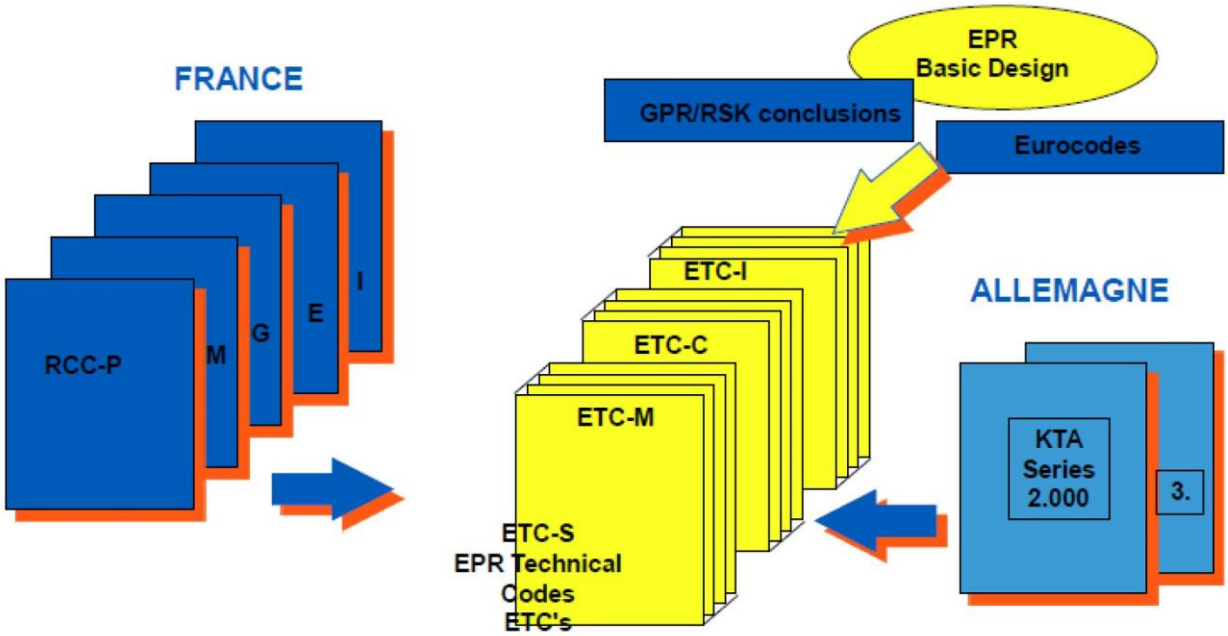

Figure 3: Need for common standards

ETCs were produced "by relying in certain cases on Eurocodes published at European level and by anticipating the distribution of "national appendices" designed to adapt these European standards according to the countries considered" (*UK EPR PCSR vol. 2 chapter B6*).

Generally, it seems that ETCs have been developed from existing RCCs and that, going in the opposite direction, in the last years they have been further developed and made more generic to become RCC codes again.

Below are given just examples of the ETC codes for civil works, Fire protection and mechanical components in order to demonstrate their applicability and the approach taken. It is out of scope of this report to go into details of each individual ETC Code. Our focus was more on the process of codes' development, experts involved and the mode of cooperation between two countries.

ETC-C

The EPR Technical Code for Civil works (ETC-C) was initially developed for EPR safety classified structures by EDF and German Utilities.

The ETC-C contains rules for the design, construction and testing of the EPR civil engineering structures. It describes the principles and requirements for safety, serviceability and durability conditions for concrete and steelworks structures on the basis of Eurocode design principles (European norms for structures) together with specific provisions for safety-class buildings. The section covering “design” lays down the essential actions and requirements by which buildings and structures are engineered. (*Source: UK EPR PCSR 3.8*)

ETC-F

The EPR Technical Code for Fire Protection (ETC-F): the ETC-F contains requirements for fire protection (prevention, detection and suppression) in the design of the EPR Plant. It is based on the existing French RCC-I code (June 92 issue) applicable to current French nuclear plants, on French AFNOR standards and the relevant German KTA-Safety Standards (applicable to German plants). ETC-F is the result of a long process of development using experience from the construction program of French and German nuclear plants including feedback experience from fire events occurring in these units. ETC-F states requirements for fire protection but does not propose design solutions to meet these requirements.

The third step (after two steps of improving French codes) was to merge French and German practices in a common and harmonised approach in consultation with French and German Safety Authorities. This process took place over approximately ten years coinciding with the different phases of the EPR project.

ETC-M

An important code for NNPs would have been ETC-M for mechanical components. There are indications that an ETC-M had been envisaged but that this project was later merged into the revision of the RCC-M (*Source UK-EPR PCSR chapter vol. 2 subchapter B6*).

ETCs were partly used for the French EPR project at Flamanville. They would also have been applied in Germany if an EPR had been constructed there. The UK nuclear regulator ONR also made use of the ETCs when assessing the EPR design in the framework of the Generic Design Assessment for the UK EPR Project. Therefore, if circumstances had permitted, it can arguably be reasoned that the ETCs would have been the common basis for a fleet of EPRs in France and in Germany.

In subsequent years, in the early 2000s, the work of NPI slowed down due to the decision of the German Government (Social Democrat and Green party coalition) for the nuclear phase-out. Siemens merged its nuclear business in 2001 with Framatome and a new company Framatome ANP was established. German utilities also withdrew from the EPR project by 2002, even though German experts continued to cooperate with the French experts on individual basis. Former members of the RSK were requested by DSIN to continue as “German experts” to participate in GPR meetings dealing with EPR related issues.

2 HOW WAS THE WORK PERFORMED AND THE RESULTS ACHIEVED

Description of the process that took place throughout the Franco-German cooperation was solicited from several experts involved in the process at that time. One of them, Mr. Helmut Schulz, who was personally involved in this activity, was member of the GRS for 36 years. Among other things, he was heading the Department of Component Integrity and was acting as chairman for the Standards Committee of Nuclear Components. He was member of the German Reactor Safety Commission (RSK) and served as an expert in the development of the common French-German safety requirements for new reactors based on the EPR basic design documents. As such, his first-hand experience in this project has given us the insights of how the work has been performed and results achieved, describing positive aspects and challenges during the process as well as the lessons learned.

Below is outlined the experience of experts involved with the process which led to the establishment of the "Technical Guidelines for Future PWR's and EPR Technical Codes", how the output was achieved and how the TSOs and the regulators come up with a set of codes which were acceptable to both sides.

As a more or less normal approach in a challenging long term project a step wise procedure for the performance of the Franco-German project was selected.

Schedules to guide the work and planning of meeting dates where done on an annual/biannual frame. This included interactions between all parties to achieve a coherent time table like: dates to prepare and present documents by the industry, review by TSOs and time to develop common positions and common reports, GPR/RSK preparatory and plenary meetings to prepare recommendations to German/French regulatory authorities.

At the TSO's level, rapporteurs were named for technical issues and areas.

Step 1: Development of common Safety Objectives

At the very beginning of the project, it was recognized that any cooperative activity and future design need to be oriented by safety objectives which will need to be fulfilled. In the late 1980's (after the Chernobyl accident) a broader discussion started to expand the safety concept for existing "Light Water Reactors" (LWR) and specifically developing evolutionary designs for future LWR's.

Therefore the first activities of the common work by the French and German safety organizations were the development of safety objectives. A draft was prepared by GRS and IPSN, submitted for joint review by the French and German safety authority's expert groups, (GPR, RSK). After amendments, these groups submitted a common proposal to the authorities, which adopted it in June 1993.

The updated safety objectives aimed at a decrease of large early release of radioactive elements in case of severe accidents by a considerable factor (at least 10 or even 100) compared to previous designs. The language used in the guidelines remained largely in deterministic terms. These changes needed to be reflected by the safety requirements and codes & standards.

In September 1993 a Conceptual Safety Feature Review File (CSFRF) was submitted by the EPR project team to the French and German safety authorities.

Step 2: Derive common views on essential issues

At this step, the position on a limited number of essential issues, with a potential strong impact on the development of the project was précised by the safety authorities based on GPR/RSK recommendations. The work involved for all issues followed in general a scheme like:

- questionnaire developed by IPSN/GRS given to EPR-project team

- answers of the project team by reports or other type of documents
- common technical meetings to achieve clarifications
- further review by IPSN/GRS and drafts of technical positions of the TSO's (in cases of remaining controversial preliminary positions they have been documented with related technical arguments and forwarded for further broader discussions to GPR/RSK)
- final clarification meetings (partly with GPR/RSK subgroups)
- common report on the issue by IPSN/GRS going to GPR/RSK
- GPR/RSK subgroup review meetings followed by plenary meetings
- Common position of GPR/RSK and advice to DFD
- DFD meetings to finalize position on each issue
- Final decisions regarding all issues by the regulatory authorities (DSIN/BMU) on all issues and publication. In France: Federal Gazette; in Germany: Bundesanzeiger until autumn 1995.

The process to achieve clarification on essential issues took about 18 month with a rather intensive use of resources of all actors involved.

Step 3: Review of the "Basic Design Report"

The following step in the whole process was the initiation of the basic design phase by the EPR project and the submission of the "Basic Design Report" and the "EPR Technical Codes" (ETC) to the French and German safety authorities in October 1997.

The work scheme for the review of the "Basic Design Report" files followed more or less the scheme applied before for the clarification of essential issues. The bulk of the review work was with the TSO's; their reports are then taken up by GPR/RSK meetings.

Step 4: Development of the "Technical Guidelines for future PWRs"

The work started in 1997. IPSN/GRS prepared draft documents for GPR/RSK/German experts meetings. Based on intensive discussions at a series of meetings the "Guidelines" were finalized in 2000.

At the start of the common work on the development of the EPR in 1989:

- a pronounced interest was there at the industry side to develop a common future design;
- the experience and capabilities of all actors stood at a high level;
- the majority of professionals involved have known each other from various activities.

The French and German regulatory side decided to support as their common activities also a close cooperation of the TSOs (IPSN and GRS) and the advisory bodies (GPR and RSK). IPSN and GRS had already a long standing cooperation. In addition, the common work related to the EU funded projects in the 90ies with the goal to enhance nuclear safety for WWER units was a basket of experience of challenging exercises to develop common positions on safety issues beside differences in safety requirements and safety standards.

The principle step by the authorities to enlarge the safety umbrella opened the gate for constructive discussions to reassess present requirements and work out new or refined common positions how requirements should be changed. It certainly took some extensive discussions to assure that all experts involved achieved a common understanding on the terms and definitions used in the formulation of the "common safety objectives" as well as the common positions on "essential issues". Professional translation was used in the plenary meetings providing assistance. Without the long standing expertise of most of the actors the process of these two steps but also the following ones would have taken much more time.

Taking these conditions, there was no need to establish procedures to guide the work, actors worked together without formalized procedures. Any divergent views which resulted in the working/review process of IPSN/GRS were taken as written documents for further discussion to the GPR/RSK meetings.

Regarding the ETCs the experts reviewing the draft documents were mostly personally engaged in the codes & standards development within national organizations, and therefore very familiar with the technical areas.

Step 5: Review of ETCs

Regarding the work on the ETCs the work scheme was similar to the first couple of steps shown above but not being followed by the final approval of GPR/RSK and the safety authorities due to a change in the organization of the Franco-German cooperation. Besides GRS and IPSN additional expertise was included in the review of the ETCs with experts from TUEV (German) and BCCN (French) to insure a broad coverage of experience and responsibilities. The results of the ETC's reviews by IPSN/GRS were documented for each ETC in a review report covering the work performed, review results and recommendations.

At the German side there was no formal interaction with the responsible safety standards organizations like KTA, DIN. Nevertheless the majority of the actors involved in the ETC work were personally also active in the KTA and DIN/EN/EUROCODE developments as well as in the commissions guiding safety related R&D Programs, which were partly related to safety standards. The French and German safety standards/documents which were related to the ETC have not been revised at the time.

3 HOW DID THE TECHNICAL SUPPORT ORGANIZATIONS (TSO'S) AND THE REGULATORS COME UP WITH A SET OF CODES WHICH WERE ACCEPTABLE TO BOTH SIDES, IN SPITE OF WELL-ESTABLISHED AND DETAILED CODES BEING IN EXISTENCE IN FRANCE (RCC) AND IN GERMANY (KTA)?

In France the "Basic Safety Rules" (RFS) are issued by the French regulatory authority (previously DSIN, today ASN) and the technical standards "Design and Construction Rules" (RCC) are issued as standards of the French nuclear industry by AFCEN.

In Germany the "Safety Criteria" and further guidelines are also issued on a governmental level. The technical safety standards "KTA" are established in a rather unique process where manufacturers, vendors, operators, independent safety experts (TSOs), licensing authorities and representatives of other organisations like research centres, trade unions, insurance companies work together organized in 6 interest groups and decide on the rules with a majority of 5 out of 6 votes.

In both countries final decisions remain with the licensing authority, the set of available rules and safety standards has to be seen as a frame for guidance.

Looking to the hierarchy of rules and regulations in both countries, the Guideline for the future PWR would be placed below the levels of the laws. The ETCs are interacting with the levels of basic design and construction rules, guidelines and technical standards.

As lined out in the previous section, the principle step by the authorities to enlarge the safety umbrella opened the gate for constructive discussions to reassess present requirements and work out new or refined common positions how requirements should be changed.

The inclusion of core meltdown sequences and means to mitigate related consequences were not covered by existing requirements and codes & standards regarding the overall safety concept and design. Nevertheless some system related safety improvements and procedures for accident management were already implemented in existing plants. Therefore most actors involved had good technical background in these areas. In essence for this new established part there was no conflict of interest with the previous documents.

The intention of the French and German regulatory authorities for the development of the "Guidelines" was to use this for future PWRs. The work of the TSOs and GPR/RSK was focused to establish a coherent set of requirements for an enlarged safety umbrella not being bounded by questions regarding compatibility with existing NPPs.

Regarding the ETCs, certainly all the available experience from manufacturing, operating units and previous safety cases have been taken into account to decide on improvements and clarification of requirements. The ETC-M left room (as it should always be the case) for different technical solutions to fulfil the requirements. As an example: If an EPR would have been built in Germany the primary piping would probably be using ferritic steel with austenitic cladding and not fully austenitic piping as usually being the case in France. The pre-stressed containment structure for the containment as covered in ETC-C had no equivalent KTA rule at the time and the EUROCODE was an accepted reference document.

> In summary a satisfactory agreement on the safety objectives, the overall requirements as lined out in the "Guidelines" and the ETCs have been achieved on a technical level but due to a change of the position of the German Federal Government the results of the common work have not been endorsed by the German regulatory authority due to the decision of the Federal Government on phase-out of nuclear energy.

Nevertheless the strong commitment of the main actors contributed considerably to the success of the common work. As lined out earlier, the conditions at the start of the common work regarding the capabilities and experience of all actors involved were very favourable which certainly had an impact on the success of this cooperation.

The EPR project was very helpful to keep the discussion of all actors on the requirements on future PWRs focused. A more general debate on requirements for future PWR's without reference to a preliminary design concept may have had difficulties to achieve common positions in a similar time frame.

Regarding the new elements (design which copes with core melt-down accidents) and the possibility to demonstrate an acceptable safety case it was very beneficial that some of the experts at the TSOs side as well at the advisory groups (GPR/RSK) were closely engaged in safety research projects covering experiments and analytical code developments.

Professional translations to provide assistance in the technical discussions (mainly at plenary meetings) have been helpful in two ways: to improve common understanding of terminology as well as slowing down emotions in the debate which are partly connected to misunderstanding of terminology of the English language by the French/German experts.

As a general experience in bi - or multinational meetings/discussions a well-structured but patient and friendly chairmanship paired with an adequate level of hospitality taking note of national habits are supportive to avoid unnecessary tensions and to achieve results. This was the case during the whole period of the common work.

Real proofs are outstanding to what extent the established documents are satisfactory for application in different regulatory environments. Unfortunately the final results have not been adopted by the German regulatory authority due to a change of the federal government. For the OL 3 project in Finland the regulatory authority defined their frame work based on the regulation in Finland and adopting RCC, KTA or ASME code rules as specified and reviewed.

4 FEASIBILITY STUDY

Could the process be replicated today, for any given pair or group of regulators and any given design, and under which boundary conditions it could succeed?

It is always a challenge to analyse whether a 20-year old process could work again today under fundamentally changed circumstances. Quite obviously, the Franco-German cooperation succeeded in its time in an environment which is different from today's situation. An analysis aimed at discussing a possible replication should therefore focus on those issues which are still relevant today, and at the same time evaluate whether conditions which do not exist anymore are perhaps offset by other, modern developments.

1. Specific conditions and circumstances of the Franco-German cooperation

In order to explore the possibility of replicating the Franco-German efforts, the main attributes that contributed to the success of this collaboration are summarized below. These could play a very important part also in possible future collaborative projects between two or more countries involved if they can be reproduced; otherwise, they may need to be adapted or replaced by something equivalent.

- *Strong support by governments*

The Franco-German process was triggered by industry but in parallel supported by politics at the highest level. In a joint government declaration of 6 June 1989, France and Germany agreed on a close cooperation in the field of nuclear safety. In particular, a joint safety directorate (Deutsch-Französischer Direktionsausschuss or German-French Directorate, DFD) created by the regulatory authorities ASN and BMU was tasked to evaluate how the safety requirements of both countries could be harmonized with a view to a joint safety philosophy.

For any replication, such strong political support seems essential. The governments of participating countries should make the cooperation, with all its advantages in terms of saving time and resources and of sharing experience and knowledge, part of their nuclear energy development programmes. There should be at least a joint declaration at the occasion of a summit, or perhaps even a formal agreement between the governments. If the process were left to industry and to the regulators alone, a decisive impetus would be missing.

- *Role of industry*

In the beginning of 90is there was a profound interest from the industry side in Germany and France to develop a common future design that would contribute towards easier and quicker licensing process in different countries. This aspect does not exist anymore when it comes to large reactors; the situation where several European countries had their domestic nuclear industry with their own reactor designs has profoundly changed. Currently, there is only one reactor vendor in Europe, namely Framatome with its designs EPR and ATMEA (the latter in cooperation with Mitsubishi Heavy Industries). Therefore, any replication of the Franco-German process would rather have to concentrate on two or more countries wishing to implement a foreign design which already exists. The impetus provided, in the Franco-German case, by the domestic nuclear industries of the two countries would be replaced by the desire of a foreign vendor to achieve project development and licensing of his standardised design in the participating countries, with as little changes as possible due to national regulations (see also Task 3). This would have to be taken into account.

A situation more similar to the Franco-German cooperation could arise, however, with Small Modular Reactors (SMRs). Currently SMR designs are developed by a range of industrial players in various countries, mostly, however, outside the EU (developments in the UK will not be credited here, due to the Brexit). Here, at least in theory, it could be envisaged that the industry in two or more EU countries cooperates in SMR design development, even if today there seems to be no concrete example.

- *Position of regulators and TSOs*

The experience with nuclear technology in France and Germany was at a very high level. Governmental organizations, regulatory bodies and technical support organizations were well established in both countries. Sets of laws, regulations, codes and standards already existed in both countries for a long period of time. Altogether, it can be stated that the Franco-German cooperation was cooperation between two partners being on equal level regarding technical and administrative nuclear expertise of all actors involved (vendors, manufacturers, utilities, TSOs, regulatory authorities).

The majority of professionals involved in this joint project knew each other from numerous previous activities. At about the same time, a large European efforts were devoted towards enhancement of safety of WWER (and RBMK) reactors in Eastern Europe and most of French and German experts involved in the EPR project were also involved in the WWER projects and have known each other from that time.

Experts from both TSOs (IPSN and GRS) as well as from the advisory bodies (GPR and RSK) were closely engaged in safety research projects covering experiments and analytical code developments and have also known each other from those endeavours.

In today's situation, it would be difficult to envisage a comparable cooperation of two or more regulators who would be strong enough to do the work independently. Rather, as the OECD-NEA's MDEP process has shown, the rationale for regulators to work together in reactor licensing would be different. Small and/or inexperienced regulators would participate in order to profit from information and evaluations provided by their larger and more experienced peers. Even the more established regulators would see a major reason for cooperation in saving resources in terms of their own expertise and the expertise of their TSOs – resources which are definitely much more limited as compared to the situation of French and German authorities in the 1990s.

By contrast, when it comes to personal acquaintance and working relationships, it seems that the joint work in institutions such as WENRA, ENSREG, MDEP, IAEA and the OECD-NEA today is a good basis for regulators to meet regularly and to know each other. This aspect is perhaps even more pronounced today than it was in the Franco-German project.

- *Technical approach*

At the beginning of the project a decision was taken that the approach adopted would be evolutionary rather than revolutionary.

Several safety related improvements in systems and procedures for accident management had already been implemented in several nuclear power plants, so there was already certain experience in place which gave involved experts solid technical background for the new project.

The project did not try to address the compatibility of the already existing nuclear units with the new, enhanced requirements.

At least in Germany, the process was favoured by the fact that the new reactor safety requirements had to be developed anyway, due to the fundamental political and legislative decision (1991/1994) that the consequences of a severe accident must be restricted to the plant itself.

In some cases, Franco and German safety philosophies deviated largely and large efforts were needed to harmonize them as for example, France accepted the crash of a military airplane and Germany accepted spent fuel pools being situated outside the containment.

The technical approach taken during the EPR project is described in more detail in the main body of this report and can be summarized in five steps:

Development of common Safety Objectives

Derive the common views on essential issues

Develop and review the “Basic Design concept”

Develop and review the “Technical Guidelines for future PWRs”

Develop and review the ETCs

Today, the technical approach, particularly the difficult task of reconciling two diverging safety philosophies in reactor safety, would not play the same role. This is due to the very different boundary conditions: a process today would not deal with the development of domestic reactor designs in participating countries, but would rather apply to a finished design from a foreign vendor considered for implementation in the participating countries. What is more, evaluation of the design would be done according to safety requirements which, in the past twenty years, have undergone a large extent of harmonisation (see below). The overall approach to new reactor designs has been established by the Nuclear Safety Objective in Art. 8a of the revised Nuclear Safety Directive 2009/71 as amended in 2014.

Hence, in this respect the current situation in Europe is more favourable to a common approach and some elements of the Franco-German cooperation which were necessary at that time would not have to be replicated.

With SMRs (as mentioned above), the need for a common understanding of safety requirements could arise again to a higher extent. Within the general philosophy of a graded approach of safety evaluation (see IAEA GSR-4, Safety Assessment for Facilities and Activities), SMRs with their limited nuclear inventory and with inherent safety features pose the issue whether requirements developed for large NPPs can be fully applied to them. Also, the concept of factory-built modules assembled on site opens new issues for inspection and oversight. Here, a common understanding and, if necessary, the joint development of new sets of requirements could become essential.

<u>Work conditions and organisation of the cooperation</u>

The work conditions at the beginning and during the cooperation efforts have been described in the main body of this report. Here, we can again summarize the main points that will enable us to establish the conditions for a possible future cooperation of two or more countries in a possible collaborative effort:

The sources suggest that it was a very difficult task to find a common approach between the French and the German safety approaches and between regulators and TSOs on the one side and industry on the other side. As it seems, the industry side was initially reluctant to embrace the Common Safety Approach, including e.g. the crash of a military airplane or a steam explosion. Nevertheless the strong commitment of the main actors contributed considerably to the success of the common work. The conditions at the start of the common work regarding the capabilities and experience of all actors involved were very favourable which certainly had an impact on the success of this cooperation.

As described above, a helpful element of the Franco-German cooperation was that many experts had previously been engaged in safety research projects covering experiments and analytical code development. During the meetings, professional translation and a well-structured and at the same time patient and friendly chairmanship were essential to achieve effective handling of issues and to prevent misunderstandings based on language or philosophy issues which often lead to emotional reactions and to friction in the cooperation.

These elements would definitely be essential for a similar process today. One of the criteria for appointment of participating experts from the national regulators and TSOs should be whether they have previously been engaged in international institutions and working groups; happily, this is the case today for many experts due to the work of IAEA, OECD/NEA, ENSREG, WENRA and other institutions, as mentioned above. Professional translation needs to be provided, and meetings must be organised in a way that all regulators can regularly perform the role of hosting the meetings.

Under today's circumstances, consideration should be given whether a secretariat created for that purpose would be necessary to ensure effective and efficient working conditions. The bilateral Franco-German cooperation was organised by the regulators and TSOs themselves. However, if today a group of more heterogeneous regulators would cooperate, a standing secretariat function would seem essential to further the work and to allow regulators and TSOs to concentrate on their work. The EU Commission could play a facilitating role in this (as suggested in Task 3 of this Report).

Safety requirements used for design assessment

Since the beginning of the Franco-German cooperative project on EPR almost 30 years have passed and a number of important developments in the area of nuclear safety have been achieved in the meantime, which would considerably influence the possible future cooperation between or among several countries if they were to undertake similar collaborative approach. The most important progress has been made in the development of international safety standards:

EU amended Nuclear Safety Directive:

> The amendment came as after the Fukushima Daiichi accident the European Council called on the Commission to review, as appropriate, the existing legal and regulatory framework for the safety of nuclear installations and propose any improvements that may be necessary. The Council Directive 2014/87/Euratom of 8 July 2014 brings the amendment to the Directive 2009/71/Euratom which established a Community framework for the nuclear safety of nuclear installations (it is commonly referred to as the amended Nuclear Safety Directive). The original Council Directive 2009/71/Euratom imposes obligations of the Member States to establish and maintain a national framework for nuclear safety and reflects the provisions of the main international instruments in the field of nuclear safety, namely the Convention on Nuclear Safety, as well as the IAEA Safety Fundamentals.

IAEA Safety Standards:

> The IAEA Safety Standards have three layers; Safety Fundamentals, Safety Requirements and Safety Guides. The IAEA Safety Fundamentals consist of 10 safety principles. Under the Safety Fundamentals, the IAEA Safety Requirements are produced, with the objective to set up the requirements which are necessary to fulfil 10 safety principles. The IAEA Safety Requirements are divided into two parts; General Safety Requirements (GSR) and Specific Safety Requirements (SSR).
>
> There are 7 General Safety Requirements; Part 1: Governmental, Legal and Regulatory Framework for Safety, Part 2: Leadership and Management for Safety, Part 3: Radiation Protection and Safety of Radioactive Sources, Part 4: Safety Assessment of Facilities and Activities, Part 5: Predisposal Management of Radioactive Waste, Part 6: Decommissioning and Termination of Activities and Part 7: Emergency Preparedness and Response.
>
> Specific Safety Requirements comprise of 6 volumes: NS-R-3 (Rev 1): Site Evaluation for Nuclear Installations, SSR-2/1(Rev 1): Safety of Nuclear Power Plants: Design, SSR-2/2 (Rev 1): Safety of nuclear power plants: Commissioning and Operation, SSR-3: Safety of Research Reactors, NS-R-5 (Rev 1): Safety of Nuclear Fuel Cycle facilities, SSR-5: Disposal of Radioactive Waste and SSR-6: Regulations for the Safe Transport of Radioactive Material.
>
> Underneath of the above Safety Requirements there is an entire fleet of corresponding Safety Guides which provide guidance on how to fulfil the corresponding higher level requirements.

WENRA Reference Levels report and Safety of New NPP Designs report:

> WENRA, the association of the Heads of nuclear regulatory authorities of the EU countries with nuclear power plants on their territories plus Switzerland has developed the Safety Reference Levels as a common approach and commitment to continuous improvement of nuclear safety.
>
> The objective of the development of Safety Reference Levels was to increase harmonization within WENRA countries on safety requirements issued by the regulatory bodies and their implementation in existing nuclear power plants. Initially they identified 18 areas where harmonization was considered as necessary. After the Fukushima Daiichi accident an additional 19th issue has been included to cover natural hazards (extreme weather conditions, external flooding, and seismic events). All together the document includes about 300 "Reference Levels".

In addition, WENRA has prepared another report that presents WENRA expectations for the design of new NPPs. These expectations are defined in addition to the requirements set in the IAEA Safety Requirements SSR 2/1 (Rev 1) on Design of NPPs.

European Utility Requirements:

Within Europe, the development, design and licensing of NPPs has been performed on national basis with little interaction between countries. In order to overcome this weakness, the major utilities in Europe have joined forces in 1991 and formed an organization to develop the European Utility requirements.

The European Utility Requirements consist of 4 Volumes. Volume 1 presents the main objectives and summarizes the main requirements. The Second Volume contains a set of generic nuclear island requirements. It covers most of what a plant owner has to specify for the assessment, licensing, design, supply, construction, test and operation of a NPP with light water reactor. Volume 3 includes evaluation of selected light water reactor designs against the EUR requirements. Volume 4 is a set of generic requirements for the power generation plant organized by chapters that deal with the specific systems.

Apart for the above described international safety standards, several countries have also decided to join their effort in the so called MDEP (Multinational Design Evaluation Programme) under the auspices of the OECD/NEA (see the description of MDEP process in Task 3 of this Report, chapter 3.2).

In the last 30 years also national regulators have developed much stronger and precise requirements that must be complied with if a new NPP is to be considered.

Hence, in comparison to the situation of the Franco-German cooperation which started 30 years ago, it can be stated that alignment of safety standards has gone a long way, worldwide and particularly in Europe. A major part of the Franco-German process, namely the development of common requirements (GPR/RSK Proposal for a Common Safety Approach for Future Pressurized Water Reactors, 1993; Technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors, Adopted during the GPR/German experts plenary meetings held on October 19 and 26, 2000), would today be shorter and less difficult. Starting point would not be two very diverging sets of national regulations and standards, but a largely uniform understanding of high-level requirements as shaped by the EU Safety Directive, ENSREG, WENRA and IAEA Safety Standards. This does not obliterate the need for an agreement on a common set of requirements (as explained in more detail below), but it makes this step easier. On this basis, the regulators could focus on agreeing on requirements on the detailed implementation level by agreeing on a common set of standards. There would also have to be a reconciliation (if appropriate) of different national "philosophies" and "schools of thinking" which have evolved over decades in experienced nuclear countries, as well as an agreement on assessment methodologies and on the way TSOs deal with reactor design evaluation.

Codes and standards

A basic element of the Franco-German cooperation was the development of joint codes (the ETCs). As described above, this work progressed in a promising matter but was eventually cut short by the German exit from the collaboration. The results were not lost and the ETCs were gradually used for the further development of the French RCC codes.

The situation of the Franco-German cooperation, with two countries with their own national nuclear codes and standards working together, seems unlikely to arise in the same constellation today. Rather, it can be presumed that several countries work together which do not have their own set of nuclear codes and standards. This was, and is, generally true for mid-sized or small nuclear countries and particularly for newcomer countries. In such a situation, generally the codes and standards of the vendor country of the reactor design are used if they are found to be compatible with national safety requirements.

Therefore, the focus today would probably shift from an alignment of national codes by creating a dedicated new set of codes, merging the national approaches, to an assessment of foreign codes and whether they can be used for deployment of the reactor design in the participating countries. With this changed focus, experience from the ETCs can definitely be utilised.

With SMRs, the situation could become particularly interesting as their concept of factory-built modules definitely requires acceptance of the codes and standards of the vendor country.

2. Steps for replication of the Franco-German cooperation

All these new developments in safety regulation which did not exist to that extent 30 years ago will have to be taken into account in any future joint ventures. If one would aim at replicating the Franco-German effort the following steps are suggested to be followed:

A stable commitment of all involved, vendors, TSOs and regulators from both (all) sides to such joint project will have to be established and sustained throughout the duration of the project.

At the very beginning of such joint project, common safety objectives and as far as possible common safety requirements would have to be established among the participating parties. Even if the common safety requirements are not there at the beginning of the project, it would be necessary to make clear that development of them is the project priority as all the rest would inevitably depend on the outcome of such harmonization.

Before the work on the development of common safety standards and requirements begins, the following issues would need to be addressed as a minimum:

- Define precisely the area and objectives of common work
- Define duties and responsibilities of all stakeholders involved
- Clarify compatibilities and non-compatibilities of existing regulatory frameworks
- Clarify the existing structure of all organizations involved (TSO, advisory, regulatory) in all countries involved
- Define the level of professional capabilities and experience for the experts involved
- Evaluate possible impact of different culture/mentality/language in the interaction of the experts
- Establish clear rules and procedures for interaction, communication and documentation

A careful comparison of national regulation of the country on who's soil the NPP would be built (as it is always their regulation that applies) with the above described fleet of international standards in order to extract any possible discrepancies. If the project was to aim at certifying the design that could be used in various countries (as was the case in the Franco-German effort), design compliance with the international standards should be the focus of this step.

In case that the site is already known, that would make collaboration easier as it would be clear which site-specific regulation will have to be applied.

The most probable case of future cooperation that would benefit from the Franco-German experience will most probably involve a country with a more or less complete design being offered to a third country, or two or more countries wishing to implement a reactor design from a foreign country. In these cases we would not have two countries developing together an evolutionary design based on their existing designs and Franco-German experience would not apply with its entire value. Some elements however could still be applicable.

Next step would be to establish which capabilities exist in each country being involved in the joint project. In the case of France and Germany, they both had parallel structures as depicted in Fig. 1 at the beginning of this report. Both countries had well developed TSOs, advisory committees and regulatory bodies. Today, there may be an asymmetrical setup if the structures on both sides are different. This may occur if the structure in one country is similar to that in Germany and France, with a strong institutionalised

TSO, whereas in a second country the expertise is concentrated in the regulatory body which only occasionally avails itself of topical support by TSOs, so that experts of the regulatory body of the second country would collaborate with TSO staff of the first one. If a parallel structure does not exist, it should be investigated how to overcome such situation. Today an organization that unites major European TSOs exists in the form of ETSON and its role and impact would have to be assessed.

Regarding codes and standards (at the ETC level) to be applied for design and construction there are already considerable similarities among major vendor countries. For example for large metallic components like pressure vessels, pipes, valve bodies there are large similarities among ASME (USA), RCC M (France) and KTA (Germany). Some important differences nevertheless exist in for example the area of qualification of manufacturers and constructors, role of third party inspectors, regulatory oversight or flexibility of the assessment of deviations from requirements set in the standards. All these issues would need to be carefully examined during the possible joint project.

The Franco-German cooperation had two levels;

- Development of principles and conceptual safety features in the form of the "Technical Guidelines for the Design and Construction of the Next Generation of NPPs with Pressurized Water Reactors". They covered:
    - The General Safety Principles which look at plant transient behaviour, redundancy, diversity and separation in safety systems, human-system interface, protection against internal and external hazards, use of probabilistic safety assessment and radiation protection of workers and the public.
    - The Conceptual Safety Features which look at the design of barriers (all four barriers; fuel cladding, fuel matrix, primary system, and containment) and at the safety functions (control of reactivity, cooling and confinement) and safety systems.
    - Accident Prevention and Plant Safety Characteristics,
    - Control of Reference Transients, Incidents and Accidents
    - Control of Multiple Failures Conditions and Core Melt Accidents
    - Protection Against Hazards and
    - System design Requirements and Effectiveness of the Safety Functions
- Development of EPR Technical Codes (ETCs). Six documents under ETCs covered:
    - Safety and process,
    - Mechanical components,
    - Electrical equipment,
    - Instrumentation and control,
    - Civil works,
    - Fire protection,
    - Common requirements for handling devices/ventilating.

It would be recommended that similar approach would take place also in the possible future project, whereas the first document might not be necessarily developed from scratch as in the meantime tremendous developments have been made on the international level as described above and these would obviously have to be taken into account.

Similarly, European Utility Requirements would necessarily have to be consulted before embarking on the development of codes similar to ETCs as it might not be necessary.

5 STEPS TO BE TAKEN TO REALIZE THE FULL STUDY

The feasibility study described in chapter 4 of this report has shown, that it would be worthwhile to undertake a full study with the aim of setting out the conditions for possible future cooperation between two or among several countries to establish a common grounds for designing, licensing or constructing selected nuclear power plant technologies.

If such study will be undertaken, several steps would need to be investigated. These steps have been selected based on the experience of the Franco-German project and can be summarised as follows:

Step 1: Investigation of the status of the nuclear programs and regulatory frameworks in participating countries

The Franco-German project was cooperation between two partners being on equal level with regard to technical and administrative competencies on all levels; vendors, manufacturers, utilities, TSOs, advisory bodies, regulatory authorities etc.). There has also been a long standing cooperation also in the past on different levels. This might not be the case in other collaborative project and it has to be investigated and the status clearly established.

Countries may organize licencing of their nuclear installations in different ways in accordance with their governmental structure and their traditions and needs. To define the area of cooperation it is important to identify the principle schemes of licencing, the interfaces to other authorities within national regulatory and legislative framework and other country specific rules and traditions. The objective of this preparatory work would be to identify areas where the partners in the project could cooperate directly with their existing human resources and to identify those areas where additional actors would need to be brought into the project. In some instances, some issues could be left to the individual national licensing practices as was the case in the Franco-German project where the role of the third party inspection was included in the KTA safety standards in Germany but not in the French ETC-M.

In this respect it has also to be investigated how are the technical safety standards being developed in participating countries. In the case of Germany for example it was manufacturers, vendors, operators, independent safety experts, technical support organizations, licensing authorities, research centres, representatives of trade unions, insurance companies and general public that were involved in the development and approval of these standards.

The final decisions normally remain with the licensing authorities as was (is) the case in France and Germany. It however might not be always the case, especially in some non-EU countries and the role, strength and independence of the licencing authorities have to be well and early established for all participating countries. At the same time it will have to be determined if the comprehensive set of laws, regulations, codes and standards already exist in all (both) participating countries.

Step 2: Compatibility check of basic principles and safety objectives set in relevant national legislations

The next step would be to investigate how compatible are basic safety principles and safety objectives in participating national regulatory frameworks. The hierarchy of laws, rules and regulations and technical codes would need to be compared in order to establish common grounds for future work.

The legislation of countries may differ in the way how the safety objectives are formulated. Some countries accept explicit probabilistic values, others stay with the descriptive terms such as physically impossible, very unlikely etc. Therefore it would be necessary at the beginning of the project to achieve clarification on the way how safety objectives have to be formulated.

Before a common work would start, a consistent set of safety requirements would need to be established in all participating countries. Even if the common safety requirements are not there at the beginning of the project, it would be necessary to make clear that the development of such common safety requirements is the project priority as any further steps would depend on the outcome of such harmonization.

The first activity in the Franco-German project was the development of common safety objectives. The work was given to both TSOs, GRS in Germany and IPSN in France. It is crucial in the future possible projects to determine if such competencies exist in participating countries and how can they cooperate in developing common safety objectives and basic principles. Even if this step looks very much straightforward, it took 18 months of very intensive efforts in the Franco-German project to come up with the common objectives and principles and therefore the amount of effort to achieve this step should not be underestimated. In the case of Franco-German efforts, this step included preparation of a questionnaire, common technical meetings, review of drafts, preparation of common reports for submission to higher levels in both countries for their comments and/or approval, preparation of final documents. Similar process is envisaged to take place also in the any future cooperative efforts and the full study should demonstrate how it would be achieved.

In many ways this step can be much easier achieved today as compared to 20 years ago due to developments of international standards which have happened in the meantime. Today we have the EU's amended Nuclear Safety Directive 2014/87/Euratom of 8. July 2014 which establishes a Community framework for the safety of nuclear installations and can in conjunction with other international standards be used as basic for the development of common high level safety objectives and safety principles in any future cooperative projects.

The same is true for the IAEA Safety Standards, especially its Safety Fundamentals and Safety Requirements, some of which have been revised to reflect the lessons learned for the Fukushima Daiichi accident. The most notable are changes to the Safety Requirements SSR 2/1 Rev 1 on NPP Design which bring the most recent developments in the international requirements for the design of NPPs.

WENRA has also revised their standards on Reference Levels after the Fukushima accident. Their document on the design of new NPPs can also serve as an excellent basis for the harmonization of requirements in the new collaborative project.

The same can be said also for the new Revision E of the European Utility Requirements. Being the utility requirements they are of special importance for new designs as they represent the utilities' views on the design of future NPPs. They are however available only to the utilities that participate in the development of those requirements.

All the above international standards would need to be thoroughly investigated and compared in order to come up with the best harmonized set of common requirements.

Similar harmonization process of different national licensing requirements is already under way in several participating countries under the auspices of the OECD/NEA in the framework of the so-called MDEP (Multinational Design Evaluation Programme) initiative which has the objective to make licensing among participating countries as compatible as possible.

A careful comparison of national regulation of the countries involved in the joint project with the above described fleet of international standards will have to be performed in order to extract any possible discrepancies. If the project was to aim at certifying the design that could be used in various countries (as was the case in the Franco-German effort), design compliance with the international standards should be the focus of this step.

Step 3: Determination of professional profile of available experts

In the Franco-German effort, two countries have had the same level of expertise available to them. The Full study would have to determine what are the minimum requirements on the available expertise in all participating countries for collaboration to be successful. In the Franco-German project it was very beneficial that most experts involved have known each other from some previous efforts (in their case it was mostly related to the work performed on the safety enhancement of WWER reactors in the decade before).

As indicated already in Chapter 4 on Feasibility study, before the work on the development of common safety standards and requirements begins, it should be investigated if the following issues related to professional competencies can be established:

- Define precisely the area and objectives of common work
- Define duties and responsibilities of all stakeholders and experts involved
- Clarify the existing structure of all organizations involved (TSO, advisory, regulatory) in all countries involved
- Define the level of professional capabilities and experience for the experts involved
- Evaluate possible impact of different culture/mentality/language in the interaction of the experts
- Establish clear rules and procedures for interaction, communication and documentation

Within this step it would be necessary to establish which capabilities exist in each country being involved in the joint project. In the case of France and Germany, they both had parallel structures at all levels. Both countries had well developed TSOs, advisory committees and regulatory bodies. If such parallel structure does not exist, it should be investigated how to overcome such situation. Today an organization that unites major European TSOs exists in the form of ETSON and its role, impact and possible utilization of services would have to be assessed.

In some instances, industry and regulators do not share their views on some common safety approaches. The Full study would have to investigate how to overcome such situation if it would to exist. In the Franco-German project for example, industry had difficulties to embrace in the common safety approach the inclusion of the crash of a military airplane or a steam explosion but due to strong commitment of all actors involved they were able to resolve this issue.

The Full Study should also investigate the added value of inclusion of experienced researchers with a detailed knowledge of physical phenomena, core behaviour, thermal-hydraulics, etc. or with the experimental background. However it should be pointed out in the Full Study that a proper balance between the enthusiastic researches on one hand and the experienced safety case reviewers utilizing their critical experience on the other has to be achieved.

In the Franco-German cooperation the responsible persons of the regulatory authorities, the chairmen’s of the advisory bodies as well as the project and technical group managers of the participating TSO’s have had a superior expertise and specifically the chairmen’s of the advisory bodies and TSO’s were respected authorities in the area of nuclear safety. In the circumstances of the Franco-German cooperation there was no need to develop a special formal procedure how to solve conflicts of interests or balancing divergent views. The practice to prepare written statements to issues of divergent views (including background and technical arguments) and starting further discussion in a higher level group worked quite well. This however does not necessarily have to be the case and the Full Study should explore the need for written formalization of the above activities.

Specific challenges may arise in the development of common positions (more in the codes & standard areas) if issues of different schools of thinking arise (value of pressure tests for pressurized components, containment structures, etc.) or national interests regarding the role of national industries which may be involved (e. g. limited capability of manufacturers regarding preferred technical solutions). Describing requirements more generally and developing alternative options to comply solve the issue in most cases but it would nevertheless be worthwhile to explore also this aspect in the Full Study.

Step 4: Overall schedule of the cooperative project

Depending on the reactor type, the work scope (level and volume of harmonization) a cooperative work may extend over many years up to a decade as in the case of the Franco-German example. Election periods in democratic countries are normally every 4 to 5 years. Nuclear safety and future developments are an area with a potential of controversial political debates, sometimes disruptively influencing the long term projects. In order not to lose the achieved results of cooperative work it is advisable to schedule the work in confined time pieces to achieve and publish results within shorter periods of time. Such an approach does not prevent possible disruptions (as experienced in the Franco-German cooperation) but makes any continuation easier if conditions change in the future.

The rules how the general public can participate in the final process of approval of safety requirements and codes & standards differ from country to country and will have to be investigated in the Full Study. In Germany “Guidelines” can be published as an action of the federal government. KTA safety standards are published as a draft with a 3 month period for public comments. The final results of the Franco - German cooperation have not proceeded to this step. Therefore the results do reflect the common position of the most relevant actors in the process with respect to the technical expertise; but the formal confirmations according to the national procedures were outstanding.

Today, the requirements on transparency and on information and participation of the public have substantially evolved as compared to the situation in the 1990s. Public participation is considered an essential element of licensing of nuclear installations and of preliminary steps such as pre-licensing or, even before that, the formulation (such as "White Papers") of national energy programmes. A cooperation of regulators modelled on the Franco-German approach would have to take this into account and would have to include, as an essential part, the publication of results (respecting commercially or security-related sensitive information) and giving the public the opportunity to comment and to have its say.

The Franco-German project had basically involved two distinctive steps. The first one was the development of Technical Guidelines for the design and Construction of the Next Generation of NPPs with Pressurised Water Reactors and the second one was to develop a detailed EPR Technical Codes. As mentioned above, circumstances have greatly changed in the last decades; today, cooperation of regulators would not be triggered by joint development of a reactor design by two or more EU Member States, but instead by joint evaluation of a finished design offered by a reactor vendor from a foreign country; this joint evaluation will be analysed in detail in Task 3 of this Report. Nevertheless, alignment both of regulatory requirements and of codes and standards would be an important element to facilitate such joint evaluation. Therefore, an approach similar to the Franco-German collaboration in this respect would be recommended also for future collaborative projects and the Full Study should determine the steps to be undertaken to achieve such project results.

6 CONCLUSIONS

The Franco-German EPR review in the 1990s was triggered by joint development of a reactor design by the nuclear industry of both countries, and it comprised the development of common regulatory requirements and a joint set of codes. Both work-streams followed an iterative process where requirements on the one hand and the reactor design on the other hand were developed in parallel.

A commendable element of the Franco-German cooperation was the strong and well-organised implementation, based on a governmental declaration and on agreements between all parties.

The successful creation of joint requirements is a good template for reaching an agreement on the requirements and standards to be used among the participating regulators. It resulted in a common review of the EPR design.

The Franco-German process was initiated by industry but in parallel supported by politics at the highest level. For any replication, such strong political support seems essential. For a replication the following boundary conditions must be considered:

- Strong support by governments
- Strong role of industry
- Experienced regulators and TSOs
- Technical approach
    - Develop common Safety Objectives
    - Derive the common views on essential issues
    - Develop and review the “Basic Design concept”
    - Develop and review the “Technical Guidelines for future PWRs”
    - Develop and review the ETCs
- Work conditions and organisation of the cooperation

Before the work on the development of common safety standards and requirements begins, the following issues would need to be addressed as a minimum:

- Define precisely the area and objectives of common work
- Define duties and responsibilities of all stakeholders involved
- Clarify compatibilities and non-compatibilities of existing regulatory frameworks
- Clarify the existing structure of all organizations involved (TSO, advisory, regulatory) in all countries involved
- Define the level of professional capabilities and experience for the experts involved
- Evaluate possible impact of different culture/mentality/language in the interaction of the experts
- Establish clear rules and procedures for interaction, communication and documentation

If a full study were undertaken, several steps would need to be investigated based on the experience of the Franco-German project:

- Step 1: Investigation of the status of the nuclear programs and regulatory frameworks in participating countries

- Step 2: Compatibility check of basic principles and safety objectives set in relevant national legislations
- Step 3: Determination of professional profile of available experts
    - Define precisely the area and objectives of common work
    - Define duties and responsibilities of all stakeholders and experts involved
    - Clarify the existing structure of all organizations involved (TSO, advisory, regulatory) in all countries involved
    - Define the level of professional capabilities and experience for the experts involved
    - Evaluate possible impact of different culture/mentality/language in the interaction of the experts
    - Establish clear rules and procedures for interaction, communication and documentation
- Step 4: Overall schedule of the cooperative project

Given all these factors, it would be worthwhile to undertake a full study with the aim of setting out the conditions for possible future cooperation between two or among several countries to establish a common ground for: designing, licensing or constructing selected nuclear power plant technologies.

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TASK 3 - A common European pre-licensing process: scope, content, implementation

TENDER SPECIFICATIONS OF TASK 3

EU Tender ENER/D2/2016-677 specified the requirements for Task 3 as Follows:

> "The concept of a European Reactor Design Acceptance (ERDA) was developed by ENISS within a dedicated working group of the European Nuclear Energy Forum (ENEF). A final report on the topic "WG RISKS / WG OPPORTUNITIES "EUROPEAN REACTOR DESIGN ACCEPTANCE (ERDA)" CORE GROUP – Roadmap Towards European Reactor Design Acceptance – 31 July 2012" was issued and presented to ENSREG Members during the 24th ENSREG meeting in May 2013. The ERDA proposal aims at a common design review and acceptance, the results of which would be shared among several EU Member States allowing the reactor design to be acceptable to all participating countries, except for necessary adaptation to specific local conditions, leading to effective European standardisation of reactor designs. The suggested approach is essentially based on voluntary actions by regulators and TSOs with support of the Commission.
>
> Deliverable: a detailed description of the technical content that a EU common pre-licensing process should include, taking into account the different types of reactors, the applicable safety standards and (as far as possible) the diversity of Member States national framework."

Under this task the existing approaches, both on national and on international level, towards a pre-licensing reactor design evaluation and "acceptance" were analysed, with a specific focus on the scope and content both of the review and the resulting statement by the participating authority/authorities. This is a desktop study, based on publicly available information, especially on information published by the relevant regulators. Also, the professional experience of the authors in regulatory matters on an international scale provides an essential contribution to the study.

Based on the analyses of the various approaches and a discussion of their practical advantages and drawbacks, a concept is developed for a Joint Overall Design Assessment, to be performed by the regulators of several Member States, leading to a Common Opinion on Design Acceptability.

The work undertaken under this Task 3 was performed in several steps:

1. **Introduction/ Background/ scope of work**

   This chapter sets out the purpose and scope of the study.

2. **National pre-licensing design reviews and their content**

   In this chapter, four national pre-licensing design reviews are analysed, with particular emphasis on scope and content of the review and on "best practice" to be emulated. Following this investigation, a comparison between the national examples is performed, using an overview table. There are many differences which are discussed and which provide valuable elements for the recommendations given for a European design assessment process.

3. **"Multinational" approaches in design review and their content**

   In this chapter, five multinational approaches are discussed, again with a special focus on scope and content of the review and on "best practice". The multinational approaches are compared and analysed. The experience from these different processes is evaluated and brought to fruition in the recommendations given for a European design assessment process.

4. **Proposed approach to the scope and technical content for a European design review**

For the scope/broadness of the design review, a generic list of topics is developed from the national and international examples analysed in the two preceding chapters. As concerns the depth/level of detail, the examples analysed in the report display a great variety from a very-high level approach to a very detailed assessment carrying the assessment of the relevant design-related topics as far as it would be done for a construction licence. For the purpose of this analysis, evaluation was classified into four grades, with grade 1 being rather generic and grade 4 being a full-scale assessment such as would be sufficient for a construction licence. On this basis, the Report recommends a specified depth of detail in the assessment, namely a grade 2 assessment for all aspects as a basis, plus a grade 3 assessment for selected topics. The depth of assessment is correlated to the "pyramid" of safety requirements going from laws and decrees down to project documents such as specifications. The concept leaves some leeway within a defined range of assessment, with the level of detail defined by the parties in the contractual arrangement underlying the Joint Overall Design Assessment. This proposed level of detail ensures meaningfulness of the assessment and enables, as far as possible, a smooth interface with the subsequent national licensing processes.

5. Implementation of a European design review

The Report discusses the fact that in Europe, a high level of harmonisation has been reached through the Safety Objective contained in the revised Safety Directive as well as through the Reference Levels of WENRA and, from industry side, through EUR (see Task 1). This should be a good starting point for a joint evaluation. The Report also discusses further elements of practical implementation of a design review. A contractual approach is commended, based on an agreement between participating regulators, their TSOs (if applicable) and the relevant vendor. In order to reach an agreement of participating regulators on the set of requirements, many elements of the Franco-German cooperation in the 1990s (see Task 2) could be emulated. The Report proposes a role for the European Commission, or a EU expert body designated by it, as a Clearinghouse or Secretariat for the process. Finally, the Report discusses the national implementation of the Common Opinion on Design Acceptability which is the outcome of the Joint Overall Design Assessment.

6. Short exemplary scenarios

The Report features short exemplary scenarios involving several Member States with nuclear new build projects. The joint assessment can be applied both to large reactors and to SMRs; the latter could become a relevant and effective template for implementation of such a process. Also, implementation of Gen IV designs is considered.

7. Conclusions

In the final chapter, a summary is given of the proposed concept of a European Joint Overall Design Assessment.

1 INTRODUCTION / BACKGROUND / SCOPE OF WORK

1.1 THE ERDA CONCEPT AND JOINT DESIGN ASSESSMENT

The ERDA Concept (European Reactor Design Acceptance), as laid down in the paper "Roadmap Towards a European Reactor Design Acceptance" of 31 July 2012 by the ERDA Subgroup of the European Nuclear Energy Forum (ENEF), suggests a common design review and acceptance, the results of which would be shared among several EU MS allowing the reactor design to be acceptable to all participating countries (except for site-specific issues). This could lead to an effective European standardisation of reactor designs.

The ERDA concept is based on the idea that a nuclear reactor design should be reviewed and approved in a more harmonized, efficient and consistent way rather than being separately reviewed by each national regulator in each EU Member State where a nuclear power plant of that design is to be built, as is the case now. Instead of “re-inventing the wheel” every time, ERDA looks at ways to achieve a common design review and acceptance, the results of which are shared among several EU Member States. Such a reactor design acceptance would be issued or mutually shared by a voluntary group of national regulators. As a result, a given reactor design can be built in the same way in all participating countries, except for necessary adaptation to specific local conditions.

The main task of this Task 3 report is to investigate in more detail than in the original ERDA concept the scope of the design assessment, and to give recommendations on the preferable depth of the assessment. In order to do this in a meaningful manner, it is nevertheless necessary to take a deeper look, and perhaps partly to re-think, the aims, objectives and tools of the ERDA concept because they will contribute to define the relevant scope and depth of the assessment.

1.2 JODA AND CODA

In this Task 3 report, the process will be called Joint Overall Design Assessment or JODA. The ERDA concept was the point of departure for the tender specification and the creation of the present report, but ERDA is a specific concept suggested by an ENEF subgroup in 2012. The deliberations on scope and implementation of a common design assessment by several regulators in this report go beyond the ERDA concept.

JODA is the process; the written outcome of this process will be called, in this report, Common Opinion on Design Acceptability or CODA. Besides JODA and CODA, in the text more generic words such as "design pre-licensing", "joint evaluation" or "acceptability statement" may be used as well.

1.3 PURPOSE OF A JOINT EVALUATION AND AGENDA OF RELEVANT STAKEHOLDERS

Starting point is the general ERDA idea of having several nuclear regulators of EU Member States perform a joint assessment of a given reactor design, resulting in a statement that the design is basically acceptable in each of those Member States. In any subsequent licensing process in one of those Member States, this statement would be referenced and would be relevant in a sense that the assessment, as embodied in the statement, will not be done again by the national regulator unless there is a cause.

This is a multi-faceted concept. It is important to note that the idea of a design acceptance statement does not necessarily involve multinational cooperation; in fact, it has been discussed and implemented on the national level in various countries in the last decades. If this idea is transferred to an international dimension, the notion of international harmonization, of not "re-inventing the wheel" in every country, would come on top of this, giving it an additional level of complexity. Therefore, it makes sense to analyse

- both existing national pre-licensing design reviews and
- international efforts and activities for a joint design assessment.

On a national level, there are several incentives for installing a pre-licensing design assessment:

- If several plants of the same design are planned for different sites, it is reasonable to perform the design review as a first step in order to evaluate aspects which are the same for all subsequent licensing processes. For licensing, the focus is then on site- and operator-specific topics. This is the model underlying, for example, the UK GDA process.
- Licensing processes are long, costly and fraught with uncertainty. On the industry/investor side, there is a general drive towards achieving more certainty and predictability by installing several phases of assessment and licensing and therefore to stepwise reduce the licensing risk. If the pre-licensing phase results in a positive statement by the regulator, then there is more confidence that the actual licensing process will be successfully completed as well.
- In some cases, vendors are interested in a separate design review for marketing purposes, especially if the design is intended for export and will not be used in the vendor's country. A good example for this is the French Review of Safety Options done for the ATMEA design (see 2.3 below).

On an international level, the incentives for a joint pre-licensing by the regulators of several countries are similar but not entirely the same. The multinational dimension of the process has some aspects of its own:

- The main reason for an initiative such as MDEP is the regulators' desire to employ their (limited) resources in a more efficient manner by coordinating their design review activities, instead of "re-inventing the wheel" in every country. The benefits offered by such shared assessment can be fully reaped if the safety requirements are more or less harmonized in the participating countries; to the extent this is not the case and diverging requirements remain, a solution may be to accept a design on different grounds but with the same result.
- For vendors, the main reason is the desire to achieve standardisation of reactor designs, meaning that vendors can build their design in several countries without having to adapt or even re-design it to the regulations of each country. This results in an increased efficiency in the allocation of resources and also in increased certainty and investor confidence. For Small Modular Reactors (SMRs), which mostly follow the concept of factory-built modules shipped to the site and assembled there, standardisation is absolutely essential. Their deployment will practically be impossible if a licence according to different standards is necessary in every country, thus deleting the very concept of modularisation.

1.4 PRELIMINARY REFLECTIONS ON THE CONTENT OF THE JOINT PRE-LICENSING

The content of a Joint Overall Design Assessment (JODA) has two aspects which can be considered stepwise, one after the other:

1. The scope in the sense of broadness of the assessment, meaning the range of topics to be covered; and
2. The depth of the assessment, meaning the question to which extent of detail the assessment is taken for each of the topics within the scope.

For the scope/broadness of a design pre-licensing assessment, there is an inherent limit: it must by its very nature be restricted to those aspects which are independent of a specific project; in other words, those aspects which all projects have in common. This is mainly the reactor design but excludes site-specific and operator-specific considerations.

Within this inherent limit, the scope could in theory be further restricted voluntarily, with an assessment focusing, for example, on single features of a plant.

As to the depth or level of detail of the assessment, in theory everything is possible from a very high-level and generic appraisal to a very detailed approach carrying the assessment of the relevant design-related topics as far as it would be done for a construction licence. Already in a preliminary reflection, it is apparent that there is a certain antagonism and trade-off: on the one hand, a high level of detail, comparable to that of actual licensing processes, resolves design assessment issues to a high degree and achieves the certainty which is one of the main objectives of introducing reactor design pre-licensing. On the other hand, such an in-depth assessment makes the process long and potentially cumbersome; it supposes the existence of a finished detailed design; and it raises issues of subsequent implementation of the results into the licensing processes and of dealing with changes in the detailed design which necessarily occur during construction and which would lead to deviations from the Common Opinion on Design Acceptability (CODA). These issues are even more apparent in an international cooperation of regulators where they could appear as show-stoppers.

The analysis of national and international design review processes will show how this has been addressed in practice.

2 NATIONAL PRE-LICENSING DESIGN REVIEWS AND THEIR CONTENT

2.1 UK: GDA

2.1.1 GENERAL DESCRIPTION

The Generic Design Assessment (GDA) was developed by the relevant UK regulators (today: the Office of Nuclear Regulation, ONR, and the Environment Agency, EA) in the wake of the government's 2006 Energy Review. In the Review, the UK Government had concluded that nuclear new build was necessary. As the UK Government expected that plants of the same design might be built on several sites, the regulatory process for new nuclear power stations was split into two phases: GDA (Phase 1) and Site License (Phase 2). This allows the separation of design issues which are independent from a given project and which are assessed in the GDA from specific site related issues which are dealt with in the Nuclear Site Licence process. The GDA is not mandatory; however, practice has shown that due to its advantages for reactor vendors for new reactors in UK, it regularly precedes the mandatory site license process.

The GDA process is undertaken by the main nuclear regulators, the Office for Nuclear Regulation (ONR) and the Environment Agency. The ONR focuses on nuclear safety and security whereas the Environment Agency deals with radiation issues (discharges etc.). For the purpose of this study we focus on the GDA assessment concerning nuclear safety by the ONR.

Requests for a GDA usually originate from a reactor designer/vendor. In contrast to comparable assessments in other countries, the UK GDA is mostly initiated by a vendor/operator partnership (Requesting Party). This is recommended by the regulators, as they put a special focus on the capabilities of the applicant/future licensee (see below).

The outcome of the GDA, if successful, is a Design Acceptance Confirmation (DAC) issued by the ONR, plus a Statement of Design Acceptability (SoDA) issued by the EA. If the generic design is found to be generally acceptable but some issues remain open (so-called GDA Issues), the Requesting Party will be required to provide credible resolution plans. If the regulators are satisfied with the resolution plans, they issue an interim Design Assessment Confirmation (iDAC) and an interim Statement of Design Acceptability (iSoDA).

The ONR has divided the GDA into four steps. This was a pragmatic decision, given that the assessment work to be undertaken was anticipated to take typically around four years and too many steps would increase overhead and process burden which could potentially lengthen the overall process.

The DAC is issued against a defined “reference design configuration” and a defined set of submission documentation. It remains valid for the specified design for 10 years from the date of issue (subject to no significant new information arising that questions the previous assessment). When a subsequent site-specific application is based on a design that has undergone GDA, the regulators will take full account of the work that they have already carried out.

It deserves notion that the split of the overall licensing process in two phases (GDA and site licensing) and the GDA as such is not contained in any legislation (Nuclear Installations Act); it relies purely on guidance issued by the regulators. Only the site licence process is defined by legislation.

So far, three designs have received a DAC/SoDA: EDF's and Areva's EPR in 2012, Westinghouse's AP1000 in March 2017 (an iDAC had already been issued in 2011) and Hitachi-GE's UK Advanced Boiling Water

Reactor (UK-ABWR) in December 2017. In October 2016, the GDA process was started for General Nuclear System Ltd's UK HPR1000.

2.1.2 SCOPE AND CONTENT OF THE REVIEW

The four steps of the GDA are based on the so-called “claims-arguments-evidence (CAE)” approach which is commonly used in the UK in structuring safety cases.

This GDA approach involves the following steps:

1. Preparation of the design, safety and security submissions
2. Fundamental design, safety and security claims overview
3. Overall design, safety and security arguments review
4. Detailed design, safety and security evidence assessment.

With this approach the assessment is getting increasingly detailed at each step and it allows ONR to identify key design issues and possible "show-stoppers" early in the process. The basic document throughout the GDA is the Pre-Construction Safety Report (PCSR) starting with a draft in the early steps. The safety objective is to demonstrate that construction, manufacture and installation activities will result in a plant of appropriate quality and that the constructed plant will be capable of being operated within safe limits and will be able to mitigate design basis accidents.

With respect to scope and technical content step 4 is most important. At the start of step 4 the Requesting Party (RP) has to provide detailed information supporting the safety case including research material that provides evidence to support the PCSR.

Basically, the scope of the GDA is defined by the information the Requesting Party has submitted. Therefore, the Requesting Party has control over the range of coverage of the GDA. This applies both to the scope/broadness of the assessment and to the depth/level of detail.

Concerning the scope/broadness, there is however, the requirement that the PCSR is "comprehensive", allows assessment in all relevant technical topic areas and also includes generic site criteria. With a view both to the scope/amplitude and to the level of detail, ONR requests that the provided information needs to be sufficient to allow ONR to undertake a "meaningful assessment" of the generic safety for the design. According to ONR guidance,

> *"A meaningful GDA will be one where ONR has:*
>
> - *received sufficient information on the generic design in the safety and security submissions to allow assessment in all relevant technical topic areas; and*
> - *completed a sufficiently thorough and detailed assessment of that information.*
>
> *In the above:*
>
> - *‘sufficient information’ will have been received if ONR judges that it has been provided with submissions that cover the full scope and depth necessary for ONR to carry out its technical assessments;*
> - *'thorough and detailed assessment' means that ONR has looked in detail at the submissions and judged them against the SAPs [Safety Assessment Principles], including the need to demonstrate that risks are reduced, or are capable of being reduced, ALARP. The assessment relates only to the information provided on the generic design and does not mean that ONR has received and assessed all the information necessary to permit construction and operation of a plant, based on that design."*

The relevant information is captured in the following documents which will be referenced in the DAC:

- Generic PCSR, incl. generic site envelope
- GDA Design Reference; and
- Supporting references as identified in a Master Document Submission List.

Whereas the information provided by the RP must cover the total safety case, the ONR concedes that depth and scope of ONR's assessment is unlikely to be the same across all technical areas. In ONR's view, engineering details of the design will not be available at the GDA stage, as it is normal to finalise some of these during the site-specific procurement and construction programme. Sampling assessments are sufficient to allow ONR to make a balanced judgement on the overall acceptability of the generic safety case.

A UK-specific aspect concerning the scope is that the pre-licensing design assessment does not fully exclude matters relating to the organisational structure and capabilities of the potential licensee. According to the ONR, even in the GDA stage there must be arrangements for supporting future licensees to put in place a Design Authority. Furthermore, the safety and security case produced in GDA must be developed with a potential licensee's legal duties in mind.

To illustrate the technical content and the details, some examples which are requested by ONR are given here:

- *Fault analysis comprising DBA, PSA, and Severe Accident analysis*
- *Quality management arrangements for the design and safety case production*
- *List of Initiating Event Groups in the generic PSA*
- *Commissioning test programme*
- *Thermal analysis to confirm the timescales for consequential loss of C&I and electrical equipment.*

Also, some examples from the UK EPR submission may be provided here:

- System Design Manual
- Preliminary Safety Analysis Report
- Calculation of fluid dynamic loads for the loop after LOCA at 100% Power Operation and Stretch-out
- RPV internals mechanical dimensioning (Lower internals, Core barrel upper part …).

The degree of details in these examples/ documents show that the GDA covers a substantial part of the information on the reactor design which can be given before procurement and construction and which are not related to a specific site. However, a site envelope must be defined, and site related safety aspects must be assessed for this site envelope.

To a certain degree it is up to the Requesting Party to decide what level of detail of information should be covered by the GDA. The scope is then defined in the Design Reference which lists all the documents describing the design of the reactor that the GDA submissions refer to. All documents including the supporting references are part of the Master Document Submission List.

2.1.3 RELEVANT REQUIREMENTS

As is well known, the UK regulatory system is not prescriptive but rather goal-oriented in nature. In line with this, there is no fixed set of detailed design requirements. Instead, the applicants establish safety cases which need to be formally agreed by the ONR. The assessment is being performed according to ONR guidance documents, the most important being the Safety Assessment Principles (SAPs). Under these Principles, the overall requirement is that it can be shown that risk is reduced as far as reasonably practicable (ALARP, as low as reasonably practicable).

Within this system, it is possible for applicants to introduce non-UK codes and standards and, to some extent, also documents previously used in licensing processes in other countries; however, they have to be "translated" to the UK philosophy and the regulator has to be convinced that they are a sufficient basis for a safety case. The ONR has set forth its approach in dealing, in a constructive manner, with international good practice, international standards and the assessments previously undertaken by overseas nuclear regulators in a dedicated guidance document (Health and Safety Executive, New nuclear power stations Generic Design Assessment – Safety assessment in an international context, revision 3, 2009).

2.1.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

The DAC is not legally binding – in fact, as explained above, it is nowhere mentioned in legislation. However, it has a far-reaching practical status. In its guidance and practice, the ONR has set forth that if an application for a nuclear site licence is based on a design that has undergone GDA, the regulators will take full account of the DAC. Consistent with this strong status of the DAC, ONR has established a period of validity of a DAC. This period is ten years from the date of issue (Office for Nuclear Regulation, New nuclear reactors: Generic Design Assessment, Guidance to Requesting Parties, rev. 3, September 2016, no. 128). The ten years were determined as corresponding to the ten-year interval for performing a Periodic Safety Assessment. If during the ten-year period any new information emerges which calls into question the basis of ONR's original assessment of the design, then ONR would need to consider whether the DAC (or iDAC) remains valid.

Following a GDA, ONR’s subsequent licensing assessment would centre primarily on site-specific issues that have an impact on the safety of the station and matters relating to the organisational structure and capabilities of the potential licensee.

There is another link from GDA to site-specific assessment. The DAC identifies GDA Assessment Findings. The findings are primarily concerned with the provision of additional evidence, after GDA, to confirm certain safety, security or environmental aspects as the project progresses through the detailed design, construction and commissioning stages. The site operator will need to address all the Assessment Findings from the GDA process under the nuclear site licence. In this way, the DAC provides further "pre-determination" of the contents of a licensing process.

2.1.5 BEST PRACTICE

The UK approach can be described as "pragmatic". This starts with the fact that it has been "invented" and implemented by the competent regulators without any legislation. Obviously, this is also due to the general UK legal system. But also in implementation, the regulators have chosen a pragmatic, workable approach which seems to function in practice. Successful features seem to be the division into four steps and the way to deal with outstanding issues via resolution plans and "interim" DACs.

The GDA is basically a full-scope assessment of reactor design safety (in terms of the range of topics to be covered); there is even some extension of the scope towards the future licensee who is also involved in the process. Concerning the level of detail, the regulators strike a sensible balance: on the one hand, they do not expect the Requesting Party to file information to a level of detail which would be sufficient for a nuclear site licence review; on the other hand, they require a level of submission which enables them to perform a "meaningful" assessment. This gives some flexibility within reasonable limits and seems a commendable approach.

Another remarkable aspect is the flexibility in terms of safety requirements against which the assessment is performed, based on the use of Safety Assessment Principles and with a certain leeway to introduce foreign documents as well as codes and standards. This may be an interesting feature for an multinational design review where a fixed set of detailed requirements will likely not be available and where the participating regulators will to some extent have to reconcile various regulatory approaches and philosophies.

Finally, there seems to be workable interface between the DAC, resulting from the GDA, and the subsequent assessment for issuing the nuclear site licence and subsequent assessments at milestones under that licence. GDA Assessment Findings are fixed in the GDA and require subsequent addressing and resolution. Thus, there is neither a gap nor an overlap.

2.2 US: DESIGN CERTIFICATION

2.2.1 GENERAL DESCRIPTION

In 1989, the U.S. Nuclear Regulatory Commission (NRC) established a new alternative process for nuclear plant licensing (Rule 10 CFR Part 52). This licensing regime comprises a licensing process resulting in a single combined construction and operation licence (COL) as well as two elements of pre-licensing, namely an Early Site Permit (ESP) process, and a standard plant Design Certification (DC) process. By issuing a Design Certification, the NRC approves a nuclear power plant design, independent of an application to construct or operate a plant. This approach allows early resolution of safety and environmental issues. ESPs address site safety issues, environmental protection issues, and plans for coping with emergencies, and are independent of a review of a specific nuclear plant design.

Through the Design Certification the applicant obtains pre-approval of an essentially complete plant design. This reduces licensing uncertainties by resolving all design issues and plant standardization is facilitated.

A Design Certification is valid for 15 years from the date of issuance but can be renewed for an additional 10 to 15 years.

2.2.2 SCOPE AND CONTENT OF THE REVIEW

The scope of assessment for the Design Certification is driven by the principal objective to achieve resolve all design issues ahead of COL licensing. Therefore, an application for a standard Design Certification must provide detailed information about the structures, systems, components, and design features. This requires final design information. All the relevant information is contained in the Design Control Document (DCD).

The DCD is divided into two parts (called "Tiers"). The most important high-level information in the DCD is classified as Tier 1 material. Tier 1 material is to be certified. The other part – the more detailed documents – form the Tier 2 (and Tier 2*, see below) material. This information is to be approved, but not certified, by the NRC.

The design description in Tier 1 includes the principal performance characteristics and safety functions of the plant's structures, systems, and components (SSCs). In addition, the Tier 1 document identifies significant site parameters and requirements for significant interfaces between the standard design and those aspects of the plant design that are site-specific. With respect to the mitigation of severe accidents design alternatives may be offered. Beside the description of the design the most important part of the Tier 1 material are the proposed inspections, tests, analyses, and acceptance criteria (ITAAC).

ITAAC are a very important part of the USA licensing regime and establish a substantial link between pre-licensing (design certification) and licensing. During all relevant licensing steps, the licensee must specify all ITAAC that are necessary and sufficient to provide reasonable assurance that, if the ITAAC are performed and the acceptance criteria met, the plant is built and will operate as claimed in the licensing documents.

In this context, the scope of assessment for Design Certification must cover the total plant except siting issues. As there is no further technical review of the structures, systems and components, of the design features and of the ITAAC during the following licensing process, the extent of the review during the Design Certification must be all-embracing like for a construction licence. To illustrate the technical content of the Design Certification process an example for the table of content of a Design Control Document (mainly Tier 2) is given in **Appendix 1**.

2.2.3 RELEVANT REQUIREMENTS

The requirements for design assessment are those contained in the relevant US NRC regulations. As the US approach is very prescriptive (as opposed to goal-oriented approaches such as that followed in the UK), the objective of the assessment of a design is to verify compliance with the extensive and detailed regulations.

2.2.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

Legally, the Design Certification is adopted by the NRC through rulemaking. The certification may be issued for a term of 15 years, which can be extended. No change can be made to a certification during its term unless the change is necessary for compliance with NRC requirements in effect at the time the certification was issued or to meet the statutory standard of "adequate protection". Construction and operation of a particular nuclear power plant is authorised by way of the combined construction and operating licence (COL). If there is a Design Certification for the chosen reactor design, the applicant for a COL will reference it. The issues resolved by the Design Certification and by the ESP are not reconsidered during the COL review except under narrow, clearly defined circumstances. All ITAAC defined in the design certification will be effective without further review. Only detailed siting information will be reviewed additionally (unless an ESP has been issued).

2.2.5 BEST PRACTICE

The US Design Certification offers the model of a very stringent and legally well-defined pre-licensing process. The scope covers all aspects related to the safety of a reactor design; the level of detail is the same as for a construction licence. Once the Design Certification has been obtained, a major step in licensing has been taken and design safety issues are definitely resolved.

However, the fact that decisions on design acceptance are finalized and binding with respect to the subsequent licensing (COL) and construction of a nuclear power plant has also its drawbacks. The Design Control Document (DCD) contains detailed technical information and is very rigid with respect to changes. It is difficult to implement changes in the DCD as they are subject to a very formal process. Changes to Tier 1 information, considered more safety-relevant, lead to regulatory review, public comment, intervention and are certified as license amendment. Changes to Tier 2 material are governed by a less stringent process, with the exception, however, of changes to Tier 2* material which are being treated as request for a license amendment. The Tier 2* designation was created in the early 1990s to minimize the scope of Tier 1 information and, in theory, to provide greater flexibility in making changes to that information. In practice, however, the level of effort is nearly identical for Tier 1 and Tier 2* changes because both require prior NRC approval and license amendments.

This has led to immense problems in current projects as during the COL process and during construction many changes to the Design Control Document have been requested. Therefore U.S. industry urges NRC to reform the Design Certification process namely eliminating the Tier 2* designation. This puts a spotlight on the inherent difficulties of full-depth design certification.

With a view to the objectives of this report, it is apparent that the very strict US approach, which has inherent disadvantages (beside obvious advantages) already on the national level, would be very difficult to implement on an international scale. The level of detail of the assessment, the very prescriptive and detailed set of requirements and the legally binding nature of the final statement would require, for translation on the international scale, a very intensive international cooperation and in fact the establishment of a clearly defined system by intergovernmental agreements and changes in national legislation.

2.3 FRANCE: REVIEW OF SAFETY OPTIONS

2.3.1 GENERAL DESCRIPTION

The French nuclear legislative and regulatory system contains the option of an independent, generic assessment of a design, the so-called "review on safety options" (RSO; in French: *Examen des options de sûreté*).

Article 6 of Décret n° 2007-1557 of 2 November 2007 on basic nuclear installations (full title: *Décret n° 2007-1557 of 2 novembre 2007 relatif aux installations nucléaires de base et au contrôle, en matière de sûreté nucléaire, du transport de substances radioactives*) states the following:

> *“Any person intending to operate a basic nuclear installation may, prior to initiating the authorisation decree procedure related to the creation stage as specified in article 29 of the Act of 13 June 2006, ask the Nuclear Safety Authority for its opinion concerning all or some of the options it has chosen to ensure the safety of this installation.”* (Translation taken from English version of ASN report on ATMEA1)

As of today, the reference to the Act of 2006 is no longer valid; the licence for a nuclear installation (*décret d'autorisation de creation*) is now provided for in Art. L593-7 of the Environmental Code (*Code de l'Environnement*).

The review is in the competence of the nuclear regulatory body, the ASN (*Autorité de Sûreté Nucléaire*), which generally avails itself of the advice of the technical support organisation IRSN (*Institut de Radioprotection et de Sûreté Nucléaire*) and of a number of standing advisory committees, in particular the GPR (*Groupe permanent d' experts pour les réacteurs nucléares* [Advisory Committee for nuclear reactors]) and the GPESPN (*Groupe permanent d'experts pour les équipements sous pression nucléaires* [Advisory Committee for nuclear pressure equipment]). As the RSO for the ATMEA1 design has shown, the ASN relies to a very large extent on the statements of these two groups: according to the RSO report, ASN's conclusions " result essentially from the opinions issued by the GP ESPN and the GPR" (*Avis n° 2012-AV-0143 de l’Autorité de sûreté nucléaire du 31 janvier 2012 sur les options de sûreté du projet de réacteur ATMEA1*, p. 34).

By contrast, the person of the applicant for an RSO is not expressly defined. The wording "any person intending to operate a basic nuclear installation" suggests that only a future operator can apply for the review. However, this seems not adequate for the constellation that the reactor designer wishes a site-independent design assessment similar to the UK GDA or the US Design Certification. In the only instance the review of safety options was actually put into use for a reactor design, it was applied for by the reactor designer ATMEA (see the following text). Therefore, it seems that any person can apply which presents a tangible interest in the ASN review and makes it plausible to the ASN that the outcome of the review will be used.

So far, the process of an independent review of design options has been utilised only once for a reactor, namely for the ATMEA1 PWR reactor design (1100 MW_e) developed by ATMEA, a joint venture of Areva and Mitsubishi Heavy Industries (MHI). The process started in 2009 and ended in the issuing of ASN's statement in 2012. Very recently, the process has been used for the Cigéo, the planned central storage facility for nuclear waste in deep geological formations; here, ASN published its opinion on 11 January 2018. This shows the amplitude of the process, which can be utilised for any major nuclear facility (*installation nucléaire de base* as defined in French legislation). In this report, only the ATMEA review will be discussed.

Strictly speaking, the ATMEA review was not done under Art. 6 of the 2007 Decree since it was applied for by the reactor designer and since at that time there were no plans to actually build an ATMEA1 in France. It seems that the main rationale for ATMEA to apply for the design review was to use it to facilitate export of the technology to other countries.

For this reason, the review was based not directly on Art. 6 of the 2007 Decree, but on a contract entered into between ATMEA and a "consortium" formed between ASN and IRSN; this way, financing was secured – the public bodies could invoice their expenses and their work to the applicant without having to procure financing from their budget appropriation. As ASN was careful to point out, "This difference in legal aspects has, however, no impact on the technical analysis to be completed: it was carried out under the same conditions as for a basic nuclear installation intended for construction in France" (ASN report, p. 7).

2.3.2 SCOPE AND CONTENT OF THE REVIEW

There is no fixed scope for the RSO. This is evident in the wording of Art. 6 which says that the review is performed "concerning all or some of the options it [the applicant] has chosen to ensure the safety of this installation". In substance, this means the ASN reviews those elements of the design which have been submitted to it in the framework of the RSO by the applicant.

The scope of the ATMEA review was thus defined by the "safety options dossier" submitted by ATMEA. The scope was contained in an Appendix 1 to the agreement established between ATMEA and the consortium of ASN and IRSN as well as in an amendment relating to “aircraft crashes”. The elements of this dossier are listed in **Appendix 2**.

2.3.3 RELEVANT REQUIREMENTS

The review basis for the RSO comprises (see ASN report, p. 8):

- the French regulations in force on the date the agreement is signed;
- the Technical Guidelines for the design and construction of the next generation of pressurised water reactors (the document established in 2000 by French authorities together with German experts, as described in the Task 2 report, and published by ASN in 2004);
- the para-regulatory texts relating to the design of the reactors of this next generation.

With letter dated 22 September 2009 (Annex 4 to the ASN report – only available in the French version), ASN delineated the regulations applicable to the ATMEA1 design and excluded those whose scope was not affected or which were outdated. Particular emphasis was given to the "Technical Guidelines" which, according to ASN, supersede earlier regulations on PWR.

2.3.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

As Art. 6 of the Decree on INB states, ASN issues an "opinion". This wording shows that this is less than a binding design certification. Also, as we have seen, the scope of the assessment and therefore the content of the opinion is very flexible.

The main pronouncement of the ASN is contained in the following wording:

"ASN staff notes that the safety objectives adopted by the ATMEA project … are consistent with the Technical Guidelines. More generally, ASN staff considers that the above-mentioned Technical Guidelines are ***overall adequately taken into account*** *in the ATMEA1 reactor safety options. At this stage of the project (no detailed design, unknown site, consistency with the Technical Guidelines), the technical review has* ***not identified any significant incompatibility*** *with the statutory requirements and their associated regulatory guidance. (…) In so far as concerns the design options of the main nuclear pressure-retaining equipment (ESPN), ASN staff considers that the pre-design studies and the design options selected by ATMEA* ***do not show any elements liable to cast doubt, at this stage, on the use of such equipment in a nuclear reactor****. (…)"*

The opinion further raises some caveats and indicates at which points a refined assessment at a later date will be necessary, thus defining the interface with the actual licensing process:

- *"….in view, in particular, of the innovations included in the design envisaged for the ATMEA1 reactor, there must be close collaboration between the designer and the future operator to deal with these questions satisfactorily."*
- *"…some measures, in order to comply with French practices, may require changes to the design or construction in the event that the construction of such a reactor were to be envisaged in France"*
- Some questions *"require in fact, in whole or in part, choices by the reactor operator (licensee) or the equipment manufacturers. The answers to these questions thus require a more detailed definition of the design of the installation and its operating procedures.*"

The principal licence for a nuclear installation is the "*autorisation de création*" issued by ministerial decree. Article 8 of the 2007 Decree contains a list of documents to be filed in a licensing process. One of these documents is the Preliminary Safety Report (*rapport préliminaire de sûreté*). Article 10 sets forth:

"*If the installation corresponds to a model for which the safety options have already been covered in an opinion from the Nuclear Safety Authority in the conditions defined in article 6, the report identifies the questions already examined within this framework, the additional studies carried out and the additional justifications provided, in particular those requested by the Nuclear Safety Authority in its opinion. As applicable, it presents the modifications or additions made to the options which were the subject of the Authority's opinion.*"

According to Article 13 para. 1, if an RSO report has been done for the reactor design, it is also part of the set of documents underpinning the public enquiry.

There is no further provision in the 2007 Decree expressly saying that the RSO report is referenced in a mandatory way during the licensing process. However, the provisions mentioned above clearly show that the RSO report, which is made public, will be taken into account by the authority and that any deviation from the determinations presented in the RSO report would have to be justified. Another indication for this is the fact that according to Article 6, the ASN "may set a validity period for its opinion". "Validity" can only be attributed to a statement which has certain (if limited) binding effect; once the period of validity has expired, the binding effect is lifted.

In France, there was no reactor licensing process where the option of referencing an RSO was tested. An NPP using the ATMEA1 design was never actually applied for; and the EPR design employed at Flamanville, the only NPP new build in France in the last two decades, was not the subject of a previous RSO process (because the RSO did not yet exist at the time the basic design of the EPR was conceived). Therefore, there is no practical experience with the interface between an RSO and a subsequent licensing process. In future, the Cigéo repository (where the ASN opinion was issued in January 2018) may be a showcase for such an interface, but this will take some years. Even now, however, the analysis of the legal and regulatory documents defining the RSO and the use of the RSO report have shown that the RSO as such would be a meaningful reference for a licensing process and would be taken into account by the regulatory authority.

2.3.5 GOOD PRACTICE

The most interesting aspect of the French RSO is probably the flexibility of scope and technical content of the design review. As Art. 6 of the Decree on INB makes clear, it depends on the applicant whether he submits "all or some" of his safety option choices to the authority; this gives much leeway both in terms of scope/broadness and of technical content/detail of the review solicited by the applicant. In the same vein, the review can be done for safety options chosen for any significant nuclear facility, not only reactors.

In terms of process, the ATMEA review is particularly interesting as it was done not on the basis of the Decree as such, but based on an agreement between the applicant and a "consortium" of the regulator and its TSO, which both defined the scope and content of the review and settled its financing.

Finally, the RSO contains some helpful provisions on the relevance of the "opinion" on future licensing processes, even if this has not yet been tested in practice.

2.4 CANADA: VENDOR DESIGN REVIEW (VDR)

2.4.1 GENERAL DESCRIPTION

The pre-licensing Vendor Design Review (VDR) is provided as an optional service by the Canadian regulator Canadian Nuclear Safety Commission (CNSC) when requested by a vendor. Both parties enter into a Service Agreement for this purpose and agree on an overall vendor design review project plan.

The review consists of three phases:

- Phase 1 review – Compliance with regulatory requirements: CNSC staff assesses the information submitted in support of the vendor’s design and determine if, at a general level, the design intent complies with CNSC requirements.
- Phase 2 review – Pre-licensing assessment: This phase goes into further detail, with a focus on identifying potential fundamental barriers to the licensing of the vendor’s design for a nuclear power plant or small reactor in Canada.
- Phase 3 review – Pre-construction follow-up: In this phase, the vendor can choose to follow up on one or more focus areas covered in Phase 1 and 2 against CNSC requirements pertaining to a construction licence. For those areas, the vendor’s anticipated goal is to avoid a detailed revisit by CNSC during the review of the construction licence application.

An interesting aspect of the review is that while former reviews were solicited for large reactor designs such as the CANDU 6 or Westinghouse's AP1000, currently all of the ten ongoing VDRs are pertaining to SMRs.

The CNSC is to a certain extent open to taking into account regulatory feedback the vendor has obtained through an assessment/licensing process in another country. According to CNSC guidance, the CNSC will consider such material provided that the vendor submits this information and explains how that information demonstrates the design will meet Canadian requirements. The CNSC would conduct its own assessment in light of its regulatory framework but it would use the information submitted to the extent that the information is compatible with the CNSC review process.

2.4.2 SCOPE AND CONTENT OF THE REVIEW

The CNSC reviews, during Phase 1 and Phase 2, 19 focus areas. These focus areas are contained, with further explanations, in the Guidance Document GD-385, Pre-licensing Review of a Vendor's Reactor Design. The focus areas are listed in **Appendix 3** to this Report. The vendor may propose additional focus areas that are specific to the reactor design.

Therefore, concerning the scope/ broadness of the process, there is strong guidance by the regulator on a standard scope in Phases 1 and 2 but also some flexibility to add additional topics. Phase 3 is very flexible and the list of topics depends on the applicant's choice.

As to the depth of the assessment and the level of detail, this increases with the three steps introduced above. Generally, Phases 1 and 2 do not require much detail in the applicant's submissions. This is in line with the VDR's aim of achieving early identification and resolution of potential regulatory or technical issues in the design process, particularly those that could result in significant changes to the design or safety analysis. Phase 2 requires more detailed information along the 19 focus areas than Phase 1, but even Phase 2 only supposes the completion of a "Basic engineering program". According to CNSC guidance, Phase 3 should not be initiated by a vendor until the design's (non-site-specific) detailed engineering program is under way; even this does not necessitate a full detailed design.

2.4.3 RELEVANT REQUIREMENTS

Relevant for the VDR are the most recent CNSC design requirements for new nuclear power plants in Canada. These are mainly contained in two documents:

- REGDOC-2.5.2, Design Of Reactor Facilities: Nuclear Power Plants
- Design of Small Reactor Facilities (RD-367).

Besides these two basic documents, all other related CNSC regulatory documents and Canadian codes & standards are also taken into account.

2.4.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

At the end of each phase, a CNSC statement is issued with standardised wording for the main finding.

For Phase 1, it reads: "CNSC staff has determined that the design intent is compliant with the CNSC regulatory requirements and meets the expectations for new nuclear power plant [small reactor] designs in Canada".

For Phase 2, it reads: "This review provides a further level of assurance that [name of vendor] has taken into account regulatory requirements and expectations. Based on the Phase 2 review, CNSC staff concludes that there are no fundamental barriers to licensing the [name of design] design in Canada." If additional work is required to resolve issues remaining in any of the focus areas, there will be the following caveat: "This statement is subject to the successful completion of [name of vendor and reactor]'s planned activities, in particular those related to: [list of review focus areas]."

For Phase 3, due to its more flexible and customised approach, there is no standardised wording for the concluding statement.

None of the statements has a binding effect on subsequent licensing decisions. In fact, the relevant regulatory Guidance Document makes clear that "the conclusions of a design review do not bind or otherwise influence decisions made by the Commission Tribunal, with whom the authority resides to issue licences for nuclear power plants and small reactors". Nevertheless, it is difficult to see how the CNSC could adopt an entirely different view during the licensing process without new information justifying this. The CNSC makes the point that the process results in "increased regulatory efficiencies" and supposes a commitment to this end on both sides.

2.4.5 GOOD PRACTICE

The Canadian VDR seems to be put into effect smoothly and efficiently, given the large number of VDRs already performed and the ten VDRs currently ongoing. An interesting aspect of these presently ongoing VDRs, as already mentioned, is that they all concern SMR designs. For these, the Regulatory Document "Design of Small Reactor Facilities" (RD-367) contains the regulatory requirements and could be interesting as a template for the assessment of SMRs in JODA.

Another commendable aspect is the structured approach towards depth and technical content of the review, with the three well-defined phases. This is definitely a model to take into account for JODA.

The Canadian approach also seems commendable for its flexible format, including an contractual arrangement with the vendor and the establishment of a review project plan with timeline. At the same time, the core content of the review is well structured along the three phases and the content is predetermined with the 19 focus areas.

2.5 COMPARISON OF NATIONAL DESIGN PRE-LICENSING APPROACHES

2.5.1 GENERAL

When looking at the four national models presented in this chapter, the most striking difference is probably that of general approach, whether it is prescriptive and strictly defined or rather goal-oriented and flexible. This pertains to all aspects alike: scope, technical detail and binding nature of the assessment and the resulting statement. The US are on the far side of a prescriptive system, whereas the other three countries offer a more flexible and subtle approach.

2.5.2 SCOPE AND CONTENT OF THE REVIEW

The scope of the US Design Certification is exactly defined and corresponds to the range of design-specific aspects relevant for a construction licence. In the UK GDA and the Canadian VDR, the range of assessment is also expected to be "full scope" and comprehensive. In the GDA, the regulator expects that the GDA submission enables a "meaningful" assessment, but there is no mandatory list of assessment topics. Interestingly, in the GDA the scope is somewhat extended to issues concerning the future operator who must demonstrate development of the necessary "Design Authority". In Canada, there are 19 basic "focus areas" which the applicant can voluntarily extend by treating additional topics. The French RSO, in theory, is much more flexible since the scope is entirely dependent on the applicant's intentions who can submit "all or some" of his chosen safety options. However, in the only application of the RSO to a reactor design to date, the ATMEA1 process, the applicant submitted information on the full range of design issues.

Therefore, it can be summarised that in all approaches the assessment is normally "full scope" and broadly covers the entire range of generic topics associated with a reactor design.

More substantial differences appear when it comes to the depth and the level of detail of the assessment. Every assessment of a reactor design progresses from a very general level to a greater depth. In order to be able to compare the national approaches, it seems useful to correlate this process with generic phases of the assessment. Inspired by the "claim-argument-evidence" approach used in the UK GDA (see 2.1.2

above) as well as by the three phases of the Canadian VDR (see 2.4.1 above), the following grades for the depth of the assessment are proposed here:

- Grade 1: Evaluation whether the general plant design, according to the claims of the designer, complies with the main safety goals in relevant legislation and regulation.
- Grade 2: Evaluation whether the claims can be demonstrated to be based on compliance with basic requirements as contained in regulations.
- Grade 3: Evaluation of the main design features against more detailed regulations and codes and standards.
- Grade 4: Full assessment of the design against all applicable requirements as necessary for the issuing of a construction licence.

When looking at the four national processes analysed here, it is obvious that the US Design Certification corresponds to Grade 4. The prescriptive and rigid system of going into full depth in the assessment achieves the highest level of certainty embodied in the Design Certification, but it makes application for a Design Certification dependent on full design maturity and also creates huge problems with any changes to the design later on. The UK GDA process corresponds to Grade 3 which allows for all relevant issues to be evaluated up to a reasonable depth under the UK's Safety Assessment Principles, however allowing for some flexibility in the applicant's submission and leaving very detailed issues to the site licensing process. The Canadian VDR, in its Phases 1 and 2, would seem to reach up to Grade 2, whereas Phase 3 would carry some selected assessment topics up to Grade 3. The French RSO, finally, is again the most flexible approach; as the ATMEA1 review shows, it would seem that the evaluation comprises Grades 1 and 2.

2.5.3 PROCESS AND OUTCOME

In the US and in France, the process is laid out in legislation. By contrast, in the UK and in Canada, the process has been "invented" by regulators and defined in regulatory guidance. In spite of being rather "free-floating" from a legal perspective, the UK GDA and the Canadian VDR have proven to be effective and to be normally solicited by applicants. Another aspect is that in both processes, a contractual arrangement is entered into by the applicant and the regulator. In the US and in France, with their legally defined processes, such an arrangement is normally not foreseen (but has been done in the case of the ATMEA1 review in France, as explained above).

As to the binding nature of the pre-licensing statement, again there is the whole range of possibilities: whereas the US Design Certification is issued as an NRC rule and cannot be changed during its term unless there is a clearly defined cause, the Canadian VDR and the UK GDA are again, from a legal perspective, "free floating", with only an informal and non-binding expectation created by the regulators that the outcome will be taken into account in subsequent licensing processes. As indicated, this has not prevented the UK and Canada pre-licensing processes from becoming a successful instrument acknowledged by all stakeholders. As it appears, the goal of reducing regulatory risk and of creating confidence in the licensing process is effectively reached by such an informal approach. Indeed, it is difficult to see how a regulator could, without a good justification, reverse an evaluation he has performed in the near past; nor does it seem plausible that a regulator should have an interest in doing this, given the waste of resources and the risk of a loss of reputation it implies.

Looking at a multinational approach, this is an important message: a binding Common Opinion on Design Acceptability (CODA) would be very difficult to implement and indeed the binding character does not seem necessary to achieve the desired effect.

As to the details of the assessment and the process, the examples from the UK and Canada show that a contractual approach seems recommendable. It allows all parties to shape the process and to take over specific responsibilities and obligations; at the same time, it is not perceived to jeopardise neutral and robust assessment of the design by the regulator.

Finally, it is remarkable that both the UK and the Canadian regulator expressly "open the door", in their guidance documents, to the introduction of foreign design assessment results into the GDA and the VDR process, respectively. This is in line with their generally flexible approach, and it provides an important outlook to the international dimension of reactor licensing as discussed in this Report.

A summary overview comparison is given in Figure 4.

Figure 4: Summary comparison of national design pre-licensing processes

	UK GDA	**US Design Certification**	**French Review of Safety Options**	**Canadian VDR**
Purpose	Separation of design issues from site related issues	Licensing decisions finalized before construction	Clearance of design safety options chosen by vendor	Verification, at a high level, of the acceptability of a design with respect to Canadian nuclear regulatory requirements
Binding effect	Not legally binding but relevant in practice	Legally binding	"Opinion", but its use in subsequent licensing is indicated by legislation	Not binding but relevant in practice
Definition of design	Reference design configuration defined in Master Document Submission List	Design Control Document	Safety options dossier	Submitted documents; no mandatory template
Scope	"Full scope" to allow "meaningful assessment", plus some operator-specific aspects	"Full scope" excluding site and operational aspects	Flexible – applicant can submit "all or some" of the safety options he has chosen	Virtually "full scope", 19+ "focus areas"
Level of detail	Grade 3	Grade 4	Grade 1 or 2	Grade 2 plus Grade 3 for chosen topics
Safety Requirements	No prescriptive set of requirements; goal-oriented approach	US NRC rules – prescriptive approach	French regulations	Canadian regulations
Contract?	Yes	No	Depends	Yes

3 MULTINATIONAL APPROACHES IN DESIGN REVIEW AND THEIR CONTENT

In this chapter, multinational approaches for design review will be presented and analysed. As already indicated in the introductory chapter, these international approaches display an additional level of complexity as compared to the national pre-licensing examples discussed in the previous chapter. They have to deal with the same issues as national pre-licensing exercises (What is the purpose? How far can project-independent assessment go? How can results be entered into subsequent licensing processes?); beyond that, they also have to address issues of international cooperation. In particular, they have to overcome the additional hurdle of having to define common requirements and criteria against which the reactor designs can be assessed.

3.1 FRANCO-GERMAN EPR REVIEW IN THE 1990S

The Franco-German cooperation concerning the EPR in the 1990s is the topic of the Task 2 report; hence, reference can be made to the detailed description given there.

In the present context, it must be reminded that the Franco-German cooperation was about more than the review of a reactor design. It was triggered by joint development of a reactor design by the nuclear industry of both countries, and it comprised the development of common regulatory requirements and a joint set of codes. Both work-streams followed an iterative process where requirements on the one hand and the reactor design on the other hand were developed in parallel. Nevertheless, there are lessons to be learned for a joint pre-licensing process. Already in Task 2 the point was made that alignment both of regulatory requirements and of codes and standards would be an important element to facilitate a joint reactor design evaluation; therefore, an approach similar to the Franco-German collaboration in this respect would be recommended also for future collaborative projects and the Full Study (as defined in Task 2) should determine the steps to be undertaken to achieve such project results.

A commendable element of the Franco-German cooperation was the strong and well-organised implementation, based on a governmental declaration and on agreements between all parties. The successful creation of joint requirements is a good template for reaching an agreement, among the regulators participating in a Joint Overall Design Assessment, on the requirements and standards to be used; this task should be much easier today as compared to the 1990s, given the subsequent development of aligned standards under WENRA, IAEA and the (revised) Nuclear Safety Directive 2009/71.

This aspect has been particularly taken into account in the proposal for an implementation model (see section 5 below).

Even though the process was cut short by the German phase-out and the benefits were actually not reaped in Germany and only partially utilised in France, it can reasonably be supposed that implementation of a common statement on acceptability of the EPR would have been successful and would have led to deployment of a standard EPR design in both countries.

3.2 MDEP

3.2.1 GENERAL DESCRIPTION

The Multinational Design Evaluation Program (MDEP) is an initiative, started in 2006 with a Pilot Project Phase and in full working mode since 2008, taken by the regulators of originally 10, now 16 new build countries with the aim of leveraging resources and of identifying common regulatory practices. Members are the national regulators of: Argentina, Canada, China, Finland, France, Hungary, India, Japan, Korea, Russia, South Africa, Sweden, Turkey, United Arab Emirates, the United Kingdom, and the United States, plus the IAEA. The OECD Nuclear Energy Agency in Paris provides the Technical Secretariat.

MDEP's main objectives are:

- to enhance multilateral co-operation of regulators within existing regulatory frameworks;
- to encourage multinational convergence of codes, standards and safety goals;
- to implement the MDEP products in order to facilitate the licensing of new reactors, including those being developed by the Generation IV International Forum.

In order to achieve these objectives, two main lines of activity have been implemented:

- the exploration of opportunities for harmonisation of regulatory practices;
- the co-operation on the safety reviews of specific reactor designs.

The work in MDEP is done by different working groups. According to the two main lines of activity mentioned above, there are two kinds of working groups:

- Two issue-specific working groups: Codes and Standards, and Vendor Inspection Co-operation. A third working group on Digital I&C was recently (end of 2017) disbanded and its activities were merged into the NEA's CNRA (Committee on Nuclear Regulatory Activities).
- Six design-specific working groups: EPR, AP1000, APR1400, VVER, ABWR, HPR1000. The programme started with the first two; the following working groups were successively created in cases where at least three MDEP members were involved in a review of a given design. The design-specific working groups do not issue a formal overall statement about design acceptability; however they do publish "design specific common positions" on specific issues. Examples are common positions of the EPR group on "Digital I&C Design" or the "EPR Containment Heat Removal System in Accident Conditions".

Above the level of the working groups, MDEP policy and coordination is provided by the Steering Technical Committee (STC). The STC provides guidance to the design and issue specific working groups in order to ensure the programme of work is carried out to meet the MDEP goals, objectives, and expected outcomes.

3.2.2 SCOPE AND CONTENT OF REVIEW

Since there is no formal comprehensive statement of the results of design assessment as performed in the design-specific working groups and no official statement about the acceptability of the design, there is no fixed scope and content of design review. According to the MDEP document "Design-specific working groups (DSWGs) – General Terms of Reference" (Revision 7a, May 2016), each design-specific working group adopts its own programme plan which is then endorsed by the STC. However, the General Terms of Reference do provide for a "minimum set of issues" to be reflected in each working group's programme plan. This set of issues is listed in **Appendix 4** to this report.

As to the level of detail in the assessment, it seems this is highly dependent on prioritisation by the involved regulators. For some issues, assessment is carried to a high level of detail, particularly within the topics assigned to issue-specific working groups; digital I&C would be an example for an issue perceived by the regulators to justify a highly detailed treatment. For other issues, the depth of the assessment may be much lower.

3.2.3 RELEVANT REQUIREMENTS

One of the official goals of MDEP is to encourage multinational convergence of codes, standards and safety goals. This does not involve, however, that MDEP aims at directly achieving harmonisation in national standards, or at prescribing a set of standards for design assessment. Unlike WENRA, there is no focus on aligning standards and no commitment at all to implement common "reference levels" in national regulations. In principle, each regulator assesses the design against his own standards; this is feasible, given the fact that the joint design assessment does not result in a common opinion statement. Nevertheless, cooperation of regulators in issue-specific and design-specific working groups has resulted in some statements about common requirements. The STC in 2011 issued an MDEP Position Paper PP-STC-01 "MDEP Steering Technical Committee Position Paper on Safety Goals" which defines high-level requirements. Also, the Common Positions adopted by the working groups contain some agreement of participating regulators on what to require.

3.2.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

MDEP itself, in its official statements, puts particular emphasis on the key concept that national regulators retain sovereign authority for all licensing and regulatory decisions. As mentioned above, the work of the design specific working groups does not result in a comprehensive statement on the safety of the design. The specific results of MDEP work, such as Common Positions, do not have a legally binding nature and there is no official statement about how they are reflected in national licensing processes. In practice, however, it can be presumed that the impact of these results of cooperation within MDEP on subsequent licensing decisions by participating regulators will be considerable even if there is no formal link to MDEP results.

3.2.5 GOOD PRACTICE

Without any doubt, MDEP has been ground-breaking for cooperation of regulators in the assessment of reactor designs for new nuclear power plants. MDEP basically is a "club" of regulators who have taken the initiative on their own; there is no formal intergovernmental agreement underpinning their work. Therefore, from the start MDEP has followed a thoroughly pragmatic approach. The focus is on cooperation to increase effectiveness and efficiency of the regulators' work, to save resources in the assessment of internationally marketed reactor designs and to provide newcomer regulators with advice and support by their experienced peers.

Seen in the light of the objectives of ERDA and of this report, it has to be noted that MDEP has not given strict definitions of the scope of design assessment and has not identified the relevant requirements used for assessment. It has not attributed any legally relevant character to its working results and has taken care to underline that each regulator is entirely free in how he uses these results. On the other hand, MDEP has shown that such flexible cooperation appeals to regulators and its pragmatism can be said to be a success.

Concerning the scope and depth of design assessment, MDEP, while not prescribing fixed models, has at least indicated a minimum scope in the Terms of Reference of the Design-specific working groups.

A final aspect demonstrated by MDEP is that such regulatory cooperation needs the function of a secretariat which keeps the activities going and provides for a platform for communication and sharing of results. At the same time, the participating regulators clearly did not want to burden their pragmatic cooperation with too much formalism and the slow and cumbersome mechanisms of an international institution with hierarchies and permanent iterations in order to achieve consensus. Therefore, the founding regulators of MDEP decided to use the OECD-NEA as secretariat, as the NEA appears to be rather flexible and efficient in its work. For JODA, a similar secretariat function would have to be provided. This is discussed in more detail in chapter 5 below.

3.3 WNA CORDEL

3.3.1 GENERAL DESCRIPTION

The World Nuclear Association's CORDEL (Cooperation in Reactor Design Evaluation and Licensing) Working Group was established in January 2007 with the aim of promoting the achievement of a worldwide regulatory environment where internationally accepted standardised reactor designs can be widely deployed without major design changes. Its membership consists of experts from companies of the nuclear industry: vendors, operators, suppliers, service companies, engineering and consulting companies. It also welcomes representatives of international organizations.

CORDEL does not perform reactor assessment itself, but is rather a policy group which has developed approaches and initiatives for joint evaluation of reactor designs by regulators. The main policy document is the 2010 Paper "International Standardization of Nuclear Reactor Designs", the so-called "CORDEL Roadmap". In the Roadmap, a three-phase approach to harmonisation in licensing and to standardisation is proposed. The three phases are:

1. Share Design Assessment: Once a design is licensed in one country, the approving regulator should share information with other national regulators and receiving regulators should draw upon this experience. In the addition, if several regulators are concurrently reviewing the same design, they could form a team and discuss their assessment methodology (incl. criteria) and share their assessment results.
2. Validate and Accept Design Approval: Once a design is licensed in certain countries that are highly respected for their regulatory expertise, such design approval could be taken by other countries' authorities after a validation as sufficient for licensing there.
3. Issue International Design Certification: By international agreement, a procedure could be created whereby a design could be certified by a team of national regulators (from countries with a direct interest in the design). National regulators would remain responsible for assessing the adaptation of the internationally certified design to local circumstances and for the supervision of construction, commissioning and operation.

For the purposes of this Report, steps 1 and 3 are the most interesting, as they concern a joint reactor design assessment by several national regulators.

The 2010 Roadmap was preceded by a 2008 Report on "Benefits Gained Through International Harmonization of Nuclear Safety Standards for Reactor Designs" where the strong link between harmonization of requirements and the standardisation of reactor designs was explored and the numerous benefits of such harmonisation and standardisation were explained. After 2010, the CORDEL group followed up the Roadmap with several reports, two of which are relevant for our topic: the 2013 Report "Aviation Licensing and Lifetime Management – What can Nuclear learn?", where for the first time the licensing of aircraft, which involves a strong international cooperation of national and international authorities, was investigated with a view to nuclear; and, also in 2013, the report "Licensing and Project Development of New Nuclear Plants", which puts the focus on the interface between the licensing process and the industry's development of nuclear new build projects. For example, the report analyses the progress in designing a reactor through the steps basic design, detailed design and component specifications, and contains valuable insight into how this relates to contracting procedures and to licensing.

3.3.2 SCOPE AND CONTENT OF REVIEW

In developing its concept, CORDEL had to make a statement which part of an NPP the harmonised licensing process should apply to. According to the CORDEL Roadmap of 2010, this was the “core” that, beyond all issues of adaptation to site and operator, would be the same in all projects using this technology. It would cover

1. the global architecture of the plant and
2. a level of design which would be sufficient to prepare specifications for equipment procurement and which would enable the regulatory body to decide about the safety of the installation.

For the second point, which broadly concerns the depth/level of detail of pre-licensing design assessment, CORDEL made reference to the US-NRC rule 10 CFR 52.47 which defines the contents of an application for a design certification with, inter alia, the following wording: “The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant.” This would mean that the level of detail of a design assessment would be very high, corresponding to Grade 4 of the scale introduced earlier in this report (see 2.5.2 above).

By contrast, the WNA's 2013 report on Licensing contains some interesting observations on the level of design maturity needed for applying/obtaining a licence (similar considerations would apply to pre-licensing assessment). It emerges that the design maturity, as provided by industry, can be an important factor limiting the assessment during initial licensing steps – in some cases, elements of detailed design or procurement specifications are only finalised once the licence has been issued. This aspect would rather seem to limit the amount of detail to be submitted in a pre-licensing process. Altogether, it seems that the reference to a very high level of detail (corresponding to procurement specifications) in the CORDEL Roadmap is a maximum position inspired by the US system but in practical implementation CORDEL would probably also endorse a Grade 2 or Grade 3 assessment if appropriate.

3.3.3 RELEVANT REQUIREMENTS

CORDEL did not look at country- or region-specific scenarios and there is no particular recommendation on the standards to be used for reactor design assessment. Generally, the importance of harmonisation of standards as basis for the cooperation in licensing and the relevance of standardisation, by industry, of its reactor designs is very much highlighted. CORDEL has also put a particular focus on harmonisation of codes & standards, meaning the industry standards which are below regulatory requirements and serve as "translations" of these requirements (especially if the latter are goal-oriented) into detailed technical solutions which are acceptable to the authorities. CORDEL supports the ongoing work of international standardisation institutions to achieve alignment of national codes.

3.3.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

CORDEL proposes that the statement of the relevant authorities on the result of the design assessment is referenced in subsequent licensing processes. The general term used by CORDEL is "design approval" and CORDEL does not expand on its legal nature. By contrast, a so-called "design certification" would be one specific type of design approval, namely a stand-alone licence for a design, independent of a particular site or operator, which is to some extent legally binding on the issuing authority.

3.3.5 GOOD PRACTICE

CORDEL has pioneered, at the same time as the regulators with MDEP, the idea of a joint design evaluation by several national regulators, resulting in a joint statement. Quite obviously, the role of CORDEL was not as active as that of MDEP which actually did, to some extent, a design evaluation; CORDEL is more a policy group. In a certain contrast to MDEP with its very pragmatic approach and with its emphasis on not imposing any obligations on the participating regulators, CORDEL has been pushing for a more formal cooperation with effective mechanisms to implement the results in national licensing processes. This would certainly be a vital step to achieve harmonisation in licensing and assessment, similar to the aviation industry; however, it would seem challenging to implement such a system, based on international agreements and on adaptation of national legislation, in the short- or mid-term.

CORDEL has sketched the scope of the joint evaluation by defining the extent of a reactor design which should be standardised. In the 2013 Licensing paper, the WNA has additionally given valuable insight into the question to which extent industry can support a full-scope design pre-licensing, demonstrating some parallel issues such as detailed design work or contracting which may have an impact on the scope.

3.4 IAEA GENERIC REACTOR SAFETY REVIEW AND SAFETY STANDARDS

3.4.1 GENERAL DESCRIPTION

The Generic Reactor Safety Review (GRSR) is a service conducted by the IAEA staff and international experts to review the safety case of new reactor designs against relevant IAEA standards. Its aim is to enable the requesting party to understand to which extent the safety case is complete and comprehensive in addressing the requirements of the safety standards.

The GRSR has been performed for 14 designs so far (see https://nucleus.iaea.org/ sites/gsan/services/Pages/ Generic-Reactor-Safety-(GSR).aspx). Requesting party is a state member to the IAEA, not a vendor. The requesting party bears the cost of the process.

In one case (UK) the GRSR was requested for designs being at that time assessed by the regulator; obviously, the intention was to provide support and input to the national design evaluation process. In all other cases, it seems that the GRSR is requested by states for reactor designs developed by their national industry.

3.4.2 SCOPE AND CONTENT OF REVIEW

The review is conducted according to the relevant IAEA Safety Standards on safety assessment and on reactor design. The latter (SSR-2/1, Safety of Nuclear Power Plants: Design) addresses the topics which are deemed relevant by the Agency and therefore would define the scope/amplitude of the review process. A list of topics extracted from SSR-2/1 is reproduced in **Appendix 5** to this Report.

The level of detail is obviously limited, as the reactor design is reviewed against IAEA Safety Requirements which are rather generic. IAEA Safety Guides with more details are taken into account as appropriate but are considered to be only supporting information. The IAEA itself sees the purpose of the review in an "early" evaluation of a reactor design, and in verifying whether the IAEA criteria for assessment and safety of reactor designs are reflected in the safety case. Therefore, the depth of the review seems to correspond to Grade 1 of the scale introduced in 2.5.2 above.

3.4.3 RELEVANT REQUIREMENTS

The relevant requirements are the three following:

- SF-1, Fundamental Safety Principles;
- GSR Part 4, Safety Assessment for Facilities and Activities, rev. 1 (2016);
- SSR-2/1, Safety of Nuclear Power Plants: Design, rev. 1 (2016).

3.4.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

Based on the work by a team of internal and external experts, the IAEA issues a report which summarises the extent to which the safety case addresses the requirements. If needed, recommendations for improvement of completeness and comprehensiveness are provided. Attached to each report are two review sheets for GSR-4 and SSR 2/1, respectively. The sheets contain all requirements considered in the review and the reviewers' statements regarding their applicability and implementation in the safety case reviewed.

Since the IAEA is not a regulator, the report necessarily does not have a legally binding nature and there is no defined impact on national licensing processes. The report provides valuable information to the requesting state and the relevant vendor. A positive statement may also have a certain value for marketing the design.

3.4.5 GOOD PRACTICE

In the context of this Report, the most important feature of the IAEA process is probably the scope being used by the expert team, which is defined by IAEA Safety Requirements. It is rather generic, which also means that it fits all kinds of reactors. In any case, the IAEA Standards provide a firm, if generic, and transparent basis for the review. The review itself is carried only to a limited depth.

As to the process, it must be kept in mind that contrary to the other processes analysed here and contrary to the concept presented in this Report, the review is done by an entity (IAEA) which is different from the institution actually issuing a licence for a given NPP (national regulators). For this reason alone, the GRSR cannot be considered a pre-licensing, properly speaking.

3.5 GERMAN KONVOI CONCEPT

3.5.1 GENERAL DESCRIPTION

An example of a successful cooperation of regulators is the licensing of the last three German NPPs (designed by KWU) of the so-called Konvoi design in the 1980s in Germany. It may seem somewhat odd to make reference to a licensing process for German reactors in a section dedicated to international cooperation; but it must be reminded that in the German federal system, the licensing and supervision of nuclear power plants is entrusted not to any Federal regulatory authority, but to the *Länder*. The aforementioned nuclear power plants are located in three different *Länder*: Isar 2 in Bavaria, Neckarwestheim 2 in Baden-Württemberg and Emsland in Lower Saxony.

In the 1970s, the licensing of current reactors had become more and more costly and burdensome, mainly due to shifting concepts on the operator side and to constant changes in regulations and in safety assessment by regulators and TSOs. This had brought several projects to the verge of being abandoned half-way. With this experience in mind, all stakeholders agreed that the next three projects – which turned out to be the last ones actually constructed in Germany – had to be handled in a new, different way. The three operators involved, the three regulators and their respective TSOs and the designer KWU concurred that these projects were to be implemented in a uniform and harmonised fashion indicated by the name "Konvoi" (convoy).

In this concept, the three involved operators agreed on a uniform detailed design. This was not easy, given their different history and the different preferences and "philosophies" of the chief engineers. The operators committed themselves to submit, in the three licensing processes, safety analysis reports which would be identical apart from the chapter on siting, where local specificities had to be taken into account.

The three regulators, in their turn, committed themselves to a close cooperation. Most importantly, they declared their willingness to accept the results of evaluations performed by the TSOs of their fellow regulators. In parallel, the three TSOs (TÜVs) agreed to distribute the assessment work among them; it would suffice if each assessment task were to be done by one of them; the other two would endorse the results. This was laid down in written agreements.

Finally, the nuclear safety regulations – which are mainly in the remit of the Federal regulator, today the Federal Ministry of the Environment – were updated and consolidated on a very high level of safety. This provided a reasonable basis for the expectation that there would be no substantial changes in the short- and mid-term.

The Konvoi approach proved to be valid: the three NPPs were licensed in parallel within agreed timelines and went into operation in 1988/89. The harmonised design was a very successful one: all three NPPs have enviable operating records and they have been world champions in yearly nuclear electricity production several times.

3.5.2 SCOPE AND CONTENT OF REVIEW

In the Konvoi concept, there was no formal separate statement on design acceptability. Indeed, there was no pre-licensing properly speaking; instead, the positive evaluation of the design safety was one of the first and decisive steps of the licensing process. All three licensing processes were aligned and in each of them, the competent regulator and TSO made use of the assessments performed in the other two processes. Therefore, scope and content of design review were identical with the full scope assessment necessary for granting the construction and operating licence and thus with Grade 4 of the scale introduced in 2.5.2 above.

The mandatory elements of a licensing application and process are enumerated in the German Nuclear Licensing Procedure Ordinance (*Atomrechtliche Verfahrensverordnung*). The key document is the Safety Analysis Report (*Sicherheitsbericht*). The contents of the Report as expected by the regulator are described in a 1976 regulation "List of Contents and Structure of a Standard Safety Analysis Report for Nuclear Power Plants with Pressurised Water Reactor or Boiling Water Reactor" (*Merkpostenaufstellung mit Gliederung für einen Standardsicherheitsbericht für Kernkraftwerke mit Druckwasserreaktor oder Siedewasserreaktor*).

3.5.3 RELEVANT REQUIREMENTS

The licensing assessment was done against the relevant German Acts, Ordinances, technical regulations and codes in the nuclear field. As rulemaking, contrary to licensing and inspection, is a federal matter, this body of requirements is applied in a uniform fashion in Germany, regardless of the *Land* concerned. Therefore, for this example of cooperation of regulators, there was no need to perform any harmonisation or alignment of the standards to be used. Alignment was rather needed for the assessment methodology and the general "philosophy" of the participating regulators and TSOs.

3.5.4 NATURE OF THE STATEMENT AND RELEVANCE FOR SUBSEQUENT LICENSING PROCESSES

As outlined above, the Konvoi process did not involve a separate statement on design acceptability. An overall positive evaluation of design safety was integral part of the licensing processes which were managed in parallel, obliterating the need to define the interface between a pre-licensing statement and the actual licensing procedure.

3.5.5 GOOD PRACTICE

The German Konvoi approach displays some elements which could be useful for a European approach. The most important one is the cooperation between licence applicants (operators), regulators, TSOs and vendors, founded on written agreements and implemented by joint committees on several levels, resulting in effective sharing of assessment tasks and results. This could be a model for installing a JODA between several European regulators. Absent from the German model is the aspect of alignment of standards, which could nevertheless be performed on the same formal basis.

3.6 COMPARISON AND EVALUATION OF THE INTERNATIONAL APPROACHES

It is rather difficult to compare the international approaches presented in this chapter, given their heterogeneous character. There is an established process for international cooperation of regulators (MDEP), a strong cooperation of the regulators and TSOs of two countries (Franco-German EPR review), an example for cooperation of regulators within one country (German Konvoi), a design review performed by an international institution without regulatory powers (IAEA Generic Reactor Safety Review) and a policy statement by an industry group (WNA Cordel).

Whereas the German Konvoi approach was implemented within a strong national legal and regulatory framework, the prominent common characteristic of the other, truly multinational approaches is an inherent pragmatism, flexibility and non-binding character. None of the approaches is contained in a document of Public International Law, nor are the outcomes in any way binding for national licensing processes. The examples show that on the one hand there is a clearly perceived need for international design assessment, based on the different agendas described in the first chapter; on the other hand, there is reluctance to install fixed processes, mainly due to the desire of regulators to retain their full sovereignty and to the obvious difficulties of implementing a legally relevant system in a multinational context. Therefore, flexibility is a built-in feature of all of these international approaches.

With respect to scope of the assessment, all processes are based on a full-scope assessment. For the range of topics, MDEP provides a hugely relevant model with its Terms of Reference of the Design-specific working groups. The same is true for the IAEA review which is based on the Agency's own standards. When it comes to the level of detail, there is the full range from the IAEA's very generic assessment (Grade 1) to the detailed licensing assessment (Grade 4) in the German Konvoi process. MDEP again gives an interesting example with a varying degree of depth, depending on the needs perceived by the participating regulators.

A final interesting aspect concerns the participation of the vendors. Whereas the vendor, in all national processes analysed in the previous chapter, is a necessary party to the process, in fact starting it with his application, some of the multinational processes explained here do not attribute such a strong role to the vendor. In the practical approach of MDEP, vendors are involved and heard but basically the process is a discussion between regulators. The IAEA GRSR is officially requested by governments but it seems that they are sometimes acting on behalf their "national" vendors.

The following table in Figure 5 provides an overview comparison. The point must be made that the short summary can only be a very indicative presentation of the various processes.

Figure 5: Summary comparison of multinational design pre-licensing processes

	Franco-German EPR Review	**MDEP**	**CORDEL**	**IAEA GRSR**	**German Konvoi**
Harmonised requirements	Technical Guidelines and ETCs	Some harmonisation, otherwise acceptance on different grounds	To be harmonised	IAEA high level standards	National set of requirements
Outcome	Harmonisation and factual pre-licensing	No joint pre-licensing statement but practically effective cooperation	Design approval	Report with review sheets	Licence
Process	Strong project organisation backed by governments	Structure with secretariat but no intergovernmental agreement	Legal framework to be developed	Requested by Member States	Formal licensing plus contractual basis
Scope	Full scope	Minimum set of issues in ToRs	Full scope	Full scope	Full scope
Level of detail	Grade 3+	Varying, for some issues up to Grade 3+	Grades 2-4, as high as possible in the context of the project	Grade 1	Grade 4
Binding	Not legally	No	Yes	No	Yes

4 PROPOSED APPROACH TO THE SCOPE AND TECHNICAL CONTENT FOR A EUROPEAN DESIGN REVIEW

4.1 INTRODUCTION

As already set forth in the introductory chapter, a pre-licensing design review can, due to its inherent nature, only pertain to those elements of a reactor design which are independent of implementation of the design for specific nuclear power plants. As explained in the CORDEL Roadmap (3.3 above), a certain degree of adaptation of any standard reactor design will be necessitated by site- and operator-specific circumstances

Within this general limit, it has already been explained in the introductory chapter that the overall coverage of a pre-licensing reactor assessment can be split into two different aspects:

1. The scope in the sense of broadness of the assessment, meaning the range of topics to be covered; and
2. The depth of the assessment, meaning the question to which extent of detail the assessment is taken for each of the topics within the scope.

For each of these two aspects, in theory both "extensive" or "reduced" models could be envisaged. Clearly there is a connection between the extent of design assessment – and therefore the extent of certainty achieved via the final statement of the authority – on the one hand and the effort to be invested into this outcome on the other hand. A design acceptability statement with very restricted scope and depth could be rather "easy" to obtain but would be less meaningful for subsequent licensing processes than a statement based on great broadness and depth of assessment; on the other hand, the latter could be (too) difficult to achieve, especially in an international context. This leads to the observation that any Joint Overall Design Assessment (JODA) should be substantial and technical enough to effectively front-load national licensing processes but at the same time it should take into account the differences in licensing and regulation of the participating countries and should leave some room for assessment in the individual licensing processes. Hence, it should identify up to which technical depth it is reasonable and efficient to push the harmonization (versus what could remain national/operator dependent) and at which point it would become too cumbersome.

Another general observation is that the process should display a certain flexibility so as to accommodate the broadest possible range of designs with diverging technical features and philosophies – particularly with regard to SMRs with their various technologies. The process also needs to consider the different degree of progress in design development. As has been analysed in chapter 1, for vendors there may be different reasons and scenarios for applying for a JODA. In some instances, a finished detailed design might be entered into the process; in this case, the depth of the JODA is not limited by the design. In other cases, the vendor may seek a comparatively early review of a design concept in order to identify fundamental issues he would have to address in the further design work. The review of national and international models has shown that they suppose different degrees of design maturity or that they accommodate all options in this regard.

Generally, it seems that the design, and the scope of a design assessment, inherently has a subjective character – it is defined by the application. This is particularly pronounced in the French RSO but there are also subjective elements in the Canadian VDR and the UK GDA.

With this in mind, the following subchapters will deal with different elements of scope and technical content of a proposed JODA.

4.2 SCOPE OF THE ASSESSMENT, RANGE OF TOPICS

4.2.1 GENERAL

A Joint Overall Design Assessment (JODA) is about the assessment of a reactor design and a statement about its acceptability under applicable safety requirements. Such a statement is only achievable if the reactor design is evaluated as a whole, under basically all aspects relevant for nuclear safety. All national and multinational approaches are based, in general, on a full-scale assessment; the French Review of Safety Options (RSO) is the one exception which allows assessment of only selected safety issues. However, even the one RSO actually performed so far for a reactor design was voluntarily done on a full-scale basis. Therefore, the assessment within JODA should be founded on a comprehensive list of review topics covering practically the full scope of generic topics associated with a reactor design. This basic list should be open, however, for additional elements as applied for by the vendor.

Such a list could be established either in a generic document or in an individual agreement for a specific JODA. Both are feasible, and both may indeed be utilised in the process. A generic list would give general guidance based on an internationally agreed scope. Still, there should be room for adapting the standard list of topics to the individual assessment. The applicant (vendor) may also want to introduce additional topics. Therefore, the relevant scope (list of topics) for the individual JODA process should be agreed in a contractual arrangement (see below).

4.2.2 RANGE OF TOPICS

Starting point for the broadness of the assessment are the topics whose relevance for a design assessment are generally accepted.

Appendices 1 to 5 of this Report contain the five lists of review topics (designated as "focus areas for design review", "minimum set of issues to be considered for design review" or similar) identified as relevant in the depiction and analysis of national and international pre-licensing processes in chapters 2 and 3. These stem from the US Design Certification, the French Review of Safety Options (RSO), the Canadian Vendor Design Review (VDR), the Multinational Design Evaluation Program (MDEP) and the IAEA Safety Standard SSR-2/1.

All of these five lists are contained within one page each. This shows that it is a common approach to define high-level lists of assessment topics, but not to go into too much detail as this would depend on the individual features of the design and the circumstances of the application.

The five lists are different in details, in the order of issues and in the way of presenting them, but in general approach they are quite comparable, which is not surprising given the fact that in all the processes the regulators and TSOs are confronted with basically the same issue of assessing design safety against relevant high-level requirements which have undergone, on the international level, a certain process of alignment.

Starting from the five lists of topics created for national and international pre-licensing design reviews, the following structure of main elements is proposed for an *EU common pre-licensing process* (Figure 6):

Figure 6: Generic list of topics for the proposed EU common pre-licensing process

Generic list of topics for the proposed EU common pre-licensing process

Overall Design

- A description of the design, its safety functions and underlying considerations, and of the classification of structures, systems and components;
- An evaluation of the overall design against applicable safety requirements;
- The vendor's management system with respect to design development and quality assurance; vendor research and development programme;
- Fulfilment of principal technical requirements and engineering principles;

Specific plant systems

- Assessment of the various components of the reactor design, such as
  - Containment structure
  - Reactor coolant systems
  - Instrumentation and Control;
  - etc.;

Safety analyses

- Consideration of initiating events (external/internal); analysis of design basis accidents and beyond design basis accidents/severe accidents;
- Deterministic and probabilistic safety analyses;

Other requirements

- Consideration of other generic requirements such as
  - Radiation protection
  - Radioactive waste management
  - Early operation phase.

This generic scope allows for some adaptations for the individual assessment process. It represents a "core" of issues which a full-scope assessment needs to take into account in any case. It can be complemented with additional items if needed, leading to an "extended full scope" assessment.

Especially concerning requirements about external events and severe accidents, such a scope will take into account the lessons learned from Fukushima, as they have now, seven years after the accident, been consolidated in the revisions of such important safety standards as the IAEA's GSR Part 4, Safety Assessment for Facilities and Activities, rev. 1 (2016) and SSR-2/1, Safety of Nuclear Power Plants: Design, rev. 1 (2016).

4.2.3 DELIMITATION WITH RESPECT TO LICENSING PROCESS

Any pre-licensing process such as a Joint Overall Design Assessment (JODA) has to cope with the inherent issue of delimitation between the pre-licensing design assessment and the actual licensing process. In theory, the design assessment focuses on the design whereas the licensing process focuses on site- and operator-specific issues. In practice, there may be overlaps and grey zones:

- It is not always easy to separate site-independent and site-specific design aspects. Methodically, this is achieved by defining a "site envelope" with certain parameters which are used to assess the design. Certain envelope criteria can be used for performing the assessment, particularly with respect to site issues. These criteria should encompass a range of values for certain parameters which should be representative of the great majority of possible sites. In a subsequent licensing process, it would be demonstrated that the parameters associated with the particular site are contained in the envelope used for the design assessment. When defining the site envelope, there may be a choice between a more restricted and a more broad range of parameters.

- Similarly, there are some interfaces between design aspects and the assessment of the operator of a particular nuclear power plant. The UK GDA is an example where the future operator's capability plays a role already in the pre-licensing design assessment: in the GDA, arrangements for supporting future licensees to put in place a Design Authority have to be demonstrated, and the safety and security case produced in the GDA must be developed with a potential licensee’s legal duties in mind. Again, for a JODA there may be a choice for going a long way in including licensee assessment (like in the UK-GDA) or for keeping these aspects rather out of the assessment by leaving open spaces which are later filled in in the licensing process. The latter is preferable in terms of practicability, given the fact that the Common Opinion on Design Acceptability (CODA) is to be implemented in several countries that may have different ways of approaching assessment of the licensee's capability.

- Another way of addressing the various different options linked to particular projects is to present alternatives for assessment; in a subsequent licensing process, it would have to be demonstrated that the parameters for the particular project are contained in one of the alternatives.

4.2.4 DEFINING THE SCOPE IN THE CONTRACT

The range of assessment topics, while it should broadly cover the generic issues to be defined in JODA guidance, based on the IAEA Standard and the comparable lists used in the other processes analysed in this Report, should be defined in more detail individually for each JODA. The right instrument for this is the contract between the applicant (vendor and, if applicable, future operator) and the group of regulators/TSOs. In this contract, the parties should also determine the main envelope criteria. These should, of course, not lead to gaps in the core assessment of the design. As indicated, there is also the choice of genuine alternatives, meaning that variants of the design can be submitted to the assessment under different criteria.

Such individual agreement would also provide flexibility in terms of technology. This seems particularly relevant for SMRs. Whereas it should be possible to achieve a broad assessment of the safety of an SMR design with the list of generic issues as described above, certain technical features may require some modification. Besides, it may be necessary to adapt some topical issues and criteria to the specific characteristics of SMRs, such as their limited radioactive inventory, inherent safety features or modular system.

4.3 DEPTH OF THE ASSESSMENT AND LEVEL OF TECHNICAL DETAIL

4.3.1 GENERAL

Within the scope or broadness (range of topics) defined above, the next question is how far the assessment under each of these topics should go in terms of depth and technical detail. There are several aspects to this.

Ideally, the full benefits of design pre-licensing would be reaped if the design assessment goes to a great level of detail, like in the US Design Certification; in theory, this would mean that the subsequent licensing process would only deal with the design insofar as site-specific or licensee-specific aspects are touched. This would mean that the objective of arriving at a statement about the design acceptability would be reached to the fullest extent. This would seem to be a benefit both for regulators (who maximise the efficiency of the joint design review) and for operators and vendors (who achieve the greatest level of certainty and confidence).

However, national and international practice has shown that this is not always feasible. As already mentioned above, there may be different scenarios with respect to the maturity of a design entered into a Joint Overall Design Assessment (JODA). In some cases, a detailed design which has already been licensed in a non-Member State may be presented for assessment. In other cases, the vendor of a design which is yet in the designing process may seek early review. Besides, if full design maturity – a finished detail design up to component specifications – were required for triggering a JODA, there would be substantial issues concerning changes which, as it seems, cannot be avoided in practice during construction. The US Design Certification has shown that a too rigid depth of the design review creates huge problems in this regard.

Particularly in an international context, it may be an issue that the cooperating regulators may not want to do the entire depth of design assessment within the group, as the effort needed for coordination rises as the level of detail increases. Also, with more detail the use of safety requirements becomes a more intricate issue. A common assessment would suppose either harmonised requirements or an evaluation based on different requirements ideally leading to an acceptance "on different grounds". In the context of a JODA, harmonisation is already achieved within the EU for high-level requirements but further alignment remains to be done for detailed requirements and assessment criteria. Finally, a high level of technical detail in a JODA will increase the difficulties of integration of the results into national licensing processes.

4.3.2 SUGGESTED LEVEL OF DETAIL AND DEPTH OF THE TECHNICAL ASSESSMENT

Already for the evaluation of existing approaches, this report has introduced (in chapter 2.5.2 above) a systematic classification of various levels of depth of the assessment. The four Grades thus identified are:

- Grade 1: Evaluation whether the general plant design, according to the claims of the designer, complies with the main safety goals in relevant legislation and regulation.
- Grade 2: Evaluation whether the claims can be demonstrated to be based on compliance with basic requirements as contained in regulations.
- Grade 3: Evaluation of the main design features against more detailed regulations and codes and standards.
- Grade 4: Full assessment of the design against all applicable requirements as necessary for the issuing of a construction licence.

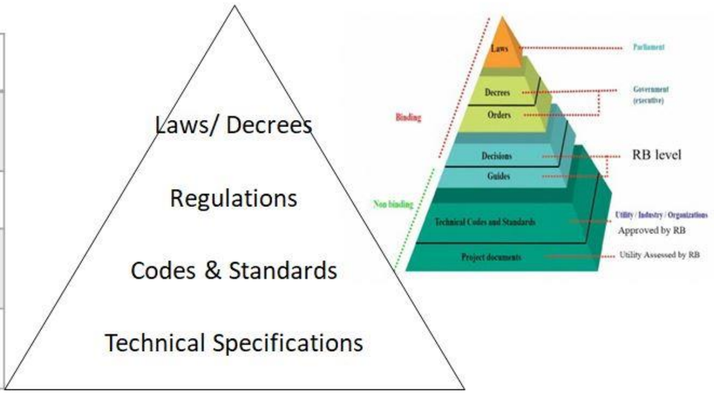
The four grades can – in a very broad fashion – be associated to the "pyramid" of nuclear safety requirements (as shown in Figure 4). Grades 1 and 2 would deal both with the high-level safety goals typically enshrined in laws and decrees and with regulations, at least those which are legally binding or those which the regulator expects the applicant to comply with in any case. For Grade 3, regulatory guidance documents would additionally come into play as well as codes and standards. For Grade 4, the assessment would be based on all of these documents plus specifications and other project documents generated by the applicant and validated, in whatever form, by the regulatory authority.

Of course, this correlation is only a broad sketch; much depends on the structure of national regulatory systems which can greatly vary.

The following Figure 7 shows the correlation of assessment steps or Grades, brought in relation to the UK GDA and the Canadian VDR, and of the "regulatory pyramid".

Figure 7: Yardstick for technical details (grading) in the assessment and corresponding requirements

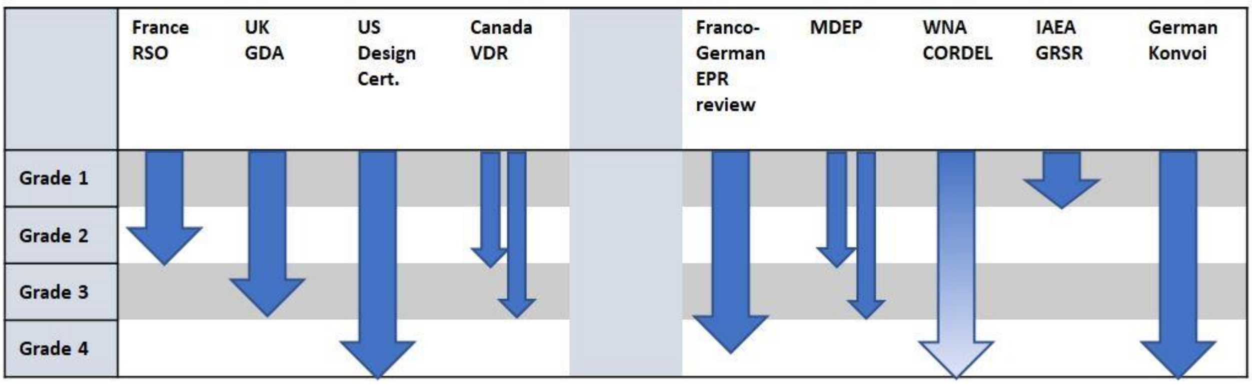
		UK GDA	Canadian VDR
Grade 1	Assessment of general plant design against safety goals	Step 2 (claim)	Phase 1
Grade 2	Compliance with basic regulations	Step 3 (argument)	Phase 2
Grade 3	Assessment of design features against more detailed requirements	Step 4 (evidence)	Phase 3
Grade 4	Full assessment of the design against all applicable requirements	(site licensing)	



In the same broad fashion, the four grades can roughly be associated with levels of reactor design maturity. Grade 1 would suppose (only) a conceptual design. A Basic Design would be the object of Grade 2 assessment. Grade 3 would call for the Detailed Design process to be already advanced to some degree, whereas Grade 4 assessment would necessitate a fully completed Detailed Design including the specifications for equipment procurement and all safety information which the regulatory body needs for the licensing decision.

The following table (Figure 8) applies the Grade approach to the level of detail employed in the national and multinational design evaluation processes analysed in the previous chapters.

Figure 8: Indicative level of technical details in the national and multinational pre-licensing processes



On the basis of this comparison and of the definition of Grades for the reactor design assessment and their correlation with requirements, a proposal for the technical content of a Joint Overall Design Assessment (JODA) can be derived.

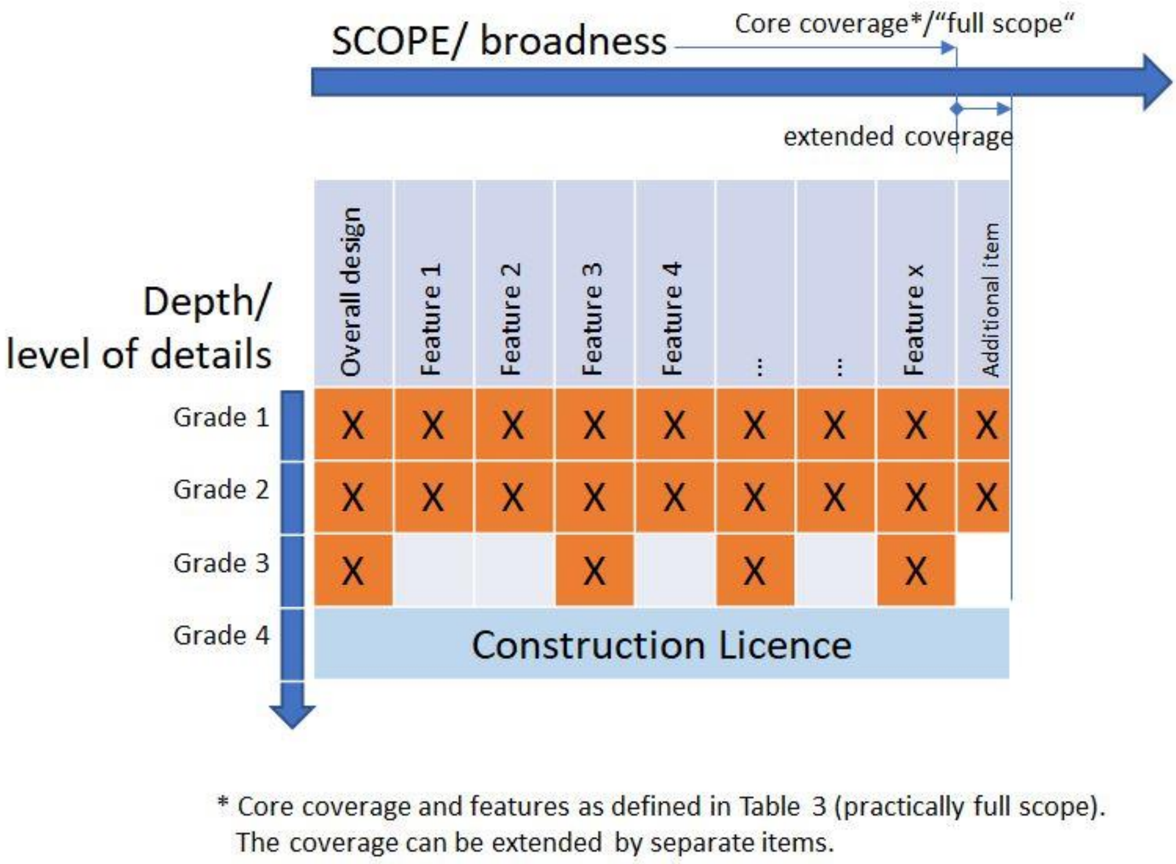
As already discussed above, a Grade 4 assessment would not be feasible in the context of a multinational design assessment such as JODA.

As to the minimum content of a JODA, Grade 1 – as exemplified in the IAEA Generic Reactor Safety Review – does not seem "meaningful" enough. On this level, it is merely assessed whether the vendor has made the right "claims" for complying with high-level safety requirements and whether the "claims" cover all relevant topics. This is the right level of detail for the purpose of the IAEA review which aims at an "early" evaluation of a reactor design. Within JODA, however, the joint assessment should be carried to a somewhat greater depth. At the same time, flexibility is also essential, as has been explained above.

This leads to the recommendation set forth in Figure 6 below. JODA should comprise, as a minimum, a Grade 2 assessment; it could, if requested by the vendor and accepted by the regulators, go further down in detail to a Grade 3 assessment. It could be envisaged to focus the Grade 3 assessment on certain topics, leaving other topics to the more general Grade 2 level. This is similar to the approach used in practice (though not expressly brought into a system) by MDEP (see 3.2.2 above). It may be advisable for selected topics where national regulations display many differences to assess them only up to Grade 2.

This proposed approach is flexible within a firm corridor of assessment depth. It allows assessment of reactor designs having completed the Basic Design stage or having been progressing into the Detailed Design.

Figure 9: Scope and technical details of the proposed EU common pre-licensing process



4.3.3 TECHNICAL CONTENT

As mentioned above, the technical content in each box of the approach shown in figure 6 – an approach which would appear both firm yet flexible – mirrors the requirements of the regulatory pyramid, although the top layer (laws & decrees) is not sufficient to reflect the grade 1 documents of the proposed JODA.

The details of the first layer (grade 1) can be labelled by the so-called "CLAIM" of the British approach which is also reflected in the Canadian design review: The requesting party must show at an overall level that the **design intent** is compliant with the regulatory requirements. The applicable requirements are in the first place high-level safety requirements which are, on the national level, typically enshrined in legislation or in government decrees. They are largely consistent in the EU and necessarily comply with the **Safety Objective** laid down by Art. 8a of the revised Safety Directive (Council Directive 2009/71/Euratom as amended by Council Directive 2014/87/Euratom). But also, more detailed regulations are to be considered. The **WENRA reference levels** are a suitable tool to embrace the national requirements for this exercise. However, in the proposed JODA documents of grade 1 it is not yet required to show that the requirements are met. In this phase the designer has to prove that he has taken into account all relevant safety goals.

In order to illustrate the level of technical details marked by grade 2 one has to note that here the designer has to show how the relevant safety goals are met based on the submitted design documents and supporting analyses. The aim of the review of the grade 2 documents is to find out if there are any fundamental obstacles that a reactor of this design can be licensed. In this exercise more detailed regulation may be utilised. In this grade 2 evaluation, still existing national divergences could appear. They could be solved by adapting and harmonising the national regulations at the occasion of a JODA or, if necessary, regulators could accept the relevant design features on different grounds. Therefore, it seems possible to carry the full-scope assessment to grade 2. After finishing the grade 2 phase the designer should be able to finalise his Preliminary Safety Analysis Report (PSAR) which is the main basis for the grade 3 review for the chosen features.

When assessing the level of technical details in grade 1 and 2 it seems advisable to submit the relevant documents and to review these progressively. In a first step, this would mean reviewing the documents of grade 1. If the outcome is positive, meaning that the **design intent** is compliant with the regulatory requirements, technical documents according to grade 2 can be submitted. Then, only if **no fundamental barriers** to licensing (so-called showstoppers) are found, documents according to grade 3 may be submitted. This corresponds to the phased approach of the Canadian VDR. “No fundamental barriers” means especially the existence of adequate acceptance criteria, compliance of the design with these criteria and the expectation that the remaining analyses and detailed engineering verifications can be done successfully.

While for grade 2 only the basic design with supporting analyses must exist, for the Grade 3 assessment the design’s (non-site specific) detailed engineering for the features under consideration must be completed. For these features the safety analysis based on the PSRA is the main content of the grade 3 submission. This involves regulatory guidance as well as industry codes and standards. Here, national differences may be more pronounced than at the level of grade 2. However, even at this level it seems possible to a large extent to reach a common ground for assessment. For pressurized water reactor a beginning could be the **“Technical Guidelines for the Design and Construction of the Next Generation of NPPs with Pressurized Water Reactors”**. These guidelines – based on the French-German **“Proposal for a Common Safety Approach for Future Pressurized Water Reactors” -** were one of the outputs of the successful Franco German cooperation in the 90s. This cooperation is a good example for a multinational licensing exercise (see chap. 3.1 and Task 2). A more general, but more recent regulatory document is the Canadian RD-337 **“Design of New Nuclear Power Plant”**. These regulations together with the guides about safety analysis (RD/GD-310, “Safety Analysis for Nuclear Power Plants”) are the basis for the corresponding phase 3 review in the Canadian VDR.

However, a full Grade 3 assessment for all issues may yet prove to be difficult on a multinational basis; it may be more reasonable, at the outset, to implement Grade 3 as an option for specific issues. The very implementation of the JODA concept may, in the course of time, lead to further harmonisation which would allow full Grade 3 assessment.

It must be noted that the described level of details is indicative. The regulators involved and the requesting party should agree on this upfront. The fact that there is a certain choice for the requesting party (vendor) which feature shall be included in the grade 3 review does not make the JODA arbitrary and does not lead to gaps in the assessment. Issues which are not (voluntarily) covered by JODA will be dealt with in full detail in the national process for issuing a construction license for a specific project. However, the upfront examination of certain issues in detail will give the requesting party regulatory certainty in technical areas where this is considered particularly relevant. Most important, it facilitates early procurement for certain components. Seen from the regulators' perspective, a detailed review of selected issues in grade 3 assessment may bring to fruition international cooperation with their peers in areas where the workload is particularly huge or where sharing of information and experience may seem particularly important. It also enables the regulator to become familiar with the design prior to the receipt of a licence application.

In summary the technical content of the proposed **JODA** is very similar to the Canadian VDR, transmuted to a multinational exercise.

4.4 THE CODA AS OUTCOME: MAIN ELEMENTS

A pre-licensing design review is a process; it results in some kind of statement by the participating regulator(s). As outlined in the introductory chapter, the written outcome of the JODA process is called, in this Report, Common Opinion on Design Acceptability or CODA.

Considering the experience from the comparison of national pre-licensing processes, the CODA should mainly comprise:

- A statement of the regulators' common opinion on acceptability of the design, which results from verification of its compliance with legal and regulatory requirements concerning nuclear safety. This "opinion" will not use legal wording but will be a strong statement. Wording could be somewhat similar to the French RSO where the authority states that the relevant safety requirements are "overall adequately taken into account" in the design and that at this stage the technical review "has not identified any significant incompatibility with the statutory requirements";
- A statement of the limits of the assessment, if necessary for each relevant issue;
- A statement of envelope criteria used in the assessment process;
- If necessary, a statement of issues to be further analysed in subsequent licensing processes. Here, the UK GDA is a useful model with the concept of GDA Assessment Findings. The findings are primarily concerned with the provision of additional evidence, after GDA, to confirm certain safety, security or environmental aspects as the project progresses through the detailed design, construction and commissioning stages. "JODA findings" could be a means, in the context of the international review, to collect and to define issues perceived by the individual regulators to be important.

Such "JODA findings" could be listed in the CODA for each regulator individually. More generally, a CODA could comprise a common opinion and, as a second chapter or appendix, individual statements by regulators, thus taking into account the remaining differences in national requirements and assessment methodologies. Of course, the aim would be to have a common opinion which is as large and broad and possible, and to restrict the individual statements to a very limited number of issues.

4.5 SAFETY REQUIREMENTS TO BE USED

For any design assessment, there needs to be a clear, consistent and fully identified set of requirements against which the assessment of the design is performed.

Within a national context, this aspect is normally not an issue since the requirements, both in legislation and in regulations, exist. It becomes crucial, however, for a multinational pre-licensing assessment, where the set of safety requirements to be used would have to be defined in advance.

There is as yet no complete set of EU safety requirements. However, there is a web of different layers of requirements which effectively combine to a coherent system. This structure of various aligned requirements has been extensively discussed in the Task 1 Report. It mainly comprises the nuclear safety objective for nuclear installations laid down in Art. 8a of the Safety Directive (Council Directive 2009/71/Euratom as amended by Council Directive 2014/87/Euratom) and the WENRA reference levels; also, the IAEA Safety Standards are relevant which have been upgraded in the past years and are considered highly relevant by all stakeholders. On the industry side, this system is complemented by EUR.

On these bases, the regulators of EU Member States should be able to identify a common set of requirements for design assessment. The process for achieving this is explained in 5.3 below. Some conceivable input for the development of such common set of requirements is mentioned in Chap. 4.3.3.

At the level of more detailed requirements, it may be envisaged that each participating regulator, to an extent clearly defined in advance, uses his own requirements for design assessment. Ideally, this would not compromise a joint assessment with a joint methodology; only the results of this assessment (e.g. a calculation) would be evaluated by each regulator against his own requirements. In practice, the vendor would strive to demonstrate that his design fulfils each of the different national requirements at the same time. This is a common element of licensing in the aviation sector: large aircraft designers such as Boeing or Airbus design their planes to an envelope of the relevant national requirements so that the same design can be licensed in the US and in the EU according to the relevant national or regional airworthiness codes.

4.6 APPLICATION OF JODA TO DIFFERENT DESIGNS

One of the prerequisites contained in the tender specifications for this report was that the scope must take into account different types of reactors. Given its inherent flexibility, the JODA/CODA approach satisfies this requirement. Its scope, based on generic documents such as the IAEA SSR-2/1 or the list of issues contained in the MDEP Terms of Reference for design-specific working groups, is not design- and technology-specific and is therefore able to take into account different types of reactors. It leaves the way open for PWRs and BWRs, evolutionary and revolutionary new designs, Generation IV designs and Small Modular Reactors (SMRs).

As has been outlined in the introduction, a JODA would be particularly relevant for Small Modular Reactors (SMRs). As many SMR designs are built on the concept of factory-built modules shipped to the site and assembled there, standardisation is absolutely essential. Their deployment will practically be impossible if a licence according to different standards is necessary in every country, thus deleting the very concept of modularisation. Since there is yet no practical experience with SMRs and no established national practice, a first-of-a-kind (FOAK) SMR would be an excellent showcase for a JODA and the benefits of a shared assessment would be brought to bear to a particularly large extent.

The scope of design assessment would be sufficiently generic to encompass SMRs. Taking into account the principle of graded approach, demonstration of some safety requirements may be more straightforward for SMRs, given their reduced power and radioactive inventory and, depending on the design, inherent safety features. On the other hand, some features of the very diverging SMR concept may be unusual as compared to large NPPs where light water reactors have evolved as the most implemented technology by far; a common pre-licensing could be particularly helpful in this respect. Valuable input for SMR assessment can be gathered from national regulatory guidance such as the CNSC's RD-367: Design of Small Reactor Facilities (see 2.4 above).

5 IMPLEMENTATION OF A EUROPEAN DESIGN REVIEW

5.1 WHEN TO CONDUCT A JODA

Actual implementation of a JODA for a given design would suppose that several regulators are tasked with assessment of the design at roughly the same time. Such a situation would arise if the design is the object of licensing applications in several Member States, or if at least there are projects under way supporting the credible expectation that the design will be implemented. As a result, there could be a fleet of reactors of the same design being built in several Member States.

An interesting situation arises if a regulator is involved in a preliminary design assessment during the selection stage for a national programme. In such a situation, the regulator may choose to participate in a JODA conducted by other regulators where the design is in the (pre-) application phase. In these cases, the regulator may only be interested in a grade 1-2 assessment; he could participate to an extent limited to such assessment.

If a given design is to be implemented in one Member State only or if implementation in further Member States is still uncertain, a JODA would not seem to provide an added value. Instead, there may be more suitable means for the regulator tasked with assessing the design – especially if it is the regulator of a somewhat smaller country – to derive benefits from international cooperation, e.g. to collaborate with a (EU or non-EU) regulator who has already licensed a power plant of this design and to make use of the reference plant concept (see INSAG-26, Licensing the First Nuclear Power Plant, 2012).

5.2 GENERAL APPROACH

For implementation of the Joint Overall Design Assessment (JODA), three options could be envisaged:

- A fixed process based on EU legislation, the result being legally effective in the relevant Member States;
- A fixed structure involving a JODA organisation without legal powers, created by a Commission decision;
- A flexible ad-hoc process facilitated by the Commission.

The first option would require EU legislation, both for installing the evaluation process according to uniform standards and for creating, in the national legislative and regulatory systems of the Member States, the prerequisites for mandatory acceptance of the Common Opinion on Design Acceptability (CODA) and its legal status as obligatory reference for the national licensing processes. In this version, JODA would not be merely a pre-licensing process, it would actually supplement parts of the licensing processes to be conducted in the relevant Member States.

However, this is expected to be extremely difficult. EU legislation would be addressed only to Member States with a nuclear programme; and even some of these may not necessarily be interested and willing. The necessary changes in national legislation, in order to secure legal effect of the CODA in national licensing processes, could in theory be made obligatory via a Directive. However, given the different national licensing systems and diverging approaches to administrative law and administrative process in general, this would be extremely complicated and would probably meet considerable reluctance on the part of the Member States.

Finally, a huge issue is participation of the public. If the CODA were to legally pre-determine the outcome of the reactor assessment during the national licensing process, participation of the national public during licensing would arguably be deemed to be void of substance in this respect. As compensation, there would have to be legally binding participation rights during the JODA process which would be extremely difficult to implement in the multinational context.

Therefore, it is proposed here to install JODA in a more flexible way, using the second or third of the options presented above. According to the second option, the Commission could create, via a decision, a standing body comprising the regulators of the Member States. A model for this could be the Commission decision 2007/530/Euratom establishing the European High Level Group on Nuclear Safety and Waste Management (now ENSREG). Within this standing body, JODA reviews could be organised on individual terms. The other option would be to offer a voluntary and flexible approach to interested Member States, with JODA groups being set up *ad hoc* for each assessment. This process could be facilitated by the Commission both in terms of organisational support (Clearinghouse) and of guidance. The choice of one of these two options is basically a political one and cannot be anticipated in this report.

In any case, the JODA process should be based on guidance/terms of reference defining the procedures, the scope, the use of the Common Opinion on Design Acceptability (CODA) etc. This guidance would be provided by the standing JODA body, if applicable, otherwise by the Commission; in the latter case, the Commission could work with a informal "pool" of interested Member States.

Whatever the option chosen, the generic programme should be adapted case by case to each individual JODA. The best way to achieve this is via a contractual arrangement concluded for each JODA between governments, participating regulators, their TSOs (if applicable) and the vendor (and, if applicable, the operator). Models for contractual arrangements could be the French Review of Safety Options performed for the ATMEA design, the German Konvoi licensing, the Canadian VDR and the UK GDA. More details on the contract are given below in 5.3. The EU Commission, and EU expert bodies designated by the Commission, would play an important role as "Clearinghouse", coordinating the process.

It would seem a matter of political decision whether regulators of non-Member States should be offered the option of participating in the concept as a whole, or in individual JODAs. This would not seem far-fetched at least for those countries with whom the EU has a working relationship in nuclear safety, such as Switzerland or the Ukraine, or for a post-Brexit UK.

5.3 PREPARATION, APPLICATION AND START OF THE PROCESS

An individual JODA process would start with an application by the reactor design vendor, in a constellation where a reactor design may be built in several Member States. Since this is proposed to be a voluntary and flexible process, it does not make much sense to define very strict application criteria. If there is no real prospect of implementation of the design, the regulators will not be interested in performing a JODA.

The vendor may team up with operators who plan to use the design for a new build project. This would not only enhance the credibility that the design might really be implemented in the participating countries. Experience has also shown that operators have an important role in achieving, or in preventing, standardisation of designs; it may well be that different operators have to agree on common requirements and expectations in advance in order to be able to order a standard design. On the other hand, an additional level of complexity is added if operators from different countries join the process. Therefore, participation of operators in the application should not be made mandatory and it should be decided on a case-by-case basis whether they are involved or not.

Since this is a voluntary process not backed by legislation, it is not envisaged that an application (which satisfies all requirements) gives the applicant a claim that a JODA process is effectively started. Effectively, all participating stakeholders need to agree that the JODA gives added value.

The application could be filed with the EU Commission, which then initiates a process involving the relevant countries.

5.4 ESTABLISHING COMMON GROUND FOR EVALUATION

In comparison to the situation of the Franco-German cooperation (as explained in Task 2), the alignment of safety standards has gone a long way, worldwide and particularly in Europe. A major part of the Franco-German process, namely the development of common requirements, would today be shorter and less difficult. Starting point would not be several very diverging sets of national regulations and standards, but a largely uniform understanding of high-level requirements as shaped by the EU Safety Directive, ENSREG, WENRA and IAEA Safety Standards (see 4.5 above). This does not obliterate the need for an agreement on a common set of requirements but it makes this step easier. On this basis, the regulators could focus on agreeing on requirements on the detailed implementation level by agreeing on a common set of standards. There would also have to be a reconciliation (if appropriate) of different national "philosophies" and "schools of thinking" which have evolved over decades in experienced nuclear countries, as well as an agreement on assessment methodologies and on the way TSOs deal with reactor design evaluation.

Most likely, designs entered into a JODA process will come from outside the EU. Participating regulators would have to perform an assessment of foreign codes and whether they can be used for deployment of the reactor design in the participating countries. It must be reminded that codes and standards are not legally binding, so there would be flexibility for participating regulators; but obviously the goal would be to achieve a comprehensive common position. To achieve this objective and to demonstrate that it can be done, experience from the Franco-German ETCs can definitely be utilised.

There may be a need to develop a common understanding for a limited number of essential issues with a potential strong impact on the development of the project. Here again, the Franco-German collaboration may be used – with some modifications – as a template. In a similar way, the issue-specific working groups of MDEP (see 3.2 above) have demonstrated both the fact that particular aspects quickly reveal themselves to be of crucial importance, and the way how to tackle these aspects.

With a view to the Franco-German collaboration, a scheme for resolving such issues among the participating regulators could look like this:

- Use of a questionnaire to identify and address these issues;
- Common technical meetings on expert level to achieve clarifications, with the involvement of the applicant;
- Final clarification meetings on senior expert level, resulting in a report with recommendations;
- Meetings of a policy/steering group;
- Final decisions regarding all issues by the regulatory authorities and communication within the respective Member States of these decisions.

With such an approach to evaluation standards, it seems that a common ground on the requirements used for evaluation can be reached. As indicated in 4.5 above, it could additionally be envisaged that regulators accept some design features on different grounds, based on their national regulations.

5.5 CONTRACTUAL BASIS FOR COOPERATION

For each individual JODA process, a contract should be concluded. This could be based on a template provided by the EU Commission.

Parties to the contract should include at least the participating regulators and their TSOs (if applicable) as well as the applying vendor (plus operator, if applicable) and the EU Commission. Ideally, the governments of the participating countries should also become parties. This would result in a certain oversight and control of the process outcomes by the governments and would introduce an additional extrinsic motivation for the regulators to perform the JODA in an efficient and effective manner with a meaningful outcome.

The contract could contain the following provisions and definitions:

- The participating parties: governments, regulators, TSOs (if applicable), the vendor of the reactor design, future operator(s);
- The working structure of the participating organisations as well the terms of reference for the supporting (project) office;
- The relevant set of safety requirements and, if necessary, the process to define those requirements which are not common at the outset (see previous section);
- Scope and technical detail of the assessment, particularly for grade 3 assessment which can be restricted to chosen issues;
- The major steps of the process, particularly with a view to the stepwise grade 1-3 assessment and the possible issuing of interim statements;
- Milestones and projected timeline;
- The broad allocation of assessment work to each of the participating regulators/TSOs;
- Financing (obviously, the applicant would have to bear the cost of the process).

It must be stressed that contracts are an important element of administrative law and that a contractual arrangement of this kind does not at all predetermine the outcome of the assessment and does not result in the regulators being biased. As the French regulator ASN stated at the occasion of the contractual arrangements for the ATMEA assessment, the agreement had no impact on the technical analysis to be completed; the analysis was carried out under the same conditions as under an administrative licensing process for a nuclear installation in France (see 2.3.1 above).

5.6 PROCESS

Based on the contractual arrangement, the JODA process with the joint evaluation by regulators would start. Regulators would meet in working groups on different levels; there would be a Steering Committee and working groups for various elements of the assessment.

As explained above in 4.3.3, based on the experience gained from national pre-licensing processes, it seems advisable to perform the assessment in several steps with progressing depth of evaluation. These steps should be aligned with the grade 1-3 assessment. From a practical viewpoint, it may seem reasonable to envisage some sort of statement at the end of each step.

Even if this is a transitory cooperation, focussed on the achievement of a CODA, experience has shown that there is need for a standing entity to perform the function of secretariat (such as the OECD-NEA for MDEP). The secretariat would ensure the necessary coordination, would supervise compliance with agreed timescales and would incentivise and facilitate the joint assessment work. In cases of conflict, the secretariat would mediate and, if necessary, escalate the issue to chief regulators in order to achieve quick and efficient resolution.

The secretariat function could be performed by the EU Commission or by an expert body designated by it, such as the JRC Petten.

The regulators would basically define by themselves how the assessment work is best distributed among them. It would make sense that individual regulators take over the assessment of issues where they have specific experience and expertise or which, by tradition, are specifically relevant in their national licensing processes. Some regulators rely, in practice, on support by national TSOs (such as France's IRSN); these would be included in the process.

In spite of this efficient sharing and distribution of workload, it must also be stressed that one of the main aims of design assessment performed by a regulator is that the regulator becomes familiar with the design so he can fully discharge his role in licensing, supervision and enforcement. Therefore, it is essential that all participating regulators understand all assessment steps, even those performed by the others. An exchange of information and open discussion is essential.

This is also true for another reason: each regulator is held fully accountable, in his home country, for his licensing decisions, for the evaluations and recommendations he provides to his government, and for the information he gives to the public. For this reason, each regulator must be fully aware of all steps of the assessment process. To give an example: if the regulator in country A issues a licence and this licence is challenged in court by a member of the public or an NGO because of a safety topic which is held to be unsatisfactorily addressed, the regulator must himself defend his decision in court; he cannot call upon the regulator of country B to come to court and testify how he has evaluated the critical issue.

Therefore, it is crucial that all participating regulators fully endorse the CODA and are aware of its full content.

Even if the JODA process as sketched here is not legally binding, participation of the public should not be neglected. Today, public participation is considered an essential part of licensing and design assessment. Therefore, there should be mechanisms to provide that the public in the participating countries is provided with relevant information and that people can have their say on the evaluation. The UK GDA process may be a good template for this. Since some elements of the assessment pertain to proprietary or commercially sensitive information, it may sometimes be necessary to provide summaries or excerpts instead of full texts.

5.7 THE OUTCOME: COMMON OPINION ON DESIGN ACCEPTABILITY (CODA)

The outcome of the JODA, which is a process, would be the Common Opinion on Design Acceptability (CODA).

As already discussed, it does not seem realistic to define, as outcome, a legally relevant decision to be mandatorily implemented in the participating Member States. This is why the statement has been designated as a (common) "opinion". This is modelled, for example, on the French Review of Safety Options which also results in an opinion issued by ASN. As has been explained above, the characterisation as "opinion" does not prevent the ASN statement from having a specified effect on subsequent licensing processes.

In the context of JODA, it is very important that the outcome (CODA) is a "common" opinion, jointly supported by the participating regulators. As explained above, it could be envisaged that the individual regulators can express some reservations on single points and define issues to be clarified in the subsequent national licensing process. Quite obviously, however, this should not be exaggerated and should not jeopardise the overall character of the CODA as a joint statement. The opinion should also be "common" in the sense mentioned above that it is endorsed, with full knowledge and expertise, by all regulators.

Since the process is not mandatory and there is no legally defined way of making the CODA obligatory for licensing processes in Member States, the Common Opinion refers, in the CODA concept, to the design "acceptability" and does not result in a binding "acceptance". It is expected, however, that regulators will not arbitrarily overturn their statement later on; in practice, issues considered in the CODA will be seen as settled unless new information comes up. In the national and international evaluation programmes examined above, there was no apparent case where the regulatory statement, even if it did not have a legally binding character, was later overturned or disregarded without cause.

The characterisation as statement on "acceptability" also indicates that the legally binding decision on acceptance of the reactor, along with the licensing decision on giving green light for the nuclear power plant as a whole, will occur later in the national process, under national procedural rules and participation of the national public under formal arrangements.

The CODA should specify a period of validity, for example 10 years. It should also contain a list of the reference documents used which is thus "frozen" for the period of validity.

There should be one CODA at the end of the process. However, as mentioned above, it seems reasonable for the participating regulators or for the Secretariat/Clearinghouse to issue interim statements when one of the assessment grades has been successfully completed.

5.8 REFERENCE TO THE CODA IN NATIONAL LICENSING PROCESSES

For any nuclear power plant constructed and operated in an EU Member State, a national licensing process must be conducted. In a legal and regulatory perspective, JODA is a pre-licensing exercise and the CODA does not constitute a licence in a legal sense. It does not entitle the designer or a future operator to construct or operate a nuclear facility; the regulatory green light can only be given by the national regulator.

As discussed in the first part of this section, it is not recommended here that the CODA is given legal effectiveness, making it by itself a mandatory reference for the following national licensing process. Nevertheless it should have a meaningful impact on national licensing processes.

In countries which feature a pre-licensing design assessment in their regulatory system, such as France with its Review of Safety Options, the CODA could be copied and pasted into a national regulatory statement which fulfils the criteria for such a design assessment, provided that the national process has a comparable scope and content. In France, for example, the CODA could be transformed into an ASN "opinion". As a consequence, it would have the effect on subsequent licensing processes as specified in French law for such an opinion. This would be a straightforward solution.

In countries without such pre-licensing design assessment, it seems sufficient, within the flexible approach recommended here, to rely on the factual effect of the CODA. It is to be expected that regulators take the CODA into account when issuing the construction and operation licence for a nuclear power plant. As discussed above, there is normally no reason and no incentive for a regulator to change a well-founded evaluation unless new information arises.

5.9 HANDLING OF SUBSEQUENT CHANGES TO THE DESIGN

Experience from national and international pre-licensing processes has shown that subsequent changes to the design, after the acceptance statement has been issued, are a real issue in practice. There are various reasons for this. Design development, from Basic Design to Detailed Design and the definition of procurement specifications, is often a process conducted in parallel with pre-licensing activities and with commercial progress in project development. Also, changes to the Detailed Design may become necessary in the implementation of the design for a specific project, based on site conditions and on specific considerations of the operator.

Such changes are, as such, in a certain contradiction with the basic aim of a JODA to reach a statement on acceptability of a specific design as presented to the regulators by the applicant; ideally, the design would be "frozen" at this stage and implemented without changes. This is one of the reasons why in this Report, the depth of the assessment is recommended to include Grade 3 as a maximum (see chapter 4.3.2 above), since there will be many modifications in those aspects assessed under Grade 4, whereas Grade 2 and 3 issues may not be liable to change to this extent.

Since, however, changes may nevertheless occur, there are two possible ways of dealing with them:

- For subsequent changes to the design, the vendor could be obliged to apply for a modification of the CODA.
- The opposite option would be to leave the CODA as it is and to deal with the changes in the subsequent national licensing processes.

The first option (modification of the CODA) is more stringent with a view to the aim and purpose of the pre-licensing process and ensures continued standardisation of the design. However, it is very difficult to implement. It could seem to compromise the status of the CODA if the CODA is subject, over time, to a number of modifications. It requires huge effort to re-start a (limited) JODA where regulators jointly assess the design changes. And, in case the change is triggered by considerations within a specific project, driven by site- or operator-specific circumstances, the other regulators may not see the necessity of amending the CODA if this is not relevant for "their" projects.

Therefore, it is recommended here that design changes affecting the CODA are dealt with in the individual national licensing process. To the extent the design is modified by these changes, the CODA loses its pre-licensing effect and the issue would have to be fully evaluated by the national regulator.

6 SHORT EXEMPLARY SCENARIOS

Considering large NPPs, new build within the EU is unfolding in a rather restricted way, so it is not easy to develop an exemplary scenario on the basis of current developments. Looking into the future, SMRs and Gen IV designs may give a new impetus, but under circumstances difficult to predict today.

6.1 LARGE-SCALE NPPS

The situation that several countries plan to implement the same reactor design can most likely arise for Member States in Central and Eastern Europe with the current Russian VVER 1200 design (also called AES-2006). Hungary has taken steps to implement the Paks-II (units 5+6) project and start of construction is expected soon; this project, if it actually proceeds, may be at a too advanced stage to profit from joint pre-licensing. The Czech Republic and Poland are contemplating nuclear energy projects in the mid- and long-term; the Czech Republic as replacement of existing reactors, Poland in the framework of a new nuclear energy programme. These two countries could join forces to perform a joint assessment of the AES-2006 as well as of any other design which could come into play. They could be joined by the Slovak Republic which seems to be contemplating a new unit at the Bohunice site, provided this project proceeds.

Bulgaria may be another candidate to join this cooperation; however, the prospect of new build seems currently muddled, as projects for new units at Belene and Kozloduy have not been pursued effectively. It is difficult to predict how this situation will evolve.

In this scenario, it may also be envisaged to include Finland with its regulator STUK. Finland has an operating VVER in Loviisa and there is a new build project in Hanhikivi implemented by Fennovoima, based on AES-2006 technology. Site preparation has started and Fennovoima expects the construction licence to be issued in 2019. STUK arguably does not need to cooperate with other regulators, as it is very familiar with the technology and the Hanhikivi project is already advanced. However, STUK has always been very active and responsible in the international context and has displayed a keen interest in supporting small or newcomer regulators in the interest of nuclear safety worldwide. In our context, STUK might be invited to join the design evaluation in order to support the other regulators.

6.2 SMRS

A fairly recent development on the international nuclear market is the drive towards small modular reactors (SMRs). SMRs are defined as reactors with not more than 300 MWe. Concepts are very different but one important feature common to most designs is that they consist of factory-built modules shipped to the site and assembled there. As has been pointed out, with this concept SMRs are particularly vulnerable to diverging national requirements and they are largely dependent on harmonisation and standardised implementation. Therefore, they are natural candidates for international pre-licensing. Another strong argument for international pre-licensing is the lack of experience worldwide in licensing and constructing SMRs; a first-of-a-kind (FOAK) licensing process would definitely benefit from a strong cooperation of several regulators who join their forces. The fact that there is no national established licensing practice for SMRs may also increase the willingness of the national regulators to participate.

A great variety of SMR designs is currently being developed all over the world but so far actual implementation is somewhat stalling. It is difficult to predict whether there will be a wave of SMR new build or whether the concept will not really take off commercially. Advantages and downsides of SMRs, especially in an economic context, cannot be discussed here. In any case, many governments, for example in UK or in the US, are taking an increased interest, considering SMR development as boosting their high-tech industries.

In an SMR scenario, several EU countries such as Poland, the Czech Republic or France, plus eventually the UK (consequences of Brexit to be evaluated), could team up to evaluate several SMR designs, both domestically developed (e.g. in France or the UK) and from foreign countries. If SMRs really turn out to be an attractive option, it could be envisaged that a larger number of Member States participate and that a range of SMR designs is jointly assessed and subsequently selected for implementation by the participating countries according to their specific needs (e.g. electricity generation or process heat for neighbouring industrial facilities). In proportion to the number of countries and designs, huge synergy effects could be reaped.

6.3 GEN IV DESIGNS

In a more distant perspective, JODA could be envisaged for pre-licensing of Generation IV reactor designs. These are advanced reactor concepts considered to be able to supply clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while ensuring high levels of proliferation resistance and security from terrorist attacks. Research and development is coordinated by the Generation IV International Forum (GIF) to which Euratom is a member.

As none of the six Gen IV designs chosen for R&D by GIF is near to detailed design and to actual deployment, it is difficult to predict when and how implementation of Gen IV reactors will occur. In any case, such implementation calls for an international framework where countries agree on common design requirements for this new generation of reactors. MDEP in its official scope mentions Gen IV designs in a prominent position; in practice, MDEP work has so far focussed on Gen III (or Gen III+) designs which are ready for implementation. Quite obviously, JODA could be a tool for joint evaluation, among EU Member States, of Gen IV designs.

7 CONCLUSION

This Task 3 report deals with the concept of a multinational design pre-licensing performed by regulators of several EU Member States. In this context, the report introduces and defines the concept of a Joint Overall Design Assessment (JODA) resulting in a Common Opinion on Design Acceptability (CODA). This is somewhat broader and at the same time more detailed than, but not contradictory to, the concept presented by the ENEF ERDA (European Reactor Design Acceptance) Subgroup in their 2012 Roadmap.

In accordance with the tender specifications, the focus of the report was put on scope and technical content of the pre-licensing assessment. In this context, the report distinguishes between two aspects:

1. The scope in the sense of broadness of the assessment, meaning the range of topics to be covered; and
2. The depth of the assessment, meaning the question to which extent of detail the assessment is taken for each of the topics within the scope.

Basically, the scope/broadness should be "full scope" and encompassing the entire range of safety analysis of a reactor design, as it is defined, in a more or less coherent fashion, by national and international assessment standards such as the IAEA's SSR-2/1, Safety of Nuclear Power Plants: Design, and MDEP's General Terms of Reference for design-specific working groups. This can be adjusted in details to fit the specific needs of the participating regulators and the vendor by entering into contractual arrangements. These arrangements could incorporate elements facilitating the project-independent pre-licensing assessment, such as envelope criteria or the possibility of alternative solutions.

Concerning the depth and technical content of the assessment, a level of detail such as that required for the issuing of a construction licence will probably not be achievable in the framework of a multinational cooperation and may not even be useful; on the other hand, the assessment must be "meaningful", i.e. substantial and technical enough to effectively front-load national licensing processes. Therefore, this Report proposes to implement a phased approach with an increasing level of detail modelled on existing pre-licensing design reviews such as the Canadian vendor design review. This approach follows three grades:

- Grade 1: assessment whether the general plant design, according to the claims of the designer, complies with the main safety goals in relevant legislation and regulations;
- Grade 2: assessment whether the claims can be demonstrated to be based on compliance with basic requirements as contained in national regulations;
- Grade 3: evaluation of the main design features against more detailed regulations and codes and standards.

For JODA, the technical content to be assessed for all topics should in substance be Grade 2. For particular issues defined in each JODA process, the assessment could also be taken to Grade 3. This proposed level of detail ensures meaningfulness of the assessment and enables, as far as possible, a smooth interface with the subsequent national licensing processes.

Thus, JODA can be summarised as featuring a "full scope but limited depth" assessment, subject to a certain degree of flexibility. For any particular assessment, this basic flexibility requires contractual agreements to freeze the criteria and to provide certainty to all parties involved.

The flexibility also applies to the reactor designs eligible for the process. Under the general guidance mentioned above (supplemented by SMR-specific aspects if necessary), the process would be open to all technologies.

Concerning the safety requirements utilised for design assessment, the situation in the EU seems favourable in that high-level requirements have been established in a uniform fashion by the revised Nuclear Safety Directive 2009/71, the WENRA reference levels and the common adherence to IAEA Safety

Standards; this is complemented by industry's EUR, as discussed in detail in the Task 1 Report. This level of harmonisation should greatly facilitate the joint assessment. Remaining differences in detailed requirements, and in general regulatory approach and "philosophy" as well as in assessment methodology need to be discussed in advance and settled in the contractual arrangements; a good model for reaching a common ground is the Franco-German cooperation of the 1990s where common safety goals and ETCs were defined. If necessary, it could also be considered that individual regulators, to an extent agreed beforehand, can utilise their own national requirements. They could still accept the design, based on "different grounds" by comparison to their peers; if necessary, they can express, in the CODA, some reservations on single points and define issues to be clarified in their subsequent national licensing process.

Finally, in terms of process, this Report advocates a realistic and pragmatic approach which does not necessitate a new legislative framework which would be difficult to implement. The process relies on the factual importance of the joint outcome (the CODA) which will, as practice has shown, be utilised and referenced by participating regulators unless new information arises casting doubt on the results. Likewise, no mandatory legal interface to national licensing processes is proposed. In this concept, the diversity of Member States' national frameworks can be taken into account as such and need not be addressed by changes in national legislation and regulation which would be difficult to impose.

All in all, the Joint Overall Design Assessment (JODA) would offer a voluntary and flexible but at the same time meaningful and effective tool for performance of a joint design assessment by the regulators of several EU Member States.

APPENDIX 1: US DESIGN CERTIFICATION: EXAMPLE FOR THE TABLE OF CONTENT OF A DESIGN CONTROL DOCUMENT (MAINLY TIER 2)

Source: U.S.NRC APR1400 Design Control Document and Environmental Report (nrc.gov/reactors/new-reactors /design-cert/apr1400/dcd.html#dcd) and APR1400 Design Control Document Tier 2 (nrc.gov/docs/ML1328/ ML13281A742.pdf)

Part	Chapter	Title
		Submittal Letter
Tier 1	1-3	**Tier 1 DCD** …………..
Tier 2	1	Introduction and General Description of the Plant
	2	Site Characteristics
	3	Design of Structures, Systems, Components, and Equipment
	4	Reactor
	5	Reactor Coolant and Connecting Systems
	6	Engineered Safety Features
	7	Instrumentation and Controls
	8	Electric Power
	9	Auxiliary Systems
	10	Steam and Power Conversion System
	11	Radioactive Waste Management
	12	Radiation Protection
	13	Conduct of Operations
	14	Verification Programs
	15	Transient and Accident Analyses
	16	Technical Specifications
	17	Quality Assurance and Reliability Assurance
	18	Human Factors Engineering
	19	Probabilistic Risk Assessment and Severe Accident Evaluation

APPENDIX 2: FRANCE, REVIEW OF SAFETY OPTIONS: SCOPE OF THE ATMEA REVIEW DEFINED BY THE "SAFETY OPTIONS DOSSIER" SUBMITTED BY ATMEA

Source: ASN, ATMEA1 reactor, Review of safety options, CODEP-DCN-2011-070548 Report of 24 January 2012, p. 12 (http://www.french-nuclear-safety.fr/ content/download/54088/368528/version/1/file/2012-01%20ASN %20report%20 CODEP-DCN-2011-0700548%20Atmea%20safety%20option%20review% 20synthesis.pdf)

- Report on Safety options (safety design basis) for the ATMEA1 reactor
- Report on Compliance with the Technical Guidelines for the design and construction of the new generation of pressurised water nuclear reactors
- Technical reports for each of the following topics:
  - Complementary Requirements to the ASME code for ATMEA1 Basic Design
  - Specific systems
    - Safety injection system
    - Residual Heat Removal System and Containment Spray System
    - Severe Accident Heat Removal System
    - Electrical design
    - Overpressure protection
    - Containment Design
    - Confinement Functions
    - I&C General architecture
    - Reactor Pressure Vessel Internals
    - Steam Generators
    - Reactor Coolant Pumps
    - Control rod drive mechanisms
    - Reactor pressure vessel
    - Reactor coolant line
    - Main steam line and feedwater line
    - Pressurizer
  - Hazards, initiating events
    - Protection against internal hazards
    - External hazards
    - Large Commercial Airplane crash
    - Post Fukushima short-term experience feedback
  - Safety assessments
    - PSA Level 1

APPENDIX 3: CANADA: VENDOR DESIGN REVIEW (VDR): FOCUS AREAS REVIEWED DURING PHASES 1 AND 2 OF A DESIGN REVIEW

Source: CNSC, Pre-licensing Review of a Vendor's Reactor Design, GD-385, May 2012, p. 6 (http://nuclearsafety. gc.ca/eng/acts-and-regulations/regulatory-documents/ published/html/gd385/index.cfm)

1. general plant description, defence in depth, safety goals and objectives, dose acceptance criteria
2. classification of structures systems, and components (SSCs)
3. reactor core nuclear design
4. fuel design and qualification
5. control system and facilities:
    a. main control systems
    b. instrumentation and control
    c. control facilities
    d. emergency power system(s)
6. means of reactor shutdown
7. emergency core cooling and emergency heat removal systems
8. containment / confinement and safety-important civil structures
9. beyond design basis accidents (BDBAs) and severe accidents (SA) prevention and mitigation
10. safety analysis (deterministic safety analysis, probabilistic safety analysis) and internal and external hazards
11. pressure boundary design
12. fire protection
13. radiation protection
14. out-of-core criticality
15. robustness, safeguards and security
16. vendor research and development program
17. management system of design process and quality assurance in design and safety analysis
18. human factors
19. incorporation of decommissioning in design considerations.

APPENDIX 4: MULTINATIONAL DESIGN EVALUATION PROGRAMME (MDEP): MINIMUM SET OF ISSUES TO BE CONSIDERED BY A DESIGN-SPECIFIC WORKING GROUP

Source: Multinational Design Evaluation Program (MDEP), Design-specific working groups (DSWGs) General Terms of Reference, Revision 7a, May 2016, p. 3 (https://www.oecd-nea.org/mdep/documents/dswgs-tor.pdf)

- Fukushima Daiichi NPP accident lessons learnt;
- Severe accidents;
- Probabilistic safety assessment;
- Technical specifications;
- Digital instrumentation and control / electrical distribution;
- Design basis accidents or equivalent and transient response;
- Fire protection;
- Human factors engineering issues (control room design);
- Radiation protection;
- Radiological waste management;
- Unique design features affecting safety;
- Accident categorisation / event categorisation;
- Treatment of external / internal events;
- Civil and structural engineering of critical structures;
- Adequacy of heat sinks;
- Shutdown safety;
- Balance of plant systems;
- Commissioning tests (validation of design);
- Early operation phase.

APPENDIX 5: IAEA: LIST OF SAFETY ASSESSMENT TOPICS EXTRACTED FROM: IAEA SSR-2/1 (REV. 1) SAFETY OF NUCLEAR POWER PLANTS: DESIGN, 2016

PRINCIPAL TECHNICAL REQUIREMENTS

Fundamental safety functions
Radiation protection in design
Design for a nuclear power plant
Application of defence in depth
Interfaces of safety with security and safeguards
Proven engineering practices
Safety assessment
Provision for construction
Features to facilitate radioactive waste management and decommissioning

GENERAL PLANT DESIGN

Design Basis
Design for safe operation over the lifetime of the plant.
Human factors
Other design considerations
Safety analysis (deterministic and probabilistic)

DESIGN OF SPECIFIC PLANT SYSTEMS

Reactor core and associated features
Reactor coolant systems
Containment structure and containment system
Instrumentation and control systems
Emergency power supply
Supporting systems and auxiliary systems
Other power conversion systems
Treatment of radioactive effluents and radioactive waste
Fuel handling and storage systems
Radiation protection

TASK 4 - Benchmarking of the national long term operation programmes

TENDER SPECIFICATIONS OF TASK 4

EU tender ENRE/D2/2016-677 specifies the requirements for Task 4 as follows:

> “Benchmarking and peer reviews performed by national safety authorities or IAEA expert missions are essential tool to achieve a harmonized level of safety in the EU and worldwide. Those tools are of particular interest for the numerous expected LTO in the EU. At the time being related efforts developed by regulators or the IAEA missions (e.g. SALTO) focus in priority on aging effects, less on “reasonably practicable” safety upgrades that accompany the LTO programmes. Hence, a common set of technical references integrating both aging and safety upgrades would help all involved actors to deliver programmes across the different Member States with equivalent safety levels.
>
> Deliverables: In close cooperation with nuclear safety authorities, a set of technical reference guide(s) for LTO will be prepared, covering in particular aspects related to the safety upgrades. In addition, a benchmarking of national LTO programmes against the amended directives will be defined and described (technical content of their Terms of References), particularly regarding art. 8a and 8b of amended Nuclear Safety Directive, for future use by regulators.”
>
> At the project Kick-off meeting of 8 September 2017 it was agreed that the aim of this task is to propose the approach/methodology for the discussion with the regulators, to be documented in a report. The aim is not to develop the “Technical reference guide” as specified in the TOR for the project.

1 BENCHMARKING OF THE NATIONAL LONG TERM OPERATION PROGRAMMES

1.1 BACKGROUND

The convergence of nuclear safety levels within the European Union (EU) received the initial impetus with the publication of WENRA reference levels (RLs) [4.1], which all member states (MSs) were required to include in their regulatory regime – as a part of Harmonization of Reactor Safety in WENRA Countries programme that was to be completed by 2010. The 2011 Fukushima accident followed by the EU post-Fukushima stress test and implementation of national action plans (NAcP) resulted in numerous safety improvements across the fleet of NPPs in the EU Member States (MSs). Subsequently, WENRA updated the reference levels (RLs) to account for lessons learned from Fukushima accident [Ref. 4.2]. Articles 8a and 8b of the amended Nuclear Safety Directive 2014/87 EURATOM (App. 1) clearly established the requirement that the high level of safety is achieved in all MSs aimed at practical exclusion of releases that would have long term environmental effects.

However, during the Stress test and NAcP process, there was no comparison or benchmarking undertaken, so we still do not know what safety levels have been achieved at EU NPPs. Although the NSD clearly established the requirement that the high level of safety is achieved in all MSs aimed at practical exclusion of releases that would have long term environmental effects, the level of fulfilment of the Articles 8a and 8b of the amended NSD by EU NPPs is not known.

Ensuring high level of safety of nuclear plants LTO requires systematic and comprehensive safety reviews to cover all aspects important to the safety of NPP and address, in an appropriate manner, the consequences of the cumulative effects of plant ageing and plant modifications, equipment requalification, operating experience, current standards, technical developments, organizational and management issues, as well as siting and, depending of the nature of the project, environmental aspects. This can be achieved through the following two commonly used methods/practices:

a. Performing a periodic safety review (PSR) in accordance with the IAEA SG [4.3] or an equivalent comprehensive safety assessment and implementing appropriate safety improvements to deal with identified safety issues; and thus to ensure throughout LTO period the existence of effective defence-in-depth (DinD) and the availability of the fundamental safety functions of Control-Cool-Contain that are essential for practical exclusion of major releases. (PSR is a forward-looking instrument for maintaining the safety of NPP operation.)

b. Implementing systematic ageing management programme to deal with material degradation of SSCs in order to ensure that the effects of ageing will not prevent SSCs from performing their safety functions throughout the NPP service life.

Benchmarking and peer reviews performed by national safety authorities or IAEA expert missions are essential tools to achieve a harmonised level of safety in the EU and worldwide. Those tools are of particular interest for the numerous existing and expected LTO programmes in the EU. Present LTO relevant review activities by regulators and the IAEA (e.g. SALTO) focus on managing material ageing effects, less on 'reasonably practicable' safety upgrades that accompany the LTO.

The aim of Task 4 is to propose an approach where the safety improvements/measures, regardless whether they originate in the PSR, the LTO, the stress test or national regulator or operator initiative, would be benchmarked– . The aim for EU NPPs to reach equivalent high level of nuclear safety consistent with the NSD is an underlying objective to be dealt with by EU member states after completion of the LTO benchmark (while using results of the LTO benchmark).

Benchmarking compares a status of e.g. a facility with a particular standard and shows where it is in relation to that standard. In the Task 4, the aim was to conceptualise the approach were EU NPPs are to be benchmarked against the Articles 8a and 8b of the amended NSD 2014/87. Since these Articles establish high level requirements (mainly related to what needs to be done) rather than defining the safety level to be achieved, practical benchmarking criteria (LTOC) needs to be developed that would facilitate benchmarking against the Article 8b of the amended NSD 2014/87.

The benchmarking LTO programmes needs to look at the plant in totality - internal and external events, human factors, ageing etc. PSR performed according to IAEA SG looks at the plant in totality, including the areas listed in Art. 8b. *(However, the EU MSs have not been closely following this SG, and the codes and standards against which PSR is done have not been uniform. As a result, outcomes of such PSRs are not comparable and therefore at present not usable in the desired benchmarking of LTO programmes.)* Relevant attributes of each NPP (including the original design and all subsequent safety improvements/measures, regardless whether they originate in a PSR, LTO upgrades, the stress test or national regulator or operator initiatives) are to be compared with LTOC, providing three possible outcomes: LTOC fully met/ LTOC partly met/LTOC not met.

1.2 OBJECTIVE OF TASK 4

The objective of Task 4 is to propose an approach/concept for benchmarking/assessing the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87. This approach is to take into account all plant safety improvements/measures that have been implemented, including those in response to a PSR, a programme for LTO, the Stress test or that resulted from a national regulator or operator initiative [4.4].

Note. Benchmarking provides a snapshot of an organizational performance and shows where it is in relation to a particular standard – for Task 4, Articles 8a and 8b of the amended NSD 2014/87.

1.3 SCOPE OF TASK 4

Task 4 will develop a proposal of the approach/concept for benchmarking/assessing the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87 which will include:

- The scope (areas of review) of the LTO programme benchmarking

  Note. There is a need to look at the plant in totality - internal and external events, human factors, ageing etc. PSR performed according to IAEA SG looks at the plant in totality, including the areas listed in Art. 8b.[2]

[2] WENRA RLs for existing nuclear power plants cover the topic of PSR in Issue P. Among other things, PSR shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available. P 1.4: All reasonably practicable improvement measures shall be implemented by the licensee as a result of the review, in a timely manner.

- An explanation of benchmarking criteria (LTOC) to be used and, for illustration, examples of LTOC for selected areas of review

  Note. Article 8b of the Directive 87/2014 establishes some high level requirements (mainly related to what needs to be done) rather than defining the safety level to be achieved. Therefore, practical benchmarking criteria (LTOC) needs to be developed that would facilitate benchmarking against the Article 8b of the amended NSD 2014/87.

- An outline of the benchmarking method
- A limited-scope example of application of the proposed LTO benchmarking method

  Note. The aim of Task 4 is not to develop the "Technical reference guide" as specified in the TOR for the project [4.4].

1.4 STRUCTURE OF THE TASK 4 REPORT

Section 2 outlines the scope (areas of review) of the benchmarking of NPP LTO programs against the Articles 8a and 8b of the amended NSD 2014-87.

Section 3 presents examples of benchmarking criteria (LTOC) in review areas of Ageing, and External hazards (sub-area of PSR safety factor Hazard analysis).

Section 4 outlines the benchmarking approach/method (intended for self-evaluation by NPPs and presentation in country reports that would be subjected to a peer review).

Section 5 provides an example of application of the proposed LTO benchmarking method to two NPPs using relevant available data (from the EU Fukushima Stress Test and 2017 WENRA Topical Peer Review on Ageing Management) in review areas of Ageing, and External hazards (sub-area of PSR safety factor Hazard analysis).

Section 6, Conclusions, suggests steps of further development and identifies foreseen benefits of implementation of the proposed LTO benchmarking method.

2 SCOPE OF BENCHMARKING OF LTO PROGRAMMES

This section outlines the scope (areas of review) of the benchmarking of NPP LTO programs against the Articles 8a and 8b of the amended NSD 2014-87 (App. 1).

In order to benchmark the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014-87, the benchmark method is to take into account all plant safety improvement that have been implemented, including those in response to a PSR, a programme for LTO, the Stress test or that resulted from a national regulator requirement or operator initiative. To accomplish that, there is a need to look at a plant in totality – design, internal and external events, human factors, ageing, etc. as in a PSR performed according to IAEA Safety Guide on PSR [4.3].

In line with the IAEA Safety Guide on PSR, WENRA Reference Levels (RLs) [4.1] for existing nuclear power plants cover the topic of PSR in Issue P. Among other things, PSR shall identify and evaluate the safety significance of deviations from applicable current safety standards and internationally recognised good practices currently available. And, P 1.4 states: All reasonably practicable improvement measures shall be implemented by the licensee as a result of the review, in a timely manner. PSR includes determination of appropriate safety improvements to deal with identified shortfalls in safety standards; however, as noted above, LTO benchmarking is limited to identifying those shortfalls. Considering the foregoing, the proposed scope (areas of review) for benchmarking/assessing the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014-87 consists of the following 14 PSR Safety Factors grouped in five subject areas [4.3] that are intended to cover all aspects important to safety of an operating NPP.

Plant

1. Plant design
2. Actual condition of systems, structures and components
3. Equipment qualification
4. Ageing

Safety analysis

5. Deterministic safety analysis
6. Probabilistic safety assessment
7. Hazard analysis

Performance and feedback of operating experience

8. Safety performance
9. Use of experience from other plants and of research findings

Management

10. Organization and administration
11. Procedures
12. Human factors
13. Emergency planning

Environment

14. Radiological impact on the environment

The proposed benchmarking approach/concept is based on the benchmarking method used in the 1st Ageing Benchmark of Borssele Benchmark Committee [4.5]. In such an approach, relevant attributes of each NPP would be compared with LTO benchmarking criteria (LTOC) providing three possible outcomes: LTOC fully met/ LTOC partly met/LTOC not met.

Recognizing that the Articles 8a and 8b of the amended NSD 2014-87 establish high level requirements (mainly related to what needs to be done) rather than defining the safety level to be achieved, practical benchmarking criteria – LTOC would be developed for each of the above PSR safety factors aimed to facilitate self-evaluation by NPPs and presentation in country reports that would be subjected to a peer review.

To get an overall benchmark assessment of the plant against the above LTOC, a PSR-like global assessment should be also included in the LTO benchmark which would summarize the outcomes of individual safety factor reviews and their interaction, all safety improvements implemented (which may relate both to the plant and to the operating organization), all remaining deficiencies and, taking this into account, to assess the overall level of plant safety.

NOTE. *(This note is using quotations from or paraphrases statements of the IAEA Safety Guide on PSR, Ref. 4.3.)* In a PSR, each safety factor is reviewed using current methods and the findings are assessed against current safety standards and practices. Paragraph 6.11 of Ref. [4.3] states "the global assessment should review the extent to which safety requirements relating to the concept of defence in depth and the fundamental safety functions (reactivity control, core cooling and the confinement of radioactive material) are fulfilled. The adequacy of the plant's defence in depth may be demonstrated by reference to the five levels defined in Ref. [4.6]." Reasonable and practicable corrective actions and/or safety improvements are determined and an implementation plan is agreed *(as part of the PSR)*, with account taken of the interactions and overlaps between safety factors. The global assessment should take into account all the positive and negative findings from the PSR, and the corrective actions and/or safety improvements proposed, and should assess the overall level of safety that will be achieved at the nuclear power plant following the PSR.

As noted above, PSR includes determination of appropriate safety improvements to deal with identified shortfalls in safety standards; however LTO benchmarking is limited to identifying those shortfalls. Currently, decisions on specific safety upgrades to address any shortfalls identified by any of the licensee's detailed assessments, the results of a PSR, or an LTO benchmark, are made in agreement with the national regulators. The 'Reasonably practicable' safety upgrades are plant-specific, proposed by a licensee and, case by case, agreed with the regulatory authority taking into account a given specific safety issue. An assessment of the adequacy of any proposed safety improvements needs to consider the complete representation of nuclear safety, i.e. accounting for the findings from all 14 PSR safety factor reviews , as this allows undertaking a 'global risk' judgement on continued plant operation (i.e. throughout the LTO) with any deficiencies remaining after implementation of all safety improvements. This is important because specific safety challenges might be compensated by different measures, those being hardware, procedural or operational improvement (or limitations), commonly leading to "reasonably practically" safety improvements. In particular, the assessment should ensure the existence of effective defence-in-depth of the availability of fundamental safety functions of Control-Cool-Contain that are essential for practical exclusion of major releases. (For illustration of 'reasonably practicable' safety improvements, two examples are included at the end of this section.)

Nevertheless, a benchmarking of the safety level achieved or proposed to be achieved at NPPs in the EU, in terms of fulfilling the requirements of the Article 8 as of the NSD need to consider the totality of the approach to nuclear safety that originates in the actual plant design and siting and then considers the upgrades/modifications implemented or proposed. The individual upgrades cannot be compared across the fleet of reactors because the design (or siting) vulnerabilities are different; also, specifics of safety improvement concepts depend on the variety of other issues, including historical developments, technical feasibility of a solution, available resources, etc. Consequently, what might be "reasonably practicable" at a certain NPP, might not all be possible at another. Even more what might be reasonably practicable with one design, might be either too much or too little in another.

One could use several examples to illustrate the challenge when trying to benchmark or even comment on "reasonable practicable" safety improvements. One is the filtered vent modification for containment.

While many plants could have a sequence where the pressure in the containment, especially in the long term, would go beyond the structural integrity of the containment, there are designs where, due to the strength of the containment or its capacity, this is not the case. Also for the plants where, e.g. the RPV failure could be "practicality eliminated", filtered vent is generally of less importance. Another example are various modifications to protect the base plate of the containment from eventual melt through, e.g. wet cavity arrangements. For the plants where the in-vessel retention is possible and prepared for though hardware modifications or operation/procedurals arrangements, measures related with base plate are not needed.

This illustrates a necessity of holistic consideration when benchmarking the safety level achieved across the fleet, where e.g. the findings across the 14 safety factors within PSR could be compared rather than individual safety related modifications. In this way, an understanding of the upgrades selected, i.e. how do they deal with actual (or perceived) weaknesses of the design, would be readily available. This would allow for an assessment of safety level reached even in cases where a very different individual safety upgrades were adopted.

In this connection, a formal guidance on reasonably practicable safety improvements would be useful for PSR implementation (as well as for a follow up to outcomes of the LTO benchmark) and for achieving a harmonized nuclear safety level across EU, Ref. [4.7] by WENRA provides relevant guidance.

Implementation of the benchmarking of LTO programmes would be fairly straightforward, should the PSRs in EU MS be performed using the same process (articulated by the IAEA Safety Guide on PSR) and a uniform reference/benchmark level constituted by applicable safety requirements, codes, standards and practices. Outcomes of such PSRs would be comparable and directly usable in the desired benchmarking of LTO programmes.

Although WENRA RLs Issue P for PSR have resulted in convergence of PSR practices of EU member states, there are still differences that do not permit use of outcomes of current PSRs of different MS in the benchmarking of LTO programmes. It is envisaged that the benchmarking the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87 would encourage EU member states to harmonize PSR implementation in line with the IAEA PSR Safety Guide and thus with the WENRA Reference Level Issue P on PSR. And, this would go a long way towards reaching an equivalent level of nuclear safety across EU NPPs.

Examples of application of 'reasonably practicable' safety improvements

UK's Long Term Safety Reviews (LTSRs) that began in the late 1970s provided the basis for judging the adequacy of the safety case for operation of the Magnox reactors beyond their initial investment life of 20 -25 years (LTSRs were PSR predecessors). Early LTSRs identified a number of significant safety issues. One of them was a Post trip cooling issue that required a reliable supply of feed water to boilers – as far as reasonable, meeting modern standards. Taking into account that Magnox reactors are slow to respond to faults and can withstand losing water supply for up to 20 hours without exceeding safety limits, the licensee proposed installing diverse and segregated pumps, with multiple new feed inlet points and connections being made by flexible hose after the fault has occurred.

The regulator, NII wanted a fully engineered system, but was prepared to listen to the licensee's proposals. NII accepted the licensee's proposals subject to a successful demonstration which took place at night with lighting extinguished and staff wearing breathing apparatus in areas defined by NII as unacceptable. The demonstration was satisfactorily completed within 52 minutes, demonstrating a large margin.

A flexible approach was adapted to the resolution of Magnox safety issues, taking account the features of the reactors. Experience with Magnox LTSRs showed that systematic and comprehensive safety reviews and a flexible approach for dealing with safety issues can improve plant safety and thus help prevent serious plant accidents.

In contrast, 2011 Fukushima accident revealed that the PSRs Japanese NPP operators were to perform were not sufficiently comprehensive and rigorous – they did not follow recommendations of the IAEA Safety Guide on PSR. As a result, a need for safety improvements had not been identified and as a result, *before the 2011 accident, Fukushima's General Electric Mark I BWRs were operating without adequate protection against external hazards (tsunami) and without an effective containment system.*

3 EXAMPLES OF LTO BENCHMARKING CRITERIA

Article 8b of the Directive 87-2014 establishes some high level requirements (mainly related to what needs to be done) rather than defining the safety level to be achieved. Therefore, practical benchmarking criteria (LTOC) need to be developed that would facilitate benchmarking against the Articles 8a and 8b of the amended NSD 2014-87. In line with the IAEA Safety Guide on PSR and WENRA Reference Levels, these criteria will be derived from applicable current safety standards and internationally recognised good practices.

According to section 4.2 of this report, "… the proposed benchmarking approach/concept is based on the benchmarking method used in the 1st Ageing Benchmark of Borssele Benchmark Committee [4.6]. In such an approach, relevant attributes of each NPP would be compared with LTO benchmarking criteria (LTOC) providing three possible outcomes: LTOC fully met/ LTOC partly met/LTOC not met."

This section presents examples of LTOC for two benchmarking areas of review, namely Ageing, and External hazards (sub-area of PSR safety factor Hazard analysis), including their brief explanation.

3.1 AGEING MANAGEMENT CRITERIA

The ageing review focuses on the question to what extent ageing management is a well-managed process. The review consists of an evaluation of ageing management governance and of ageing management implementation.

Ageing Management Criteria (AMC) are based primarily on IAEA Safety Guide on Ageing Management [4.8] and the work of the 1st Borssele Benchmark Committee [4.5]. This Safety Guide provides recommendations for managing ageing of systems, structures and components important to safety, including recommendations on key elements of effective ageing management and an outline of a review of ageing management for long term operation. The Guide contains generic attributes, which should be part of every component-specific Ageing Management Programme (AMP) (Table 2 of Ref. 4.8).

In addition, the AMC are compatible with IAEA Safety Reports Series No. 57, Safe Long Term Operation of Nuclear Power Plants [4.8], IAEA Safety Guide on PSR [4.3], WENRA Reference Levels on Ageing [4.2], and US NRC Generic Aging Lessons Learned [4.10].

The criteria developed are on one hand aimed at evaluating the ageing management governance (i.e. documentation of a plants policy, organization and methodology for ageing management) and on the other hand at evaluating the way ageing management is implemented in practice for a selected number of relevant systems, structures and components. They are presented in the following tables in Appendix 4.2:

Table 4.1 provides AMC for a high level evaluation of governance documents of the overall plant AMP, i.e. documentation of NPP policy, organization and methodology for ageing management that should provide direction for effective ageing management.

Table 4.2 presents AMC for results-oriented evaluation of SSC-specific AMPs, i.e. the evaluation of the extent to which the SSC-specific AMPs (of a representative sample of major SSCs important to safety) have the generic attributes of an effective AMP.

Table 4.3 presents AMC for results-oriented evaluation of the AMP scope for LTO (i.e. SSCs covered by the overall plant AMP).

Table 4.4 presents AMC for the evaluation of TLAAs for LTO.

NOTE: Topics to be reviewed are in the first column; the second column presents associated assessment bases/criteria to be used in the review.

Examples of AMC for a representative sample of major SSCs important to safety and of TLAAs are in Tables 4.5 – 4.10. These AMC have been derived from US NRC GALL report [4.9] and relevant IAEA Tecdocs.

Tables 4.5 – 4.7 present AMC for results-oriented evaluation of three SSC-specific AMPs: AMPs for FAC of high energy carbon steel piping and valve bodies, Buried piping and tanks, and Insulation materials of electrical cables

Tables 4.8 – 4.10 present AMC for the evaluation of three specific TLAAs: TLAAs for Reactor vessel neutron embrittlement, Metal fatigue of class 1 piping, and Environmental qualification of electrical equipment.

3.2 EXTERNAL HAZARDS CRITERIA

Hazards analysis (internal and external) is safety factor #7 in the list of safety factors of PSR (see Section 4.2 above and Ref. 4.4). According to para 5.74 of this safety guide:

"The objective of the review of hazard analysis is to determine the adequacy of protection of the nuclear power plant against internal and external hazards, with account taken of the plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of the period covered by the PSR, and current analytical methods, safety standards and knowledge."

Furthermore, the para 5.75 of the safety guide states that:

"For each internal or external hazard identified, the review should evaluate the adequacy of the protection, with account taken of the following:

- The credible magnitude and associated frequency of occurrence of the hazard;
- Current safety standards;
- Current understanding of environmental effects;
- The capability of the plant to withstand the hazard as claimed in the safety case, based on its current condition and with allowance given to predicted ageing degradation;
- The appropriateness of procedures to cover operator actions claimed to prevent or mitigate the hazard".

The list of hazards analysis includes:

- Internal hazards: fire (including measures for prevention, detection and suppression of fire), flooding, pipe whip, missiles and drops of heavy loads, steam release, hot (cold) gas release, deluge and spray, explosion, electromagnetic or radio frequency interference, toxic and/or corrosive liquids and gases, vibration, subsidence, high humidity, structural collapse, loss of internal and external services (cooling water, electricity, etc.), high voltage transients, loss or low capacity of air conditioning (which may lead to high temperatures).
- External hazards: floods, including tsunamis, high winds, including tornadoes, fire, meteorological hazards (extreme temperatures, extreme weather conditions, high humidity, drought, snow, build-up of ice), sun storm, toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants, volcanic ash), hydrogeological and hydrological hazards (extreme groundwater levels, seiches, seismic hazards, volcano hazards, aircraft crashes, external missiles, explosion, biological fouling, lightning strike, electromagnetic or radio frequency interference, vibration, traffic, loss of internal and external services (cooling water, electricity, etc.).

IAEA recommends that the review of the external hazard analysis should constitute a well-managed process. As part of this process, "… a list of relevant … external hazards that may affect plant safety should be established. Where a list of relevant internal and externals hazards has already been established, this should be reviewed for completeness" [para 5.76 of IAEA SSG-25].

The external hazards criteria (EHC) are aimed at facilitating review of the 'adequacy' of protection of an NPP against relevant external hazards. The general EHC are presented in Appendix 4.2, Tables 4.11 – 4.14; the tables 4.12 – 4.14 are templates that are to be used when developing EH specific criteria.

Table 4.11 provides EHC for the *overall management of external hazards*.

Table 4.12 provides the general review criteria for *site characteristics and EH analysis* (design vs. current) for an external hazard. EH-specific criteria should be developed for each external hazard taken into consideration for LTOC.

Table 4.13 provides the general criteria for reviewing the adequacy of current *EH safety analysis and its results* indicating plant robustness against the current EH. Such criteria should be developed for each external hazard taken into consideration for LTOC.

Table 4.14 provides the general EHC for reviewing implementation of *follow-up actions (physical and procedural improvements)* resulting from current deterministic/ probabilistic safety analysis for current external hazard. Such EHC should be developed for each external hazard taken into consideration for LTOC.

Hazard-specific EHC for Seismic Hazard and Flooding Hazard that will be used in an example of application of the proposed LTO benchmarking method are in Tables 4.15 – 4.18 of Appendix 4.2.

Note 1. EHC are based on recommendations of relevant IAEA safety standards and safety guides included in References 4.11 – 4.20.

Note 2. Each NPP is to be reviewed against the same EHC to determine the extent to which the EHC are met.

Note 3. Topics to be reviewed are in the first column; the second column presents associated assessment bases/criteria to be used in the review.

4 APPROACH FOR BENCHMARKING OF LTO PROGRAMMES

This section outlines the benchmarking approach/concept - intended for self-evaluation by NPPs and presentation in country reports that would be subjected to a peer review.

The proposed LTO benchmarking approach/concept is based on the benchmarking method used in the 1st Ageing Benchmark of Borssele Benchmark Committee (BBC) [4.5]. In such an approach, relevant attributes of each NPP are compared with LTO benchmarking criteria (LTOC) providing three possible outcomes: LTOC fully met/ LTOC partly met/LTOC not met.

In the 1st BBC Ageing Benchmark, plant-specific attributes of an NPP were compared to the Ageing Management Criteria (AMC) presented in Section 3. The scope of the LTO Benchmark project is much wider, consisting of 14 PSR Safety Factors (Section 4.2) each of which will have an associated Safety Factor specific LTOC like the Ageing Management Criteria (AMC) and External Hazards Criteria (EHC) that are presented for illustration in Section 3.

Plant-specific characteristics/attributes of an NPP will be evaluated against the 14 Safety Factor specific LTOC and findings documented in Safety Factor specific review matrices that will identify safety significant shortfalls/deficiencies which may relate both to the plant and to the operating organization.

To get an <u>overall LTO Benchmark assessment of a plant</u> against the above LTOC, a PSR-like global assessment should be included in the LTO benchmark which would summarize the outcomes of individual safety factor reviews and their interaction - indicating all significant shortfalls/deficiencies, safety improvements planned and the overall level of plant safety with respect to the Article 8b of the amended NSD 2014-87.

The benchmarking approach is depicted in Figs. 10 and 11, and outlined below. Associated tables 4.19 – 4.23 are in Appendix 3.

Note. Preparation of an NPP Global LTO Benchmark assessment report for each NPP reviewed (step 5 of the proposed benchmarking approach) may prove to be difficult; a pragmatic concept will need to be developed based on experience in a pilot project. Another challenge will be in comparing/integrating individual NPP Global LTO Benchmark assessment reports to obtain an overall EU picture on safety levels of EU NPPs.

OUTLINE OF THE BENCHMARKING APPROACH

1. <u>Develop LTOC for each Safety Factor</u> based on relevant IAEA standards and guidelines, i.e. 14 sets of Safety Factor specific LTOC.

   1.1 Subdivide each Safety Factors into Review areas, as appropriate.

   1.2 For each Review area, identify Review topics, as appropriate.

   1.3 For each Review topic, identify Review topic specific LTOC criteria.

   For illustration, see Fig. 11 showing a subdivision of the Ageing Safety Factor into 8 Review areas; a subdivision of each Review area into relevant Review topics with associated Ageing Management Criteria (AMC) is in Tables 4.1 – 4.8 of App. 2.

The following steps are to be applied to a specific NPP to be benchmarked.

2. For each Safety Factor, perform review against the Safety Factor specific LTOC that were developed in Step 1, i.e. evaluation of the extent to which the plant complies with the LTOC; and document them in Review area specific review matrices; see Table 4.19 (in App. 3) for a Safety Factor-specific Generic Review Matrix to be used for each Review area.

   In each Review area specific review matrix, indicate the adequacy of implementation (LTOC fully met, LTOC partly met or LTOC not met), and in the Remarks column the basis for a particular rating as well as a reference to a document on which the rating is based. See Table 4.20 (in App. 3) for an example of the SF-specific Generic Review Matrix application to one Review area of the Ageing Safety Factor.

3. For each Safety Factor, prepare a SF- specific Report Card that gives a summary of results of the Safety Factor evaluation against LTOC, identifying significant shortfalls in compliance to relevant LTOC and NSD 2014-87 Art. 8b requirements. See Table 4.21, App. 3 for a Generic Report Card and Table 4.22. App .3 for an example of its application.

4. Integrate 14 SF-specific Report Cards into a Global LTO Benchmark assessment matrix (Table 4.23, App. 3) of a specific NPP indicating the overall level of plant safety with respect to the Article 8b of the amended NSD 2014-87. The Global assessment matrix summarizes NPP’s compliance with the LTOC criteria as documented in the 14 SF-specific Report Cards.

5. *Prepare an NPP Global LTO Benchmark assessment report* for each NPP reviewed. This report should consist of (a) the Global assessment matrix summarizing NPP’s compliance with the LTOC criteria prepared in step 4, and (b) a narrative assessment indicating in simple qualitative terms the overall level of plant safety with respect to the Article 8b of the amended NSD 2014-87. The narrative Global LTO Benchmark assessment should provide an overall impression on the overall plant safety based with respect to the Article 8b of the amended NSD 2014-87 based on the NPP compliance with the LTOC criteria in the 14 SF review areas.

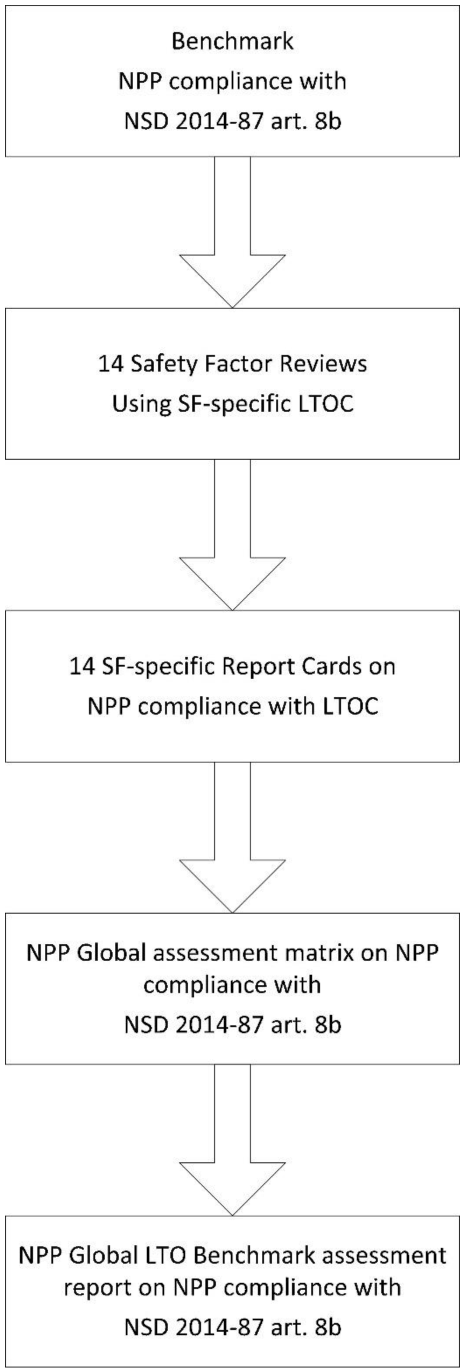


Figure 10: Approach for benchmarking of LTO programmes

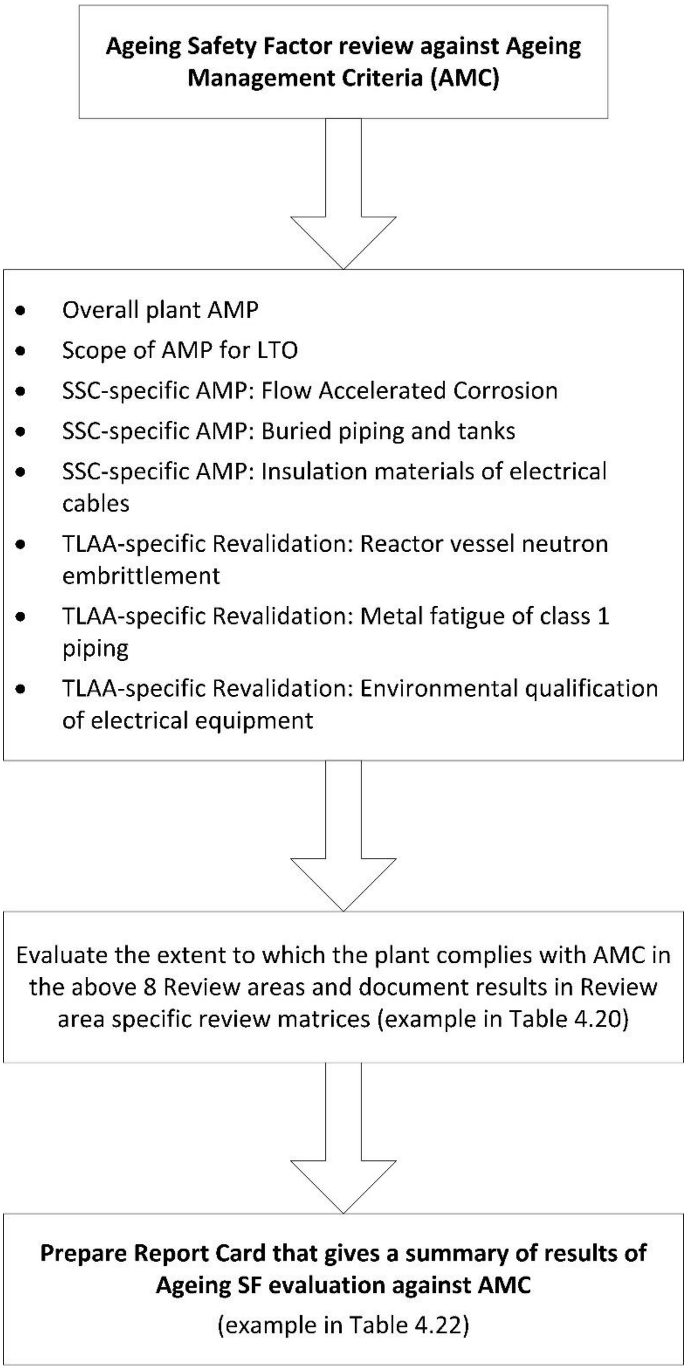


Figure 11: Benchmarking review against SF-specific LTOC: Ageing Safety Factor example

5 EXAMPLES OF APPLICATION OF THE PROPOSED LTO BENCHMARKING METHOD

This section presents examples of application of the proposed LTO benchmarking method to two NPPs using relevant available data (from the 2017 WENRA Topical Peer Review on Ageing Management and EU Fukushima Stress Test) in review areas of Ageing, and External hazards (sub-area of PSR safety factor Hazard analysis).

The relevant data were extracted from the respective 2017 WENRA Topical Peer Review on Ageing Management and national reports on the EU Fukushima Stress Test , and were evaluated against the relevant LTO Benchmark criteria (LTOC), i.e. the Ageing Management Criteria (AMC) and the External Hazard Criteria (EHC) that are presented in Section 4.3 and associated Appendix 4.2 and Appendix 4.3. Extracting the data relevant for the example application of the proposed LTO benchmarking method from the above national reports required a 'detective' work because the structure and topics AMC and EHC do not match the prescribed structure and topics of the national reports. Since some statements made in the reports are not backed-up by proper documentation, the consultant had to make a judgement call about their accuracy. In such cases, the consultant had to make additional investigation looking for clarification in internet documents publicly available, a task which is tedious and not always successful. However, it should be noted that in the proposed LTO benchmarking approach, the NPP self-evaluation of the extent to which the plant complies with the LTOC should be straightforward and relatively easy for NPP or TSO subject matter experts in the 14 areas of review that are identical with 14 PSR Safety Factors.

Results of the sample application of the proposed LTO benchmarking method in the review areas of Ageing and External hazards presented in appendices 4.4 and 4.5. They indicate the following compliance with AMC and EHC.

For NPP X:

- Overall, the AMC for NPP X Ageing Management are 'Fully met', taking into account that Scope/topics of the Topical Peer Review do not match the AMC scope/review areas for Safety Factor Ageing.
- Overall, the EHC are 'Partly met'. The amount of information in the national report on EU Post-Fukushima Stress Test is not entirely adequate for checking compliance with all EHC.

For NPP Y:

- Overall, the AMC for NPP Y Ageing Management are 'Fully met', taking into account that Scope/topics of the Topical Peer Review do not match the AMC scope/review areas for Safety Factor Ageing.
- Overall, the EHC for NPP Y External Hazards are 'Fully met' based on the evaluation of information provided in the national report on the EU Post-Fukushima Stress Test.

In the consultant's opinion, the sample application of the proposed LTO benchmarking method to two EU NPPs showed a viability of this method and is ready for presentation to and interaction with EU regulators aimed at the method's further development.

6 CONCLUSIONS

The scope of the proposed benchmarking approach consists of the 14 PSR Safety Factors that cover all aspects important to safety of an operating NPP – providing for an assessment of the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014-87. Recognizing that different safety upgrades (hardware or procedural/operational might be selected by different plants) can lead to comparable safety level, the outcome of the proposed benchmarking approach is independent of specific details of individual upgrades proposed. The benchmarking approach does not support a direct comparison of individual “reasonably practicable” modifications, each of those being seen in a broader nuclear safety concept defined by the NSD. However, the approach provides for taking into account different safety improvement alternatives and for understanding as to why these different safety upgrades were adopted (the decision criteria behind those).

The approach on how the methodology for such a benchmark could be structured and implemented is discussed in this report. The report presents an example of a limited application of the proposed LTO benchmarking method to two NPPs using relevant available data (from the EU Fukushima Stress Test and 2017 WENRA Topical Peer Review on Ageing Management) in review areas of Ageing, and External hazards (sub-area of PSR safety factor Hazard analysis).

Therefore, the actual LTO benchmark methodology would need to be (further) developed, under the auspices of ENSREG and/or WENRA, to enable establishing a common approach that would facilitate achieving a harmonised high level of nuclear safety through the EU consistent with the NSD. In this process, the practical steps on how to account for safety upgrades that are “reasonably practicable” in specific context needs to be developed. This is to be followed by a pilot application that would allow for the assessment of the practicably of the approach in the context of benchmarking as required by ENSREG.

Noting the shown viability of the proposed LTO benchmarking method and envisaged benefits of its implementation, the following steps for further development are suggested:

1. The LTO benchmarking approach described herein is presented to EU regulators at a workshop and their feedback is sought.
2. *If the feedback is positive,* a Working Group of EU regulators is established aimed at guiding further development of the proposed LTO benchmarking approach. (A small WG composed of 3-4 members is suggested.)
3. The LTOC for the remaining 12 Safety Factors/ areas of review is developed, taking into account EU regulators comments (from step 2).

   LTOC will be derived from applicable current safety standards and internationally recognised good practices and will be in line with the IAEA Safety Guide on PSR and WENRA Reference Levels.

   The practical benchmarking criteria (LTOC) are developed in a way that would facilitate benchmarking against the Article 8b of the amended NSD 2014/87. This means limiting the number of criteria to essential nuclear safety characteristics (standards) for each Safety Factor/ area of review. Only criteria needed to ensure throughout LTO period the existence of effective defence-in-depth and the availability of the fundamental safety functions of Control-Cool-Contain (that are essential for practical exclusion of major releases) should be included in LTOC.

   The Ageing Management Criteria and External Hazard Criteria presented in this report can be used as a guide.
4. The WG reviews proposed LTOC in the 14 Safety Factors/ areas of review to arrive at practical LTOC satisfactory to the EU regulators.

5. Pilot study of the LTO benchmarking method is implemented by 2 – 3 NPPs using the LTO benchmarking method.

Preparation of an NPP Global LTO Benchmark assessment report for each NPP reviewed may prove to be difficult; pragmatic solutions will need to be developed based on experience in a pilot project. Another challenge will be in comparing/integrating individual NPP Global LTO Benchmark assessment reports to obtain an overall EU picture on safety levels of EU NPPs.

6. Based on the experience from the pilot study and any comments from DG ENER, ENSREG and WENRA, the approach is finalised, and documented it in a technical report, e.g. a guidelines document for benchmarking the safety level of EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87.

<u>FORESEEN BENEFITS OF IMPLEMENTATION OF THE PROPOSED LTO BENCHMARKING METHOD</u>

Reflecting the experience gained in assessing various benchmarking approaches and possibilities for their use in the contextual framework of the NSD, the following benefits of implementation of the proposed LTO Benchmarking method are envisaged:

a. It would provide a significant underlying incentive to EU regulators and NPP operators to harmonize their PSR implementation in line with the IAEA PSR Safety Guide and thus with the WENRA Reference Level Issue P because it would facilitate a consistent implementation of the LTO benchmark for the EU NPPs.

b. It would provide a qualitative assessment of the safety level of individual EU NPPs against the Articles 8a and 8b of the amended NSD 2014/87.

c. It would provide an overall picture of the safety level of EU NPPs and, like peer reviews under the Convention on Nuclear Safety and other peer reviews (e.g. OSARTs), provide an associated motivation for improvements where indicated.

d. It would enhance transparency and trust among the EU public that a harmonised high level of safety is being achieved across the EU Member States.

The underlying objective of the LTO benchmark and the ultimate goal is to achieve the harmonized high level of safety of all European NPPs consistent with the Articles 8a and 8b of the amended NSD 2014/87. This requires from EU MS follow up actions to address outcomes of the LTO Benchmark; these actions should deal effectively with any shortfalls (LTOC 'not met' of 'partially met') identified by the LTO Benchmark for their NPPs.

In the examples of application of the proposed LTO benchmarking to plants X and Y in review areas of Ageing and External hazards (see Section 5) , this would mean determination of appropriate 'reasonably practicable' safety improvements to address identified shortfalls in the compliance with the LTOC (indicated in LTO benchmark assessment matrices in App. 4 and 5). As noted in Section 2, assessments of the adequacy of proposed safety improvements need to take into account findings from all 14 PSR safety factor reviews (LTO benchmark reviews), and make a 'global risk' judgement on continued plant operation with any deficiencies remaining after implementation of all safety improvements. This means taking into account interactions and overlaps between safety factors as well as positive and negative findings of the benchmark review.

7 REFERENCES

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[4.13] INTERNATIONAL ATOMIC ENERGY AGENCY, External Human Induced Events in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.1, IAEA, Vienna (2002)

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[4.20] Volcanic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-21, IAEA, Vienna (2012)

APPENDIX 1 COUNCIL DIRECTIVE 2014/87/EURATOM

amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations

SECTION 2 **Specific obligations**

Article 8a

Nuclear safety objective for nuclear installations

1. Member States shall ensure that the national nuclear safety framework requires that nuclear installations are designed, sited, constructed, commissioned, operated and decommissioned with the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding:

(a) early radioactive releases that would require off-site emergency measures but with insufficient time to implement them;

(b) large radioactive releases that would require protective measures that could not be limited in area or time.

2. Member States shall ensure that the national framework requires that the objective set out in paragraph 1:

(a) applies to nuclear installations for which a construction licence is granted for the first time after 14 August 2014;

(b) is used as a reference for the timely implementation of reasonably practicable safety improvements to existing nuclear installations, including in the framework of the periodic safety reviews as defined in Article 8c(b).

Article 8b

Implementation of the nuclear safety objective for nuclear installations

1. In order to achieve the nuclear safety objective set out in Article 8a, Member States shall ensure that the national framework requires that where defence-in-depth applies, it shall be applied to ensure that:

(a) the impact of extreme external natural and unintended man-made hazards is minimised;

(b) abnormal operation and failures are prevented;

(c) abnormal operation is controlled and failures are detected;

(d) accidents within the design basis are controlled;

(e) severe conditions are controlled, including prevention of accidents progression and mitigation of the consequences of severe accidents;

(f) organisational structures according to Article 8d(1) are in place.

2. In order to achieve the nuclear safety objective set out in Article 8a, Member States shall ensure that the national framework requires that the competent regulatory authority and the licence holder take measures to promote and enhance an effective nuclear safety culture. Those measures include in particular:

(a) management systems which give due priority to nuclear safety and promote, at all levels of staff and management, the ability to question the effective delivery of relevant safety principles and practices, and to report in a timely manner on safety issues, in accordance with Article 6(d);

(b) arrangements by the licence holder to register, evaluate and document internal and external safety significant operating experience;

(c) the obligation of the licence holder to report events with a potential impact on nuclear safety to the competent regulatory authority; and,

(d) arrangements for education and training, in accordance with Article 7.

APPENDIX 2 AGEING AND EXTERNAL HAZARDS BENCHMARK CRITERIA

Table 4.1. AMC for high level evaluation of the overall plant AMP

Background info. NPP governance documents should provide direction for effective ageing management, i.e. should address NPP policy, organization and methodology for ageing management.

Review topics	Assessment basis/criteria
	Principle: Before an effective ageing management programme can be implemented, appropriate AMP strategy, AMP organizational arrangements and AMP methodology should be established and documented in NPP governance documents on AMP/LTO.
AMP organization	Ref. Section 4 of IAEA Safety Guide on Ageing management
1. AMP strategy, i.e. policy and principles for managing ageing	Senior NPP management should set out in NPP documents on AMP/LTO the policy and principles of the overall plant AMP including: • Objective of the overall plant AMP aimed at maintaining required integrity and functional capability of SSCs important to safety; • Scope of the overall plant AMP which includes SSCs important to safety.
2. AMP organizational arrangements	The overall plant AMP should have a structure and organization, documented in NPP procedures and org. charts, which facilitates that all assigned functions are performed satisfactorily. This includes: • AMP coordinator supported by a competent and sufficient staff to facilitate coordination of relevant NPP and external programmes, and AMP improvement; • Mechanisms to incorporate industry and in-house operating experience and R&D results relating to age-related degradation of SSCs; • Communication links with external agencies and other NPPs, specialized NPP teams and consultants in areas in which the AMP organization is not self-sufficient.
AMP methodology	Ref. Section 4 of IAEA Safety Guide on Ageing management
3. Data management in support of AMP	• Procedures should be available for the collection and record keeping of data in support of AMP as well as a description of the system.
4. Screening process to identify SSCs for AM	• A systematic screening process should be documented which is aimed to focus resources on those SSCs that can have a negative impact on the safe operation of the plant and that are susceptible to ageing degradation. • List of SSCs covered by AMP should be available.
5. Ageing management review	• Methodology should be documented for performing AM reviews for SSCs selected by the screening process to acquire information on the understanding, monitoring, and mitigation of SSC ageing mechanisms and effects. • Reports on the AM reviews should be available; and relevant AM reviews (e.g. prepared by the owners' group, suppliers or support organizations) should be used to minimize duplication of effort.
6. Condition assessment process	• Guidance should be available for performing condition assessment of SSCs selected by the screening process to determine current performance and condition of the SSCs and to

Review topics	Assessment basis/criteria
	provide prediction of future SSC performance, including their ageing degradation and, where feasible, remaining service life.
7. Process for the development of SSC-specific AMPs	Guidance should be provided for development of SSC-specific AMPs: • Review of current SSC-specific aging management practices in light of applicable AM reviews and condition assessments to either confirm their effectiveness or identify appropriate modifications of the current practices. • SSC-specific AMPs should have the attributes given in Table 2.

Table 4.2. AMC for evaluation of SSC-specific AMPs (general)

(of a representative sample of major SSCs important to safety)

Attribute/ Review topic	Description/ Assessment basis/criteria
1. Scope of the AMP based on understanding ageing	• Systems, structures (including structural elements) and components subject to ageing management; • Understanding of ageing phenomena (significant ageing mechanisms, susceptible sites): o SSC materials, service conditions, stressors, degradation sites, aging mechanisms and effects; o SSC condition indicators and acceptance criteria; o Quantitative or qualitative predictive models of relevant aging phenomena.
2. Preventive actions to minimize and control ageing degradation	• Identification of preventive actions; • Identification of parameters to be monitored or inspected; • Service conditions (i.e. environmental conditions and operating conditions) to be maintained and operating practices aimed at slowing down potential degradation of the structure or component.
3. Detection of ageing effects	• Effective technology (inspection, testing and monitoring methods) for detecting ageing effects before failure of the SSC.
4. Monitoring and trending of ageing effects	• Condition indicators and parameters to be monitored; • Data to be collected to facilitate assessment of structure or component ageing; • Assessment methods (including data analysis and trending).
5. Mitigating ageing effects	• Operations, maintenance, repair and replacement actions to mitigate detected ageing effects / degradation of SSCs.
6. Acceptance criteria	• Acceptance criteria against which the need for corrective action is evaluated.
7. Corrective actions	• Corrective actions if a component fails to meet the acceptance criteria.
8. Operating experience feedback and feedback of R&D results	• Mechanism that ensures timely feedback of operating experience and R&D results (if applicable), and provides objective evidence that they are taken into account in the AMP.
9. Quality management	• Administrative controls that document the implementation of the AMP and actions taken; • Indicators to facilitate evaluation and improvement of the AMP; • Confirmation (verification) process for ensuring that preventive actions are adequate and appropriate and all corrective actions have been completed and are effective; • Record keeping practices to be followed.

Table 4.3. AMC for assessment of the scope of AMP for LTO

Background info. The following table presents *guidance for evaluation of the AMP scope for LTO* (i.e. SSCs covered by the overall plant AMP) which is based on IAEA Safety Reports Series No. 57, Safe Long Term Operation of Nuclear Power Plants.

AMP for LTO is focused on ageing management of passive and long-lived SSCs (those without moving parts or without a change in configuration or properties) that are not subject to replacement based on a qualified life or specified time period. Aging management of active and/or replaceable components important to safety (e.g. pumps - except casing, valves - except body, motors, diesel generators, air compressors, control rod drives, ventilation dampers, pressure transmitters) is excluded from the AMP for LTO as it is to be provided for by Environmental Qualification Programme and Maintenance Programme.

Review topics	Assessment basis/criteria
SSCs within the scope of an overall plant AMP	• List of SSCs covered by AMP should be available. • List should include SSCs important to safety as well as SSCs that do not have safety functions but whose failure could prevent other SSCs from performing their intended safety functions. • List should include passive and long-lived SSCs (those without moving parts or without a change in configuration or properties) that are not subject to replacement based on a qualified life or specified time period, including: ○ reactor vessel ○ CANDU pressure tubes ○ reactor coolant system pressure boundary ○ steam generators ○ pressurizer ○ piping and tanks (including buried piping and tanks) ○ pump casings ○ valve bodies ○ core shroud ○ component supports ○ pressure retaining boundaries ○ heat exchangers ○ ventilation ducts ○ containment ○ containment liner ○ electrical and mechanical penetrations ○ equipment hatches ○ seismic Category I structures ○ electrical cables and connections ○ cable trays

Table 4.4. AMC for evaluation of TLAAs for LTO (general)

(of a selected sample of TLAAs)

Review topics	Assessment basis/criteria
Revalidation of TLAAs	• Revalidation of the original TLAAs should be done with respect to the assumed period of LTO. • The revalidation should confirm function and safety margins necessary for the whole period of LTO. • The revalidation of TLAAs should be documented in an update of the SAR. • If a TLAA cannot be revalidated, appropriate corrective or compensatory measures should be proposed for managing ageing effects of an S/C during LTO. • Revalidated TLAAs and any S/C specific corrective or compensatory measures for managing ageing effects should have been reviewed and accepted by the regulatory body.

Table 4.5. AMC for Flow Accelerated Corrosion AMP

Background info. The AMP includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. It relies on implementation of the EPRI guidelines (NSAC-202L-R2) or similar predictive codes for an effective FAC program. Operating experience shows that the above program, when properly implemented, is effective in managing FAC in high-energy fluids carbon steel piping and components.

Review topic	Description/ Assessment basis/criteria
1. Scope of the AMP based on understanding ageing	• Carbon steel lines and valve bodies containing two phase as well as single phase high-energy fluids • FAC, also termed erosion-corrosion, is an ageing mechanism involving corrosion and erosion in the presence of a moving corrosive fluid, leading to the accelerated loss of material. Susceptibility may be determined using the review process outlined in of the EPRI guidelines, including CHECWORKS or a similar predictive code.
2. Preventive actions to minimize and control ageing degradation	• The FAC program is an analysis, inspection, and verification program; thus, there is no preventive action. • Monitoring of water chemistry to control pH and dissolved oxygen content, and selection of appropriate piping material, geometry, and hydrodynamic conditions, are effective in reducing FAC. • Parameters to be monitored or inspected: wall thickness of piping and components
3. Detection of ageing effects	• Ultrasonic and radiographic testing at susceptible locations to detect wall thinning
4. Monitoring and trending of ageing effects	• CHECWORKS or a similar predictive code is used to predict susceptible locations of wall thinning and inspection schedule in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. • Inspection results are evaluated to determine if additional inspections are needed to assure that structural integrity will be maintained between inspections, and identify corrective actions.
5. Mitigating ageing effects	• Water chemistry to control pH and dissolved oxygen content • Materials, geometry, and hydrodynamic conditions resistant to FAC
6. Acceptance criteria	• Use inspection results in CHECWORKS to calculate the number of refuelling or operating cycles remaining before the component reaches the minimum allowable wall thickness. • If calculations indicate that an area will reach the minimum allowed wall thickness before the next scheduled outage, the component is to be repaired, replaced, or re-evaluated.
7. Corrective actions	• Prior to service, components for which the acceptance criteria are not satisfied are re-evaluated, repaired, or replaced.
8. Opex feedback and R&D results feedback	• Mechanism that ensures timely feedback of operating experience and R&D results (if applicable), and provides objective evidence that they are taken into account in the AMP.

Review topic	Description/ Assessment basis/criteria
9. Quality management	• QA procedures, review and approval processes, and administrative controls implemented in accordance with applicable regulatory requirements (e.g. in USA, 10 CFR Part 50, Appendix B).

Table 4.6. AMC for Buried Steel Piping and Tanks AMP

Background info. The AMP includes (a) preventive measures to mitigate corrosion by maintaining external coatings and wrappings, and (b) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried steel piping and tanks. The AMP review includes evaluation of programs to manage loss of material due to general, pitting, and crevice corrosion and MIC of steel piping (with or without coating or wrapping), piping components, and piping elements buried in soil.

Review topic	Description/Assessment basis/criteria
1. Scope of the AMP based on understanding ageing	• Buried steel piping and tanks that are within the scope of LR • Loss of material in these components, which may be exposed to aggressive soil environment, is caused by general, pitting, and crevice corrosion, and microbiologically-influenced corrosion.
2. Preventive actions to minimize and control ageing degradation	• Underground piping and tanks are coated during installation with a protective coating system, such as coal tar enamel with a fiberglass wrap and a kraft paper outer wrap, a polyolefin tape coating, or a fusion bonded epoxy coating to protect the piping from aggressive soil environment • parameters to be monitored: coating and wrapping integrity
3. Detection of ageing effects	• Coatings and wrappings are inspected by visual techniques whenever they are excavated during maintenance (so called Opportunistic inspections). Damaged wrapping or coating defects, e.g. coating perforation or other damage, is an indicator of possible corrosion of the external surface of piping and tanks. • Inspections are performed in areas with the highest likelihood or a history of corrosion problems, within the areas made accessible by a maintenance activity.
4. Monitoring and trending of ageing effects	• It is anticipated that one or more opportunistic inspections may occur within a ten-year period. Prior to entering the period of LTO, the NPP is to verify that there is at least one opportunistic or focused inspection is performed within the past ten years. During LTO, the NPP is to perform a focused inspection within ten years, unless an opportunistic inspection occurred within this ten-year period. Any credited inspection should be performed in areas with the highest likelihood of corrosion problems, and in areas with a history of corrosion problems. • Results of previous inspections are used to identify susceptible locations.
5. Mitigating ageing effects	• protective coating systems, such as coal tar enamel with a fiberglass wrap and a kraft paper outer wrap, a polyolefin tape coating, or a fusion bonded epoxy coating to protect the piping from aggressive soil environment
6. Acceptance criteria	• Any coating and wrapping degradations are reported and evaluated according to site corrective actions procedures.
7. Corrective actions	• Prior to service, components for which the acceptance criteria are not satisfied are re-evaluated, repaired, or replaced.

Review topic	Description/Assessment basis/criteria
8. Opex feedback and R&D results feedback	• Mechanism that ensures timely feedback of operating experience, and objective evidence that Op-ex is taken into account in the AMP. • plant-specific operating experience is evaluated because the inspection frequency is plant-specific and depends on the plant operating experience.
9. Quality management	• QA procedures, review and approval processes, and administrative controls implemented in accordance with applicable regulatory requirements (e.g. in USA, 10 CFR Part 50, Appendix B).

Background info. The AMP includes (a) preventive measures to mitigate corrosion by maintaining external coatings and wrappings, and (b) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried steel piping and tanks. The AMP review includes evaluation of programs to manage loss of material due to general, pitting, and crevice corrosion and MIC of steel piping (with or without coating or wrapping), piping components, and piping elements buried in soil.

Table 4.7. AMC for Insulation Materials of electrical cables important to safety AMP

Background info. In USA, ageing management of insulation materials of electrical cables important to safety is implemented through several AMPs.

- Ageing of electrical cables important to safety that could be exposed to harsh environment accident conditions (in-containment cables) is managed by Equipment qualification (EQ) programs (Section X.E.1 of GALL report [4.9]). EQ programs are viewed as aging management programs (AMPs) for license renewal.
- Ageing of electrical cables important to safety that are not subject to EQ requirements is managed by AMPs that have the attributes (ten elements) of a generic AMP described in Section XI.E.1 of GALL report [4.9].
- Ageing of electrical cables important to safety used in instrumentation circuits that are not subject to EQ requirements is managed by AMPs that have the attributes (ten elements) of a generic AMP described in Section XI.E.2 of GALL report [4.9].

EQ programs are viewed as aging management programs. They manage thermal, radiation, and cyclical aging of electrical equipment important to safety, including insulation materials of electrical cables, through the use of aging evaluations/ qualification methods so that qualified equipment, in its end-of-life condition, will meet its performance specifications during and following design basis accidents under the most severe environmental conditions postulated at the equipment's location after such an accident. EQ programs establish component 'qualified life' or 'qualified condition', and EQ components are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

In the USA, EQ does not include equipment located in mild environments, and typically relies on 'qualified life'. Some European countries (e.g. France and Germany) include in their EQ programs equipment important to safety located in mild environments and utilize 'qualified condition'. EQ programs that make use of qualified condition require component condition monitoring. (Typically, electrical parameters cannot be used as condition indicators (CIs) to indicate material properties and predict ageing of cable insulation.)

The following table presents AMC criteria for AMP of insulation materials of electrical cables important to safety which are a hybrid of the IAEA and USNRC guidance.

Attribute	**Description/**Assessment basis/criteria **/**
1. Scope of the AMP based on understanding ageing	• Electrical and I&C cables important to safety. • Polymeric insulating/jacket materials combined with other ingredients provide necessary mechanical, electrical and fire retardant properties. Knowledge of cable materials is necessary for cable monitoring and assessment. • Radiation and temperature-induced embrittlement in the presence of oxygen can lead to failure of conductor insulation. • Cable EQ provides the basis for cable AMP. • Standards: IEEE 323, IEC 60780, IEC 544-5, CSA 290.13 • IAEA guidance: TECDOC-1188
2. Preventive actions to minimize and control ageing degradation	• Environmental monitoring (T and radiation) to identify thermal hot spots using IR thermography and radiation hot spots using dosimeters • Correct steam, water, oil, etc. leaks affecting cables • Maintain thermal insulation on high temperature lines and equipment • Maintenance of heating, ventilation and A/C equipment • Parameters monitored: ambient environment - T, radiation dose rate
3. Detection of ageing effects	• <u>Visual/tactile inspection</u> of a representative sample of accessible electrical cables installed in adverse localized environments (hot spots) for cable and jacket surface anomalies, such as embrittlement, discoloration, cracking, or surface contamination. • <u>Condition monitoring</u> of selected cables using in-situ indenter measurements, thermal analysis of micro-samples, or tensile tests of samples from cable deposits or removed cables.
4. Monitoring and trending of ageing effects	• Trend of observed cable degradations from periodic visual inspections. • Trend of cable failures/replacements. • Trend of CIs determined by EQ, e.g. elongation at break, indenter modulus, oxidation induction time, or oxidation induction temperature; cable specific CI baseline data (including material data) are needed for trending of ageing effects.
5. Mitigating (detected) ageing effects	• Reduce severity of environmental conditions • Relocate cables to avoid hot spots
6. Acceptance criteria	• Cable <u>qualified life</u> determined by EQ. Qualified life of a cable may be extended by reanalysis of an ageing evaluation taking account of excess conservatism incorporated in the prior evaluation, such as the assumed ambient temperature of the cable or unrealistically low activation energy. • Cable <u>qualified condition</u> in terms of measurable CIs determined by EQ.
7. Corrective actions	• Replace degraded cables, cables with expired qualified life which cannot be extended, and cables which show greater degradation that the qualified condition given by a specific CI.
8. Operating experience feedback and feedback of R&D results	• NPP specific mechanisms that ensures timely feedback of op-ex and R&D results, and evidence that both plant-specific and relevant generic op-ex is taken into account in the AMP.

Attribute	**Description/**Assessment basis/criteria **/**
9. Quality management	• NPP specific – QA procedures, review and approval processes, and administrative controls in accordance with regulatory requirements. • EQ programme documentation is controlled under the QA programme. • Qualification files are maintained at the plant site in an auditable form. • Indicators of AMP effectiveness: trends of hot spots (T and radiation) by location, observed cable degradations, and cable failures/replacements.

Table 4.8. AMC for revalidation of Reactor Vessel Neutron Embrittlement TLAA

Background info. During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the reactor vessel beltline region and the region above and below the beltline of light-water nuclear power reactors. The following analyses of the reduction of fracture toughness of the RVs for 40 calendar years are TLAAs and must be evaluated for the period of LTO:

- Neutron Fluence Values
- Pressurized Thermal Shock (PTS)
- Upper-Shelf Energy
- Pressure-Temperature (P-T) Limits

The regulatory review determines whether the NPP has provided sufficient information and whether the regulatory evaluation supports conclusions of the following type:

The NPP has provided an acceptable demonstration that, for the reactor vessel neutron embrittlement TLAA

1. The analyses remain valid for the period of extended operation, or
2. The analyses have been projected to the end of the period of LTO, or
3. The effects of aging on the intended function(s) will be adequately managed for the period of LTO.

The regulator also concludes that the FSAR Supplement contains an appropriate summary description of the reactor vessel neutron embrittlement TLAA evaluation for the period of LTO.

Review topics	Assessment basis/criteria
Revalidation of RV neutron embrit. TLAAs	• Revalidation of the above TLAAs should be done with respect to the assumed period of LTO. • The revalidation should confirm function and safety margins necessary for the whole period of LTO. • The revalidation of TLAAs should be documented in an update of the FSAR. • If a TLAA cannot be revalidated, appropriate corrective or compensatory measures should be proposed for managing RV embrittlement during LTO. This should include a discussion of the flux reduction program (e.g., operation with a low leakage core design and/or integral burnable neutron absorbers) if necessary, and an identification of the viable options for managing the RV embrittlement in the future (incl. RPV material surveillance program and plant modifications (e.g., heating of ECCS injection water) which could limit the risk associated with postulated PTS events, more detailed safety analysis to show that the PTS risk for the facility is acceptably low, the potential for RPV thermal annealing). • Revalidated TLAAs and any corrective or compensatory measures for managing RV embrittlement during LTO should have been reviewed and accepted by the regulatory body.

Table 4.9. AMC for revalidation of Metal Fatigue of Class 1 Piping TLAA

Background info. The fatigue resistance of Class 1 piping during the period of LTO is an area of review. In addition, consideration of the effects of coolant environment on component fatigue life for LTO is an area of review. Metal components may be designed or analyzed based on requirements in the ASME Boiler and Pressure Vessel Code or the American National Standards Institute (ANSI) guidance or corresponding standards. ASME Section III Class 1 fatigue analysis requires the calculation of the "cumulative usage factor" (CUF) based on the fatigue properties of the materials and the expected fatigue service of the component. The ASME Code limits the CUF to a value of less than or equal to one for acceptable fatigue design. ANSI B31.1 specifies allowable stress levels based on the number of anticipated full thermal cycles.

In USA, NRC reviewer determines whether the applicant has provided sufficient information and whether the staff's evaluation supports conclusions of the following type:

The applicant has provided an acceptable demonstration, that for the metal fatigue TLAA

1. The analyses remain valid for the period of extended operation, or
2. The analyses have been projected to the end of the period of extended operation, or
3. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation (Ref. X.M1 of the GALL report).

The staff also concludes that the FSAR Supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation.

Review topics	Assessment basis/criteria
Revalidation of Metal fatigue TLAAs	• Revalidation of the above TLAAs should be done with respect to the assumed period of LTO. • The revalidation should confirm function and safety margins necessary for the whole period of LTO. • The revalidation of TLAAs should be documented in an update of the FSAR. • If a TLAA cannot be revalidated, appropriate corrective or compensatory measures should be proposed for managing fatigue of Class 1 piping during LTO. • Revalidated TLAAs and any corrective or compensatory measures for managing fatigue of Class 1 piping during LTO should have been reviewed and accepted by the regulatory body.

Table 4.10. AMC for revalidation of Environmental Qualification of electric equipment TLAA

Background info. EQ programs are viewed as aging management programs (AMPs) for license renewal. EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on US NRC 10 CFR 50.49 (Environmental qualification of electric equipment important to safety for nuclear power plants) qualification methods. EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses for license renewal.

Acceptance Criteria

NPP must demonstrate one of the following:

- The aging evaluation existing at the time of the renewal application for the component remains valid for the period of LTO, and no further evaluation is necessary.
- Qualification of the equipment is extended for the period of LTO by testing, analysis, or operating experience, or combinations thereof, in accordance with the CLB. (For this option, the reanalysis of an aging evaluation is performed to extend the qualification by reducing excess conservatisms incorporated in the prior evaluation, such as the assumed ambient temperature or unrealistically low activation energy.)
- The effects of aging on the intended function(s) will be adequately managed for the period of extended operation. (Acceptable generic AMP in GALL report, Section X.E.1.)

Review topics	Assessment basis/criteria
Revalidation of EQ TLAAs	• Revalidation of the EQ TLAAs should be done with respect to the assumed period of LTO. • The revalidation should confirm function and safety margins necessary for the whole period of LTO. • The revalidation of TLAAs should be documented in an update of the FSAR. • If a TLAA cannot be revalidated, an acceptable AMP should be proposed for managing aging during LTO. (Such an AMP includes the following EQ actions to maintain component aging effects within the bounds of the qualification basis: ensuring component specific installation and periodic maintenance, monitoring environmental conditions, evaluating the impact of any 'hot spots' when they are identified and taking appropriate corrective actions, considering relevant operating experience, maintaining EQ files at the plant site in an auditable form.) • Revalidated TLAAs or the proposed AMP/EQ program for LTO should have been reviewed and accepted by the regulatory body.

Table 4.11. Overall management of external hazards

Background info. The goal of the overall management of external hazards is to ensure the delivery of required safety functions and operator actions by providing adequate protection against relevant external hazards for SSCs important to safety, including the control room and the emergency control centre.

Review topics	Assessment basis/criteria
1. EH management program/ function	• NPP TSO should have a dedicated program or function that provides for an effective management of EH throughout NPP service life. • EH management program/ function should satisfy current national and international design codes, safety standards and guides • Experience of hazards and operating practices at nuclear power plants and at other relevant facilities should be taken into account. • Experience from managing actual events (for example, external floods, seismic events and tornadoes) should be used to improve existing procedures at the plant.
2. Relevant external hazards	• A list of relevant external hazards that may affect plant safety should be established. • Where a list of relevant externals hazards has already been established, this should be reviewed for completeness. • The following external hazards should be considered: • Floods, including tsunamis; • High winds, including tornadoes; • Fire; • Meteorological hazards (extreme temperatures, extreme weather conditions, high humidity, drought, snow, build-up of ice); • Sun storm; • Toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants, volcanic ash); • Hydrogeological and hydrological hazards (extreme groundwater levels, seiches); • Seismic hazards; • Volcano hazards; • Aircraft crashes, external missiles; • Explosion; • Biological fouling; • Lightning strike; • Electromagnetic or radio frequency interference; • Vibration; • Traffic
3.Documentation	• Documentation of EH analysis and safety analysis should include: elements of EH analysis process and safety analysis (deterministic or probabilistic) • background material for EH analysis and safety analysis

Review topics	Assessment basis/criteria
	• raw data (external hazards site records) and processed data (i.e. hazard curves), computer software and input and output files, • reference documents used for calculation (codes, standards, safety guides, best industry practices, etc.); • results of intermediate calculations and sensitivity studies.
4. Regulatory review	The overall management of external hazards should be reviewed and accepted by the national regulatory body.

Table 4.12. General EHC for review of current Site Characteristics and EH analysis

Background info. The table presents general review criteria for site characteristics and EH analysis (design vs. current). EH-specific criteria will be prepared for each relevant EH.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard analysis should be up to date and valid. • The analysis and/or methods should take account of the site characteristics and relevant international practice.
2. Site characteristics and EH analysis (EH-specific)	• For each relevant EH, a credible magnitude/intensity and associated frequency of occurrence of the hazard (hazard curves with different confidence levels) should be established. • Where, the EH magnitude/ and associated frequency of occurrence has already been established, this should be reviewed taking into account EH site specific records and relevant OPEX. Examples of EH-specific site characteristics and EH analysis results: • Seismic hazard parameters: site specific acceleration design response spectra, related time histories and ground motion durations due to both near and far field spectral ground motion. Other parameters may include peak ground acceleration, peak ground velocity, peak ground displacement. • Flooding hazard parameters: height of the water, the height and period of the waves (if relevant), the warning time for the flood, the duration of the flood and the flow conditions
3. Regulatory review	EH site characteristics and results of EH analysis should be reviewed and accepted by national regulatory body.

Note: Hazard curve = a representation of the annual frequency of exceeding a given parameter of the external hazard. For example, for seismic hazard, the most used given parameter is peak ground acceleration (PGA). Hazard family curves = a family of hazard curves for various confidence levels

Table 4.13. General EHC for review of adequacy of protection of NPP against external hazards (safety analysis)

Background info: This table presents criteria for review of the adequacy of current EH safety analysis and its results. The objective of the review of EH safety analysis is to determine the adequacy of protection (robustness) of the plant against external hazards, with account taken of current plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of LTO, and current analytical methods, safety standards and knowledge.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard safety analysis should be up to date and valid. • The analysis and/or methods should take account of the current plant design, site characteristics, condition of SSCs important to safety (both at present and predicted for the end of LTO) and relevant international practice. • Changes in plant design, the prevailing climate, the potential for floods and earthquakes, and transport and/or industrial activities near the site should be considered.
2. Desired outcomes of EH safety analysis	*Note: The adequacy of the preventive and mitigating measures can be evaluated by deterministic safety analysis or PSA.* EH safety analysis should demonstrate that: • The frequency of occurrence and/or the consequences of the hazard are sufficiently low so that either no specific protective measures are necessary, or • The preventive and mitigating measures in place are adequate, or • Identify any deficiencies in the preventive and mitigating measures.
3. Deterministic Safety Analysis applicable to the current external hazard	• Identification of differences between the results of the Deterministic Safety Analysis for the current EH (for example, Seismic Margin Assessment (SMA)) vs. that performed for the design basis EH. • Identification of potential cliff-edge effect for the current EH.
4. Probabilistic Safety Analysis (PSA) applicable to the current external hazard	• Identification of differences between the results of the Probabilistic Safety Analysis (for example, Seismic Probabilistic Safety Analysis (SPSA) performed for the current EH vs. that performed for the design basis EH. • Identification of potential cliff-edge effect for the current EH.
5. Robustness of the plant: - HCLPF (deterministic safety analysis methodology) - Fragility, CDF, LRF (probabilistic safety analysis methodology)	• Comparison between plant capacity expressed as HCLPF capacity for the current site characteristic parameter vs. the design (deterministic methodology) • Comparison between the plant capacity (fragility capacity), CDF and LRF for the current site characteristic parameter vs. the design (probabilistic safety analysis). • Confirmation there is no cliff-edge effect for that EH.
5. Regulatory review	Adequacy protection of NPP against EH determined by safety analysis should be reviewed and accepted by the national regulatory body.

Note. High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant.

- Fragility curve = probability of failure as a function of specific site characteristic level
- Family of fragility curves = a family of fragility curves for various confidence levels
- CDF = core damage frequency; LRF = large (early) relief frequency; PGA = peak ground acceleration

Table 4.14. General EHC for review of implementation of follow-up actions from EH safety analysis for current external hazards

Background info. This table presents criteria for the review of the implementation of follow-up actions based on the results of current deterministic/ probabilistic safety analysis for current external hazard. Practically, it is a verification of **t**he appropriateness of any physical improvements and procedures for operator actions aimed to prevent or mitigate the current external hazards.

Review topics	Assessment basis/criteria
1. Physical design modifications for detection and/or mitigation of consequences of the impact of specific current external hazard on plant safety	Physical design modifications reflect the findings and recommendations from Safety Analysis reports.
2.Procedures for mitigating the consequences of the impact of the current external hazard on plant safety	Procedures for mitigating the potential consequences of the impact of the current external hazard reflect the findings and recommendations from Safety Analysis reports.
3.Regulatory review	Physical design modifications and procedures for mitigation the potential consequences of the impact of the current EH should be reviewed and accepted by the national regulatory body.

Note. This table will be used in the benchmark review of EH only if there are any follow-up actions (improvements) based on the results of current deterministic/ probabilistic safety analysis for current external hazard.

Table 4.15. Seismic EHC for review of current Seismic Site Characteristics and Seismic EH analysis

Background info. The table presents review criteria for seismic site characteristics and Seismic Hazard analysis (design vs. current) and current analytical methods, safety standards and knowledge.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard analysis should be up to date and valid. • The analysis and/or methods should take account of the site characteristics and relevant international practice.
2. Seismic Site Characteristics and Seismic Hazard analysis	Seismic site characteristics review: • Evaluation of geological and geotechnical aspects at site area. • Information on pre-historical, historical and instrumentally recorded earthquakes in the region is collected, documented and used for determination of the seismic hazard. • Investigation for potential for surface faulting at site. • Consistency of selected ground motion attenuation relations with the tectonic features of the region. • Adequacy of uncertainty analysis of seismic sources parameters, ground motion models and site response for evaluation of site seismic hazard. Deterministic Seismic Hazard Analysis (DSHA) elements to review: o Evaluation of the seismo-tectonic model for the site region using seismic sources identified based on tectonic characteristics, earthquake occurrence probability, and the type of magnitude–frequency relationship; o Evaluation of maximum potential magnitude for each seismic source; o Selection of the attenuation relationships for the site region as a function of earthquake magnitude and seismic source-to-site distance; • Evaluation of the ground motion at site as the enveloping contribution of all sources considered. • Evaluation of site specific response spectrum as the maximum potential seismic hazard level of the site. Probabilistic seismic Hazard Analysis (PSHA) elements to review: • Evaluation of the seismo-tectonic model for the site region in terms of the defined seismic sources, including uncertainty in their boundaries and dimensions; • Evaluation for each seismic source of the maximum potential magnitude, earthquake occurrence probability and the type of magnitude–frequency relationship, together with the uncertainty associated with each evaluation; • Selection of the attenuation relationships for the site region, and assessment of the ground motion uncertainty. • Derivation of hazard curves for different confidence level; • Site-specific response spectra at free field or any specified condition from the hazard. • A family of seismic hazard curves has been developed for various confidence levels.
3. Regulatory review	EH site characteristics and the results of seismic hazard analysis should be reviewed and accepted by national regulatory body.

Note: - Hazard curve = a representation of the annual frequency of exceeding a given parameter of the external hazard. For example, for seismic hazard, the most used given parameter is peak ground acceleration (PGA) or acceleration response spectrum.

Hazard family curves = a family of hazard curves for various confidence levels

Table 4.16. Seismic EHC for review of adequacy of protection of NPP against seismic EH (safety analysis)

Background info: This table presents criteria for review of the adequacy of the current seismic hazard safety analysis and its results. The objective of the review of seismic safety analysis is to determine the adequacy of protection (robustness) of the plant against seismic hazard, with account taken of current plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of LTO, and current analytical methods, safety standards and knowledge.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard safety analysis should be up to date and valid. • The analysis and/or methods should take account of the current plant design, site characteristics, condition of SSCs important to safety (both at present and predicted for the end of LTO) and relevant international practice. • Changes in plant design, the prevailing climate, the potential for floods and earthquakes, and transport and/or industrial activities near the site should be considered.
2. Desired outcomes of seismic safety analysis	*Note: The adequacy of the preventive and mitigating measures can be evaluated by deterministic safety analysis or PSA.* Verification that: • The frequency of occurrence and/or the consequences of the seismic hazard are sufficiently low so that either no specific protective measures are necessary, or • The preventive and mitigating measures in place are adequate, or • Deficiencies in the preventive and mitigating measures are properly identified.
3. Deterministic Safety Analysis applicable to the current seismic EH	Deterministic Seismic Safety Analysis results vs. current site seismic hazard • Seismic Margin Assessment (SMA) elements reviewed are properly addressed: o RLE selection; o Selection of the assessment team; o Plant familiarization and data collection; o Selection of success path(s); o Determination of seismic response In-Structure-Response-Spectra (ISRS) of structures for input to capacity calculation; o Systems walk-down to review preliminary success path(s), select success path(s) and SSCs; o Seismic capability walk-down; o HCLPF calculations (SSCs and plant); o Peer Review; o Enhancements; and o Documentation • Confirmation there is no cliff-edge effect for the current seismic hazard.
4.PSA applicable to the current seismic EH	Probabilistic Seismic Safety Analysis results vs. current site seismic hazard • Seismic PSA elements reviewed are properly addressed: o Seismic Hazard Analysis; o Data collection and plant familiarization;

Review topics	Assessment basis/criteria
	○ Structural Response Analysis including SSI or Equipment Structure Interaction; ○ Seismic Fragility Evaluation; ○ Systems/Accident Sequence Analysis; ○ Risk Quantification; ○ Peer review Requirements; ○ Documentation Requirements; • Confirmation there is no cliff-edge effect for the current seismic hazard.
5. Robustness of the plant: HCLPF (deterministic safety analysis methodology) Fragility, CDF, LRF (probabilistic safety analysis methodology)	• Confirmation that the plant capacity expressed as HCLPF capacity is higher than the current seismic hazard OR safety improvements are needed (deterministic methodology). • Confirmation that the plant capacity expressed as fragility capacity is higher than the current seismic hazard OR safety improvements are needed (probabilistic safety analysis). • There is no cliff-edge effect for the current seismic hazard at site.
6. Regulatory review	Adequacy protection of NPP against seismic hazard determined by safety analysis should be reviewed and accepted by the national regulatory body.

Note:

- High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant
- Fragility curve = probability of failure as a function of specific site characteristic level
- Family of fragility curves = a family of fragility curves for various confidence levels
- CDF = core damage frequency
- LRF = large (early) relief frequency
- PGA = peak ground acceleration
- SMA = Seismic margin is expressed in terms of the earthquake ground motion level that compromises plant safety, specifically leading to severe core damage.
- RLE = Review Level Earthquake is an earthquake sufficiently larger than the DBE to ensure that the SMA challenges the capacity of the plant SSCs so that a plant HCLPF can be determined and "weak links" (if any) can be identified.

Table 4.17. Flooding EHC for review of current Flood Site Characteristics and Flood EH analysis

Background info. The table presents review criteria for flood site characteristics and Flood Hazard analysis (design vs. current) and current analytical methods, safety standards and knowledge.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard analysis should be up to date and valid. • The analysis and/or methods should take account of the site characteristics and relevant international practice.
2. Site characteristics and EH analysis (EH-specific)	Flood site characteristics review: height of the water, the height and period of the waves (if relevant), the warning time for the flood, the duration of the flood and the flow conditions. Deterministic Flood Hazard Analysis (DFHA) elements to review: • Evaluation of maximized hypothetical storms taking into account information, knowledge and results from the assessment of the meteorological hazards; • Evaluation of the hypothetical storms at locations that produce high water effect at the site; • Validation of the model selected for calculation of storm surge elevation taking into account site characteristics. • Evaluation of the maximum flood level. Probabilistic Flood Hazard Analysis (PFHA) elements to review: • Adequacy of database; data on storm and storm surge of the region is reliable covering a long period of time and for an adequate number of gage stations; • Adequacy of synthetic time series; the time series from several locations (regional and other representative stations) is correlated to provide basis for development of synthetic time series and is valid over a longer interval than the span of local observation; • Determination of surge levels through numerical simulations: evaluation of the storms surge level takes into account the basic factors like intensity, path and duration of storm, etc. when the record covers sufficiently long period of time. • Evaluation of the hazard curve representing the distribution of flood level with corresponding annual frequency of exceedance is acceptable. • A family of flood hazard curves has been developed for various confidence levels.
3. Regulatory review	EH site characteristics and the results of flood hazard analysis should be reviewed and accepted by national regulatory body.

Note:

- The flooding hazard criteria in this table have been developed for inundation hazard only. The other two types of flooding hazard, i.e. hydrodynamic forces and clogging, have not been included.
- The inundation hazard in this table is limited to storm surge hazard. Tsunami hazard has not been included, its occurrence at European NPP sites being considered extremely unlikely.

- Storm surges = abnormal rises of water surface elevation in near-shore areas of water bodies produced by high winds together with an atmospheric pressure reduction that occurs in conjunction with a severe meteorological disturbance.
- Hazard curve = a representation of the annual frequency of exceeding a given parameter of the external hazard. For example, for flood hazard, the most used given parameter is flood level.
- Hazard family curves = a family of hazard curves for various confidence levels

Table 4.18. Flooding EHC for review of adequacy of protection of NPP against flood hazard (safety analysis)

Background info: This table presents criteria for review of the adequacy of the current flood safety analysis and its results. The objective of the review of flood safety analysis is to determine the adequacy of protection (robustness) of the plant against flood hazard, with account taken of current plant design, site characteristics, the actual condition of the SSCs important to safety and their predicted state at the end of LTO, and current analytical methods, safety standards and knowledge.

Review topics	Assessment basis/criteria
1. Attributes of analytical methods used	• The analytical methods, safety standards and information used for the external hazard safety analysis should be up to date and valid. • The analysis and/or methods should take account of the current plant design, site characteristics, condition of SSCs important to safety (both at present and predicted for the end of LTO) and relevant international practice. • Changes in plant design, the prevailing climate, the potential for floods and earthquakes, and transport and/or industrial activities near the site should be considered.
2. Desired outcomes of flood safety analysis	*Note: The adequacy of the preventive and mitigating measures can be evaluated by deterministic safety analysis or PSA.* Verification that: • The frequency of occurrence and/or the consequences of the flood hazard are sufficiently low so that either no specific protective measures are necessary, or • The preventive and mitigating measures in place are adequate, or • Deficiencies in the preventive and mitigating measures are properly identified

3. Deterministic Safety Analysis applicable to the current flood hazard.	<u>Deterministic Flood Safety Analysis results vs. current site flood hazard</u>
	• Flood Margin Assessment (SMA) elements reviewed are properly addressed:
	  o RLF selection;
	  o Selection of the assessment team;
	  o Plant familiarization and data collection;
	  o Selection of success path(s);
	  o Systems walk-down to review preliminary success path(s), select success path(s) and SSCs;
	  o Flood capability walk-down;
	  o HCLPF calculations (SSCs and plant);
	  o Peer Review;
	  o Enhancements; and
	  o Documentation
	• Confirmation there is no cliff-edge effect for the current flood hazard.

4. Flood PSA applicable to the current flood hazard	<u>Flood PSA results</u> <u>vs. current site flood hazard</u>
	• Flood PSA elements reviewed are properly addressed:
	  o Flood Hazard Analysis;
	  o Data collection and plant familiarization;
	  o Structural Response Analysis including SSI or Equipment Structure Interaction;

Review topics	Assessment basis/criteria
	○ Flood Fragility Evaluation; ○ Systems/Accident Sequence Analysis; ○ Risk Quantification; ○ Peer review Requirements; ○ Documentation Requirements; • Confirmation there is no cliff-edge effect for the current flood hazard.
5. Robustness of the plant: HCLPF (deterministic safety analysis methodology) Fragility, CDF, LRF (probabilistic safety analysis methodology)	• Confirmation that the plant capacity expressed as HCLPF capacity is higher than the current flood hazard OR safety improvements are needed (deterministic methodology). • Confirmation that the plant capacity expressed as fragility capacity is higher than the current flood hazard OR safety improvements are needed (probabilistic safety analysis). • There is no cliff-edge effect for the current flood hazard at site.
6. Regulatory review	Adequacy protection of NPP against flood hazard determined by safety analysis should be reviewed and accepted by the national regulatory body.

Note:

- High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant
- Fragility curve = probability of failure as a function of specific site characteristic level
- Family of fragility curves = a family of fragility curves for various confidence levels
- CDF = core damage frequency
- LRF = large (early) relief frequency
- FMA = Flood margin is expressed in terms of the flood level that compromises plant safety, specifically leading to severe core damage.
- RLF = Review Level Flood is a flood sufficiently larger than the Design Basis Flood (DBF) to ensure that the FMA challenges the capacity of the plant SSCs so that a plant HCLPF can be determined and “weak links” (if any) can be identified.

APPENDIX 3 LTO BENCHMARK ASSESSMENT MATRICES

Table 4.19. Safety Factor Specific Generic Review Matrix

Safety Factor: ______________ **NPP: _______________**

Review area: _______________

Review topic	**Fully met**	**Partly met**	**Not met**	**Remarks**
Topic 1				
Topic 2				
Topic 3				
Topic 4				
Topic 5				

References:

Table 4.20. Example of a SF-specific Review matrix application

NPP: Plant X **Safety Factor: AGEING**

Review area: SSC-specific AMP - Insulation Materials of electrical cables important to safety

Review topic	Fully met	Partly met	Not met	Remarks
1. Scope of AMP based on understanding ageing		√		The AMP scope of the electrical cables important to safety is defined in Ref. 1. Labelling of cables does not always give info on the type of the cables; different generations of cables are used and some have not been manufactured according to QA standards. Some cables in the reactor building are still of the 1st generation and have to be replaced asap. Replacement program of PVC cables has been completed, but also Hypalon cables have to be replaced asap, because they also contain Chlorine and tests have shown that chlorine in nuclear zones could be dangerous for the safety.
2.Preventive actions to minimize ageing degradation		√		Ref. 2 states that “No preventive action is possible on the existing cables”. Therefore, replacement of the first generation cables is needed as soon as possible.
3.Detection of ageing effects			TBD	Not addressed in references.
4.Monitoring and trending of ageing effects		√		The evolution of the mechanical and electrical characteristics is monitored by taking samples for examination. For first generation cables, however there is no baseline information.
5. Mitigating ageing effects			TBD	Not addressed in references.
6. Acceptance criteria		√		It is stated that it is mandatory to replace cables before their Projected Qualified Life expires. The criteria are not explicitly addressed.
7. Corrective actions		√		Replacement
8. Opex feedback and R&D results feedback	√			The feedback of operating experience is described in Ref. 1.

Review topic	Fully met	Partly met	Not met	Remarks
9. Quality management		√		The AMP for aging of electrical cables important for safety is part of the global quality management system. Because of lacking of reference information for first generation cables, the procedures cannot fully be applied.

Summary: In Plant X different generations of cables are used. Some cables in the reactor building are still of the first generation and have to be replaced as soon as possible. The AMP is well described but parts still under development; therefore the AMP only partly meets relevant AMC.

References: [1] … [2] …

Table 4.21 Safety Factor Specific Generic Report Card

NPP: Plant X **Safety Factor: AGEING**

Review area	AMC Fully met	AMC Partly met	AMC Not met	Remarks
Overall plant AMP				
SSC-specific AMPs				
AMP scope for LTO				
TLAA revalidation				

Table 4.22. Example of a Safety Factor Specific Generic Report Card application

NPP: Plant X **Safety Factor: AGEING**

Review area	AMC Fully met	AMC Partly met	AMC Not met	Remarks
Overall plant AMP		√		In the document "Ageing Management" the NPP operator describes the Ageing Management Program. The document is established at the end of each calendar year for the two next years. In general, Overall plant AMP fully meets all relevant AMC except for Data management in support of AMP which is yet to be implemented before the start of LTO.
SSC-specific AMPs	√ √	√		Two SSC-specific AMPs fully meet the AMC : - FAC of high energy carbon steel piping - Buried piping and tanks - AMP for Insulation materials of electrical cables partly meets the AMC.
AMP scope for LTO	√			Report entitled Long Term Operation, which has been approved by the regulatory body, presents a list of SSCs covered by AMP. AMP scope for LTO meets the relevant AMC.
TLAA revalidation	√ √	√		Revalidation of the following TLAAs done: - Reactor Pressure Vessel (RPV) - Metal Fatigue Revalidation of Environmental qualification of electrical equipment to be completed.

Table 4.23. Global Assessment matrix **NPP: Plant X**

Safety Factor	LTOC Fully met	LTOC Partly met	LTOC Not met	Shortfall & Safety significance	Ref.
Plant					
1. Plant design					
2. Actual condition of SSCs					
3. Equipment qualification					
4. Ageing					
Safety analysis					
5. Deterministic safety analysis					
6. Probabilistic safety analysis					
7. Hazard analysis					

Safety Factor	LTOC Fully met	LTOC Partly met	LTOC Not met	Shortfall & Safety significance	Ref.
Performance & OpEx feedback					
8. Safety performance					
9. Use of experience from other plants and research					
Management					
10. Organization and administration					
11. Procedures					
12. Human factors					
13. Emergency planning					
Environment					
14. Radiological impact on the environment					

APPENDIX 4 LTO BENCHMARK ASSESSMENT MATRICES FOR NPP X

NPP: X **Safety factor: AGEING**

Review area: Overall plant AMP

Review topic	Fully met	Partly met	Not met	Remarks
AMP organization				
1. AMP strategy, i.e. policy and principles for managing ageing		?		Not addressed.
2. AMP organizational arrangements	√			The AMT (aging management team) is responsible for a systematic ageing management process, incl. ageing assessments. AMT also identifies and evaluates internal and external ageing experience and developments, and if applicable, advises relevant NPP X department(s) about appropriate changes in SSC-specific AMPs. The ageing management process is part of the Integrated Management System which is under control by the QA department.
AMP methodology				
3. Data management in support of AMP	√			Specific ageing management database (COMSY, developed by AREVA Germany) contains all AM relevant information.
4. Screening process to identify SSCs for AM	√			SSCs for LTO AM were determined by a documented scoping and screening process based on IAEA SRS No. 57. They include passive SCs important to safety; ageing of active SCs important to safety is managed by preventive maintenance and surveillance. List of SSCs covered by the overall AMP is available.
5. Ageing management review (AMR)	√			AMT is also responsible for generic Ageing Management Reviews (AMRs).
6. Condition assessment process	√			Condition assessment is part of the LTO assessment which includes AMRs that are consistent with the IAEA guidelines.

Review topic	Fully met	Partly met	Not met	Remarks
7. Process for the development of SSC-specific AMPs	√			An integrated SSC-oriented ageing management process based on the AMR is implemented. AMT is owner of this process and responsible for coordinating the process. Every SSC-specific AMP has the nine attributes according to IAEA and is consistent with IGALL. In addition, ageing phenomenon (e.g. FAC, fatigue) based AMPs are implemented.

Summary. NPP X *Overall AMP, in general, fully meets the relevant AMC*; however, information on AMP strategy, i.e. policy and principles for managing ageing is not explicitly addressed in the reference.

Reference. National Assessment Report for the TPR 2017 Ageing Management, Sec.2.3, December 2017

NPP: X **Safety factor: AGEING**

Review area: Scope of AMP for LTO

AMC	Fully met	Partly met	Not met	Remarks
1. List of SSCs covered by AMP should be available.		√		SSCs in the LTO scope were identified and documented in close cooperation between NPP Y and vendor . The reference includes only a partial list. *However, it is known that NPP X has adequately identified the systems and components within the scope of long term operation, in accordance with IAEA guidelines.*
2. List should include (passive and long-lived SSCs that are not subject to replacement based on a qualified life or specified time period)				
• reactor vessel	√			
• reactor coolant system pressure boundary	√			
• steam generators	√			
• pressurizer	√			
• piping and tanks (including buried piping and tanks)		√		Only Main Coolant Lines and pressurizer surge line explicitly identified. Other piping and tanks only as part of in-scope mechanical systems.
• pump casings		√		Not explicitly identified; only as part of in-scope mechanical systems.
• valve bodies		√		Not explicitly identified; only as part of in-scope mechanical systems.
• core shroud	√			Identified only as part of RPV internals.
• component supports	√			
• pressure retaining boundaries	√			Identified as part of in-scope mechanical systems.
• heat exchangers	√			
• ventilation ducts	√			Identified as part of Heating, Ventilation and Air-Conditioning systems.

AMC	Fully met	Partly met	Not met	Remarks
• containment		√		Only Steel containment identified.
• containment liner	√			
• electrical and mechanical penetrations	√			
• equipment hatches			√	
• seismic Category I structures		√		
• electrical cables and connections	√			
• cable trays	√			

<u>Summary</u>. The reference provides only a high level, mostly system based listing of in-scope SSCs. For the purpose of the LTO Benchmark example application, NPP Y *Scope of AMP for LTO 'AMC Partly met'*. However, it is known that NPP Y has adequately identified the systems and components within the scope of long term operation, in accordance with IAEA guidelines.

<u>Reference.</u> National Assessment Report for the TPR 2017 Ageing Management, Sec.2.3.1.1, December 2017

NPP: X **Safety factor: AGEING**

Review area: SSC-specific AMP - Electrical cables

AMC	Fully met	Partly met	Not met	Remarks
1. Scope of AMP based on understanding ageing	√			Cables and wires are divided into component groups and subgroups. Groups of cables or wires with the same materials, application, age, environmental conditions can be treated as groups. For example: Spreader wiring; Wiring in electrical cabinets; Power cables; Signal cables; Cables with design base accident requirements. AM is focused on el. insulation.
2.Preventive actions to minimize ageing degradation	√			Comprehensive ageing assessment of cables and wires considered stressors (temperature, radiation, electrical field and moisture linked to cable materials) and their impacts on the materials. The aim of this assessment was to focus on cables where a possibility of significant degradation due to the considered stressors and ageing mechanisms exist. Monitoring of the environmental conditions in the containment; on-going EQ programme for cables with a design base accident requirement. Replacement of cables whose qualified life is not sufficient for the period of long term operation until 2034. Programme to monitor the environmental temperature and radiological loads in the several buildings has been carried out and will be repeated in 2026 to check the stability of the environmental conditions.
3.Detection of ageing effects	√			Detection of ageing degradation before it leads to malfunctioning of electrical circuits is the main goal of ageing management of cables and wires. Strategy: focus is on the safety cables and wires sensitive for ageing degradation; visual inspections to detect signs of ageing degradation in an early stage (e.g. color change, cable cracking or swelling); testing of mechanical and/or electrical properties if necessary; replacement if necessary. Critical cables were determined (about 200) taking into account all possible influences on ageing degradation. Visual and other inspections and tests are focused on these cables.
4.Monitoring and trending of ageing effects	√			Cable condition monitoring program has been implemented. It includes: annual or maintenance related visual inspections, elongation-at-break tests (in-situ and lab.), and measurement of electrical insulation resistance.
5. Mitigating ageing effects	√			Actions in response to cable degradation may include increased frequency of testing, additional evaluation, and replacement.
6. Acceptance criteria	√			Not addressed explicitly. EQ cables are replaced if qualified life is not sufficient for the period of LTO until 2034. Other cables would be replaced if insulation cracking observed or elongation-at-break test failed.

AMC	Fully met	Partly met	Not met	Remarks
7. Corrective actions	√			Recommendations for replacement, testing, or inspection takes into account the kind of wire/cable, the environment, safety relevance and test results of representative samples.
8. Opex and R&D results feedback				The feedback of operating experience is carried out by AMT (aging management team).
9. Quality management				The AMP for aging of electrical cables important for safety is part of the global KCB quality management system.

Summary. AMP for electrical cables and wires as described in the reference is not easy to correlate with the AMC, but in general *the relevant AMC are 'Fully met'.*

Reference. National Assessment Report off the TPR 2017 Ageing Management, Sec.3, December 2017

NPP: X **Safety factor: AGEING**

Review area: SSC-specific AMP - Buried steel piping and tanks

AMC	Fully met	Partly met	Not met	Remarks
1. Scope of AMP based on understanding ageing	√			- Part of the auxiliary and emergency cooling water system (system VF) is composed of composite concrete/steel piping buried in soil. VF-piping that is exposed to soil on the outside, sea water on the inside was replaced in 2012. Identified ageing mechanisms include freeze-thaw, erosion, corrosion, carbonation. - Rubber seals between cooling water pipe line and rigid steel flange for the intake building. - Back-up residual heat removal water cooling system (System VE) uses buried glass fibre reinforced epoxy (GRE) pipes; no relevant ageing mechanisms were identified. - The buried pipes of the low pressure fire extinguishing system UJ-system consist of GRE piping, Polyethylene (PE) piping, and partly steel piping (general corrosion).
2.Preventive actions to minimize ageing degradation	√			No actions have been defined for preventing ageing of the concealed water pipes. All ageing prevention is accomplished by the design.
3.Detection of ageing effects	√			Both visual inspection and leak testing are applied. - 3-yearly visual video inspection and 5-year settlement survey of VF lines. - Monthly functional testing and leakage monitoring of VE lines and UJ lines.
4.Monitoring and trending of ageing effects	√			- Ageing effects found or any damage to the SSCs important to safety are mapped and recorded in the inspection report according to their size and nature. - Leakage in UJ and VE lines is detected by monitoring the number of starts of the jockey pump that pressurizes the systems. With any leakage in the system, the frequency of jockey pump starts would rise. This parameter is trended in the plant's control room. - Settlements of the cooling water lines VF is measured and compared with the as-built condition on a 5-yearly basis.
5. Mitigating ageing effects	√			When it is identified that corrective action is required, a dedicated corrective action programme (remedial) for the concerned UJ, VE and VF pipelines is applicable.
6. Acceptance criteria	√			Damage is evaluated in accordance with Dutch Nuclear Safety Guide NVR-NS-G-2.6, the 2007 ASME XI Code for safety class 1, 2 and 3, relevant DIN standards, and national regulations on steam and pressure equipment.
7. Corrective actions	√			Replacement or repair, as required.
8. Opex and R&D results feedback	√			The feedback of operating experience is carried out by AMT (aging management team).

AMC	Fully met	Partly met	Not met	Remarks
9. Quality management	√			Part of the global NPP X quality management system.

Summary. AMP for Buried steel piping as described in the reference is not easy to correlate with the AMC, but in general, *the relevant AMC are 'Fully met'*. Reference. National Assessment Report of the NPP X for the TPR 2017 Ageing Management, Sec.4, December 2017

Safety Factor AGEING Report Card

NPP: X **Safety Factor: AGEING**

Review area	AMC fully met	AMC partly met	AMC not met	Remarks
Overall plant AMP: 1. AMP strategy 2. AMP org. arrangements	√			AMP strategy, i.e. policy and principles for managing ageing not addressed - not in the scope of the Topical Peer Review. AMP organizational arrangements fully met.
SSC-specific AMPs	√			AMP for Electrical cables and AMP for Concealed pipework, as described in the reference, is not easy to correlate with the AMC but in general both AMPs meets the AMC.
AMP scope for LTO		√		The reference provides only a high level, mostly system based listing of in-scope SSCs, as required by the Topical Peer Review specs. *However, it is known that* NPP X *has adequately identified the systems and components within the scope of LTO, in accordance with IAEA guidelines.*
TLAA revalidation	N/A			N/A – not in the scope of the Topical Peer Review.

Summary. Scope/topics of the Topical Peer Review do not match the AMC scope/review areas for Safety Factor Ageing. Taking this into account, NPP X Ageing Management fully meets applicable AMCs - *Overall, the AMC for NPP X Ageing Management are 'Fully met'.*

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Overall management of external hazards (seismic hazard + flood hazard)

Review area	AMC Fully met	AMC Partly met	AMC Not met	Remarks
1. EH management program/ function			√	• The Stress Test report, ref. [1] below, does not give info on a dedicated program that provides for an effective management of EHs throughout NPP service life.
	√			• The Stress Test report is following the ENSREG's guidance provided in stress tests specifications, Ref. [2] below. ENSREG requirements are consistent with IAEA recommendations on this topic (Ref. [3]).
	√			• The experience with EHs and operating practices at NPPs and other relevant facilities is accounted for as claimed in para 2.2.2 (pg. 39) and para 7.1, pg. 88-89 of Ref. [1].
	√			• Experience from managing actual external events has been used/ is being developed to improve existing procedures at the plant, para 7.2.2, pg. 92 of Ref. [1] which follow guidance provided in Ref. [4].
2. Relevant external hazards	√			• A list of relevant external hazards that may affect plant safety has been established and reviewed for completeness following the guidance provided in Ref. [2].
				• The following external hazards have been considered in stress test report:
	√			○ seismic hazard;
	√			○ external flood;
	√			○ maximum and minimum water temperatures of the River Westerschelde;
	√			○ extremely high and low air temperatures;
	√			○ extremely high wind (including storm and tornado);
	√			○ wind missiles and hail;
	√			○ formation of ice;
	√			○ heavy rainfall;
	√			○ heavy snowfall;
	√			○ lightning;
	√			○ credible combinations of the conditions mentioned above.
	√			○ external fire;
	√			○ toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants);
	√			○ aircraft crashes, external missiles (required by the national regulator but not an ENSREG requirement);
	√			○ lightning strike.

Review area	AMC Fully met	AMC Partly met	AMC Not met	Remarks
3. Documentation			√ √ √ √ √ √	Documentation on EH analysis and safety analysis is not provided: • insufficient elements of EH analysis process and safety analysis (deterministic or probabilistic) • insufficient background material for EH analysis and safety analysis • no raw data (site external hazards records) and processed data (i.e. hazard • curves), computer software and input and output files, • insufficient reference documents used for calculation (codes, standards, safety guides, best industry practices, etc.); • no results of intermediate calculations and sensitivity studies.
4. Regulatory review	√			The Stress Test report has been reviewed and accepted by the national regulator taking into consideration the NPP X commitment for future work and implementation of safety improvements.

Note

Summary: *EHC for Overall management of external hazards (seismic hazard + flood hazard) are "Partly met'.*

References:

[1] National report on the Post-Fukushima Stress Test for the X Nuclear Power Plant, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards (draft)

[4] WANO, SOER (Significant Operating Experience Report) 2011-2.

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Current Site Seismic Characteristics and Seismic Hazard Analysis

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√		√	• Insufficient information in the test report on the analytical methods and the safety standards used to address the site seismic hazard. Even so, the information used for the seismic hazard analysis is outdated. • The analysis and methods take account of the site seismic characteristics and relevant international practice available at the time the stress test report, Ref. [1], was drafted.
2. Site Seismic Characteristics			√ √ √ √ √ √	• Site seismic characteristics for design review: ○ the information for evaluation of geological and geotechnical aspects at site area is outdated. ○ the information presented in Ref. [1] on pre-historical, historical and instrumentally recorded earthquakes in the region, collected, documented and used for determination of the seismic hazard is insufficient. ○ there is no investigation for potential for surface faulting at site. ○ there is no information on the selected ground motion attenuation relations and no confirmation if they are consistent with the tectonic features of the region. ○ no information is provided on the uncertainty analysis of seismic sources parameters, ground motion models and site response for evaluation of site seismic hazard. • Sufficient up to date information on the site seismic characteristics vs. design provided and accounted for in Seismic Hazard Analysis
3. Seismic Hazard analysis (DSHA method. OR PSHA method.)			√ √ √ √	**Deterministic Seismic Hazard Analysis (DSHA)** • Below are listed the findings resulted from the review of the elements of DSHA application to NPP X: ○ there is no description of the seismo-tectonic model used for the site region accounting for seismic sources identified based on tectonic characteristics, earthquake occurrence probability, and the type of magnitude–frequency relationship; ○ there is no description of the maximum potential magnitude for each seismic source; ○ there is no description of the attenuation relationships for the site region as a function of earthquake magnitude and seismic source-to-site distance; ○ the evaluation of the overall ground motion at site does not show the contribution of all sources considered.

Review topics	Fully met	Partly met	Not met	Remarks
			√	○ as a result of DSHA, the DBE has been defined as 0.6m/sec2 (0.06g) for the alluvial ground conditions and 0.75m/sec2 (0.076g) for medium ground conditions at the basis of the pile foundation corresponding to a return period of 30,000 years. **Probabilistic seismic Hazard Analysis (PSHA) not applied.**
4. Regulatory review	√			The site seismic hazard characteristics and the results of seismic hazard analysis have been reviewed and accepted by the National regulator as “plausible” with recommendation for update and additional clarification, para 2.2, pg. 11-12 of *Post Fukushima Stress Test of the NPPX* included in Ref. [1].

Note:

- DBE = Design Basis Earthquake
- PGA = peak ground acceleration
- DSHA = Deterministic Seismic Hazard Analysis
- PSHA = Probabilistic Seismic Hazard Analysis

Summary: The Deterministic Seismic Hazard Analysis (DSHA) has been used for characterization of the seismic hazard at site. The plant was not originally seismically designed due to very low seismic hazard at site (according to records covering almost 2000 years). Later on, a DBE has been defined for the NPP as 0.6m/sec^2 (0.06g) for the alluvial ground conditions and 0.75m/sec^2 (0.076g) for medium ground conditions at the basis of the pile foundation the with a return period of 30,000 years**.** The DBE has been revised periodically as part of the 10-year PSR without a need for change. However, it has no be noted that according to IAEA SSG-9, *Seismic Hazards in Site Evaluation for Nuclear Installations*, the PGA for DBE **cannot be less than 0.1g**. Considering that the NPP X revises the DBE periodically, the redefinition of the DBE should be considered.

In general, the site seismic hazard characterization including the determination of DBE is poorly addressed lacking to meet the EHC. This observation is consistent with National regulator’s comment, item 2.1.1.3, pg. 20, of *Post Fukushima Stress Test of the NPP X* included in Ref. [1]. *Overall, EHC for current Site Seismic Characteristics and Seismic EH analysis ‘Not met’.*

References:

[1] National report on the Post-Fukushima Stress Test for the NPP X, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Adequacy of protection against current seismic hazard (safety analysis)

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√	√	√	• The analytical methods, safety standards and information used for seismic hazard safety analysis are not up to date. • The analysis and/or methods take account of the current plant design, existing site seismic characteristics, condition of SSCs important to safety at present (*but not for their condition predicted for the end of LTO*) and relevant international practice such as ENSREG (Ref. [2]) and IAEA (Ref. [3]) on this topic, • Ref. [1] takes into consideration the changes in plant design and the potential for earthquakes.
2. Desired outcomes of seismic safety analysis	√ √ √			• the frequency of occurrence and/or the consequences of the seismic hazard are sufficiently low so that either no specific protective measures are necessary; OR • additional preventive and mitigating measures in place are in place or planned; OR • deficiencies in the preventive and mitigating measures are properly identified.

3. Safety Analysis for current seismic hazard
 	
(DSSA method
OR
SPSA method.)
		√
		√
				Deterministic Seismic Safety Analysis (DSSA)
				• No proper Seismic Margin Assessment (SMA) has been performed. However, the report, Ref. [1], pg. 39, states that "*In a simplified assessment, licensee has shown that there are significant seismic margins with respect to the fundamental safety functions. The lowest HCLPF capacity of all considered SSCs has been estimated to be 0.15 g. The HCLPF capacity for many safety-relevant systems and bunkered buildings*".
				• 4 potential cliff-edge effects have been identified in Ref. [1] but mitigating measures have not been identified:
				  o *Unavailability of shift personnel after 10 hours*. There is a potential for a cliff-edge with design-exceeding earthquakes if the main control room is destroyed combined with a situation in which the site becomes inaccessible. This would lead to a high-pressure core melt scenario after the Back-up feedwater system (RS) storage tanks were drained.
				  o *Structural failure of missile shield inside containment at PGAs > 0.3 g.* Such a scenario may induce a core melt while containment integrity is not ensured.

Review topics	Fully met	Partly met	Not met	Remarks
		√ √ √		○ *Possible failure of the containment filtered venting system*. The filtered venting system is not qualified for the design-basis earthquake. ○ *Possible inoperability of the fire-fighting systems in buildings 01, 02 and 35.* The fire-fighting systems in buildings 01, 02 and 35 in principle are not qualified for the design-basis earthquake. **Seismic Probabilistic Safety Analysis (SPSA) not applied.**
4. Robustness of the plant (DSSA method. OR SPSA method).	√		√ √	**Deterministic Seismic Safety Analysis (DSSA)** • Confirmation that the plant capacity expressed as HCLPF capacity is higher than the Design Basis Earthquake (DBE) OR safety improvements are needed (deterministic methodology). • No Spent Fuel Pool capacity (HCLPF) was provided • Cliff-edge effects for BDBE have been identified (see item 3 above). **Seismic Probabilistic Safety Analysis (SPSA) not applied.**
5. Regulatory review	√			• The adequacy protection of NPP against a BDBE determined by safety analysis has been reviewed and accepted by the National regulator with additional comments for improvement, para 2.2, pgs. 11-12 of Post Fukushima Stress Test of the NPP X included in Ref. [1].

Note: For stress report evaluation, the current seismic hazard is Beyond Design Basis Earthquake (BDBE)

Note.

- High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant
- PGA = peak ground acceleration
- SMA = Seismic Margin Assessment. Seismic margin is expressed in terms of the earthquake ground motion level that compromises plant safety, specifically leading to severe core damage.
- RLE = Review Level Earthquake is an earthquake sufficiently larger than the DBE to ensure that the SMA challenges the capacity of the plant SSCs so that a plant HCLPF can be determined and "weak links" (if any) can be identified.
- BDBE = Beyond Design Basis Earthquake

- DSSA = Deterministic Seismic Safety Analysis
- SPSA = Seismic Probabilistic Safety Analysis

Summary: NPP Y performed a “simplified assessment” for estimation of the seismic capacity of the SSCs required to perform safety functions. Using the MIN-MAX methodology, NPP Y concluded that the NPP has “significant safety margin”. Accordingly, the plant HCLPF was estimated as 0.15g. This gives a ratio of (1.5g/0.76g) = 1.97. This is almost 2 x DBE of the plant or 1.5 x DBE recommended by IAEA.

Regarding the criteria on **Desired outcomes of seismic safety analysis,** the stress test report meets all three options, but one will be sufficient.

Regarding the evaluation of the safety margin and cliff-edge effects, the national regulator identifies in pgs. 21-22 of the *Post Fukushima Stress Test of the NPPX X* included in Ref. [1] several shortcomings which require additional work to fix them:

- Estimation of PGA above which the loss of fundamental safety functions or severe damage to the fuel (in vessel or in fuel storage) becomes unavoidable.
- Estimation of PGA that would result in loss of integrity of the reactor containment.
- Examination of the possibility of external floods caused by an earthquake and potential impacts on the safety of the plants.
- Thorough seismic margin assessment in order to ensure that the HCLPF of supporting structures is equal or higher than the HCLPF of the systems and equipment they support.
- Evaluation of the PGA that would result in a loss of fundamental safety functions or loss of integrity of the reactor containment.

NPP X committed to perform a detailed SMA or SPSA to reduce the uncertainty margins on seismic hazard, item S3, pg. 93 of Ref. [1].
Overall EHC for adequacy of protection of NPP against Seismic Hazard (safety analysis) are “Partly met”.

References

[1] National report on the Post-Fukushima Stress Test for the NPP X, December 2011
[2] ENSREG, EU stress tests specifications, May 13, 2011

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Implementation of follow-up actions from safety analysis for current seismic hazard

Review topics	Fully met	Partly met	Not met	Remarks
1. Physical design modifications for detection and/or mitigation of consequences of the current seismic hazard on plant safety	√			• Physical design modifications coping with the impact of BDBE upon the SSCs performing safety functions have been proposed, implemented or scheduled for implementation, para 7.1, page 89, and 7.3.1, pg. 87-91 of Ref. [1].
2. Procedures for mitigating the consequences of the impact of current seismic hazard on plant safety	√ √			• Procedures for mitigating the potential consequences of the impact of BDBE upon those SSCs performing safety functions have been proposed for implementation, para 7.3.2, pg. 92 of Ref. [1]. They reflect the findings and recommendations from Safety Analysis reports and the guidance provided in Ref. [3]. • NPP X committed to perform additional studies to reduce the uncertainty of the seismic margins by performing a Seismic Margin Assessment (SMA) or a Seismic Probabilistic Safety Assessment (SPSA)., item S3, pg. 93 of Ref. [1].
3. Regulatory review	√			• Physical design modifications and procedures proposed by NPP X for mitigation the potential consequences of the impact of BDBE have been reviewed, enhanced and accepted by the National regulator, para 2.2, pgs. 11-12 of *Post Fukushima Stress Test of NPP X* included in Ref. [1].

Note: For stress report evaluation, the current seismic hazard is Beyond Design Basis Earthquake (BDBE)

Note.

- BDBE – Beyond Design Basis Earthquake

Summary: the National regulator has reviewed the NPP X stress test report and found it follows the recommendations of ENSREG, Ref. [2]. Accordingly, the stress test report, Ref.[1] has been accepted with additional recommendations for further improvements of the NPP safety against seismic hazard at site. *Overall EHC for adequacy of protection of NPP against BDBE (safety analysis) are "Partly met".*

References:

[1], National report on the Post-Fukushima Stress Test for the NPP X, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011 [3] WANO, SOER (Significant Operating Experience Report) 2011-2.

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Current site Flood Characteristics and Flood Hazard Analysis

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√ √			• The analytical methods, safety standards and information used for flood hazard analysis are up to date (for the time when the stress test report Ref. [1] was drafted) and valid. They followed the mandatory Dutch regulations, and ENSREG (Ref. [2]) and IAEA (Ref. [3]) recommendations on this topic, para 3.1.1, pgs. 43-45 of Ref. [1]. • The analysis and methods take account of the site characteristics and relevant international practice available at the time when the stress test report was drafted.
2. Flood Site Characteristics and	 √	 √	 √ √ √	• Site flood characteristics for design review: o height of the water not provided o the height and period of the waves not provided o the warning time for the flood is provided. The procedure S-VF-01 is initiated by a water level of 3.05 m + NAP or a storm warning, para 3.1.3, pg. 44 of Ref. [1]. o the duration of the flood and the flow conditions not provided • Up to date information on the site flood characteristics vs. design provided and accounted for in Flood Hazard Analysis. (Potential of tsunami flood is not addressed).
3. Flood Hazard analysis (DFHA method. OR PFHA method.)		 √		**Deterministic Flood Hazard Analysis (DFHA)** • Output of DFHA checking: o DBF parameters: ▪ flow not provided ▪ elevation = 7.3 meter + NAP due to several back-fitting measures carried out during the operation. It corresponds to a return period of 1,000,000 years, para 3.1, pg. 43 of Ref. [1]. o PMF parameters: ▪ flow not provided ▪ elevation not provided **Probabilistic Flood Hazard Analysis (PFHA)** not applied.
4. Regulatory review	√			The site flood characteristics and the results of flood hazard analysis have been reviewed and accepted by the National regulator as “sufficient and plausible” with recommendation for update, para 2.3, pg. 12 of *Post Fukushima Stress Test of the NPP X* included in Ref. [1].

DBF = Design Basis Flood

PMF = Probable Maximum Flood

NAP = The ‘sea level’ is defined by the height of the calm sea surface in relation to a horizontal standard level. The Normal Amsterdam Water Level (NAP in Dutch) is used as this standard level.

DFHA = Deterministic Flood Hazard Analysis

PFHA = Probabilistic Flood Hazard Analysis

Summary: The Deterministic Flood Hazard Analysis (DFHA) has been used for characterization of the external flood hazard at site. The DBF for the NPP was established as (7.3 meter + NAP) due to several back-fitting measures carried out during the operation. It corresponds to a return period of 1,000,000 years. **The absence of tsunami flood hazard assessment is not justified.** *Overall, EHC for current Site Flood Characteristics and Flood Hazard analysis are “Partly met”*.

References:

[1] National report on the Post-Fukushima Stress Test for the NPP X Nuclear Power Plant, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards (draft)

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Adequacy of protection against current flood hazard (safety analysis)

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√ √	 √		• The analytical methods, safety standards and information used for flood hazard safety analysis are up to date (at the time the stress test report, Ref. [1], has been drafted) and valid. • The analysis and/or methods take account of the current plant design, existing site characteristics, condition of SSCs important to safety at present (*but **not** for their condition predicted for the end of LTO)* and relevant international practice such as ENSREG (Ref. [2]) and IAEA (Ref. [3]) on this topic. • Ref. [1] takes into consideration the changes in plant design and the potential for floods.
2. Desired outcomes of flood safety analysis	√ √ √			the frequency of occurrence and/or the consequences of the flood hazard are sufficiently low so that either no specific protective measures are necessary; additional preventive and mitigating measures in place are in place or planned; or deficiencies in the preventive and mitigating measures are properly identified.
3. Safety Analysis applicable to the current flood hazard (DFSA method. OR PFSA method)		 √ √ √ √	 √ √ √ √ √ √ √	**Deterministic Flood Safety Analysis (DFSA)** • Below are the elements checked for a traditional DFSA ○ RLF selection not provided; ○ Selection of the assessment team is implicit; ○ Plant familiarization and data collection is implicit; ○ Selection of success path(s) not provided; ○ No info on Systems walk-down to review preliminary success path(s), select success path(s) and SSCs ; ○ No info on Flood capability walk-down; ○ No info on HCLPF calculations (SSCs and plant); ○ Peer Review is assumed; ○ Info on Enhancements is sufficient; and ○ Insufficient info on Documentation • A simplified DFSA method has been used for determination of the safety margins BDBF. A *“margin of 1 m exists above the DBF of (7.3 m + NAP), before the situation worsens considerably and prevention of core damage becomes difficult”,* para 3.2.1, pg. 4 of Ref. [1]. • The cliff-edge effects for Beyond Design Basis Flood (BDBF) have been identified, para 3.2.1, pg. 4 of Ref. [1]. **Probabilistic Flood safety analysis (PFSA)** not applied.
4. Robustness of the plant;	√			**Deterministic Flood Safety Analysis (DFSA**

Review topics	Fully met	Partly met	Not met	Remarks
(DFSA method. OR PFSA method.)	√			• The stress test report, Ref. [1], determined that there is sufficient safety margin (1m) against BDBF. Accordingly, no core damage could occur within this safety margin. • Cliff-edge effects for BDBF have been identified. **Probabilistic Flood safety analysis (PFSA)** not applied.
5. Regulatory review	√			• The adequacy protection of NPP against BDBF determined by safety analysis has been reviewed and accepted by the National regulator as "sufficient and plausible" with additional comments for improvement, para 2.3, pgs. 12 of *Post Fukushima Stress Test of the NPP X* included in Ref. [1].

Note. For stress report evaluation, the current flood hazard is Beyond Design Basis Flood (BDBF)

- High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant
- DBF = Design Basis Flood
- BDBF = Beyond Design Basis Flood
- DFSA = Deterministic Flood Safety Analysis
- FPSA = Flood Probabilistic Safety Analysis

Summary: NPP X performed a "simplified deterministic assessment" for estimation of the flood margins of the NPP X NPP. It concluded that there is sufficient safety margin (8.55m + DBF) against the DBF (7.3m + NAP).

Regarding the criteria on **Desired outcomes of flood safety analysis,** the stress test report meets all three options (meeting only one option will be sufficient)

The national regulatory body, reviewed and accepted that the information provided by NPP X is "sufficient and plausible".

Overall, EHC for adequacy of protection of NPP against Flood Hazard (safety analysis) are 'Partly met'.

References

[1] National report on the Post-Fukushima Stress Test for the NPP X, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards (draft)

NPP: X **Safety factor: HAZARD ANALYSIS**

Review area: Implementation of follow-up actions from safety analysis for current flood hazard

Review topics	Fully met	Partly met	Not met	Remarks
1. Physical design modifications for detection and/or mitigation of consequences of the impact of current flood hazard on plant safety	√			- Physical design modifications coping with the impact of BDBF upon the SSCs performing safety functions have been proposed, implemented or scheduled for implementation, para 7.3.1, pg. 91 of Ref. [1].
2. Procedures for mitigating the consequences of the impact of the current flood hazard on plant safety	√ √			- Procedures for mitigating the potential consequences of the impact of BDBF upon those SSCs performing safety functions have been proposed for implementation, para 7.3.2, pg.92 of Ref. [1]. They reflect the findings and recommendations from Safety Analysis reports and the guidance provided in Ref. [3]. - NPP X committed to perform studies to further increase the safety margins in case of flooding, item S2, pg. 93 of Ref. [1].
3. Regulatory review	√			- Physical design modifications and procedures proposed by NPP X for mitigation the potential consequences of the impact of BDBF have been reviewed, enhanced and accepted by the National regulator, para 2.3, pg. 12 of *Post Fukushima Stress Test of the NPP X* included in Ref. [1].

Note. For stress report evaluation, the current flood hazard is Beyond Design Basis Flood (BDBF)

Note.

- BDBF = Beyond Design Basis Flood

Summary: The national regulator has reviewed the NPP X stress test report and found it follows the recommendations of ENSREG, Ref. [2]. Accordingly, the stress test report, Ref.[1] has been accepted with additional recommendations for further improvements of the NPP safety against flood hazard at site. *Overall, EHC for implementation of follow-up actions from EH safety analysis for current flood hazard are 'Fully met'*.

References:

[1] National report on the Post-Fukushima Stress Test for the NPP X, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] WANO, SOER (Significant Operating Experience Report) 2011-2.

EXTERNAL HAZARDS REPORT CARD

NPP: X **Safety factor: EXTERNAL HAZARD ANALYSIS**

Review area	Fully met	Partly met	Not met	Remarks
1. Overall management of external hazards (seismic hazard + flood hazard**)**		√		- Documentation of seismic hazard analysis and seismic safety analysis is limited.
2. Current site seismic characteristics and seismic hazard analysis			√	- Information provided in stress test report, Ref. [1], on site characterization is outdated and insufficient.
3. Adequacy of protection against current seismic hazard (safety analysis)		√		- Information provided in Ref. [1] on how the deterministic safety analysis was performed is limited.
4. Implementation of follow-up actions from safety analysis for current seismic hazard	√			- Information provided on this review area is sufficiently detailed.
5. Current site flood characteristics and flood hazard analysis		√		- Information provided in the stress test report is limited.
6. Adequacy of protection against current flood hazard (safety analysis)		√		- Information provided in the stress test report is sufficient but some Flood EHC were not met.
7. Implementation of follow-up actions from safety analysis for current flood hazard	√			- Information provided in the stress test report is sufficient and detailed.

Summary: The national regulator has reviewed the NPP X stress test report and found it follows the recommendations of ENSREG, Ref. [2]. Accordingly, the stress test report, Ref.[1] has been accepted with additional recommendations for further improvements. The amount of information in the stress test report is not entirely adequate for checking compliance with the EHC.

Overall, the EHC are 'Partly met'.

Note. It has to be noted that the actual benchmark matrices on EH for NPP X took into consideration the information regarding the plant status by the end of 2011 when Ref. [1] was drafted. The EHC checks is based on the information provided in the stress report. It is assumed that at this time most of the findings have been resolved and recommendations have been implemented. It is expected that all EHC on seismic hazard have been fully met, however, for the purpose of this sample application of the proposed LTO benchmarking method, the EHC for seismic and flood hazards have been 'Partly met'.

Below is quoted the National regulator overall assessment of the stress test report for NPP X:

"The stress test report of NPP X *matches the quality of reports of similar plants in neighbouring countries. A clear and accountable timetable for the implementation of the measures is lacking. In that respect,* NPP X *can take example from some nuclear operators in neighbouring countries that propose a specific schedule for the implementation of the measures".*

The stress test is not over yet. The next step will be the peer reviews in the European context. Meanwhile national regulator will continue the review, including the judgement on necessity and planning of studies and measures, using the findings from other countries, adjusted standards, frameworks and the results of the international peer review".

References: [1] National report on the Post-Fukushima Stress Test for the NPP X, December 2011

APPENDIX 5 LTO BENCHMARK ASSESSMENT MATRICES FOR NPP Y

NPP: Y **Safety factor: AGEING**

Review area: Overall plant AMP

Review topic	Fully met	Partly met	Not met	Remarks
AMP organization				
1. AMP strategy, i.e. policy and principles for managing ageing	√			The overall Aging Management Program is established and documented in MD-5. It is based on U.S. NRC 10 CFR 54, Gall report and industry guideline NEI-95-10. Overall AMP is an umbrella program which ensures that all important SSCs are identified through the Screening and Scoping Process, reviewed by Aging Management Review Process and managed so that they fulfil, or are able to fulfil their intended functions until the end of NPP Y extended plant lifetime.
2. AMP organizational arrangements	√			MD-5 is an umbrella AMP plant programmatic document that defines organizational arrangements for the AM related activities. This document specifies responsibilities and actions to ensure that AM programs are implemented, regularly updated, and that all configuration control for AMP is maintained and properly managed. The overall AMP includes periodic AMP review and updating, and annual reporting to the regulatory body.
AMP methodology				
3. Data management in support of AMP	√			Technical Division is responsible for AMP implementation, including record keeping program results in a database but database description and procedures are not provided. MECL (Master Equipment Component List) is the main component database, which contains all AMP SSCs.
4. Screening process to identify SSCs for AM	√			The scoping and screening procedures address all 3 criteria specified in 10 CFR 54.4 (safety systems, non-safety systems and regulated events). SSCs that remain after applying the scoping & screening criteria are documented and stay in the scope of the AMR.
5. Ageing management review (AMR)	√			AMR is performed on those SSCs reflecting the SSCs type, material and environment using GALL report. Results of AMR reviews are documented.
6. Condition assessment process	√			In the practice used (based on generic GALL guidance which is materials and environments oriented), the actual current and predicted future condition of SSCs is taken into account in the development/verification of AMPs when addressing Operating experience.
7. Process for the development of SSC-specific AMPs	√			Development/verification of SSC-specific AMPs is based on a comparison of relevant programs and activities against GALL Report which provides standard guidelines for aging management of SSCs; and includes SSC-specific elements of an effective AMP. NPP Y AMPs fulfil the requirements of the 10 elements of GALL, Revision 2. A list of the programs is provided.

Summary. *Overall, the relevant AMC are 'Fully met'.*

Reference. Technical Review Report on the NPP Y Ageing Management Program, December 2017

NPP: |Y **Safety factor: AGEING**

Review area: Scope of AMP for LTO

AMC	Fully met	Partly met	Not met	Remarks
1. List of SSCs covered by AMP should be available.		√		The Reference has a list of AMR reports on Mechanical systems, Civil structures and Electrical commodity groups and a list of AM related plant programmes, but not a list of SSCs covered by AMP. *However, it is known that* NPP Y *has adequately identified the systems and components within the scope of long term operation.* MECL (Master Equipment Component List) is the main component database, which contains all important components, equipment and structures. All MECL components or structures that are part of AMP list are designated in the MECL database with AMP flag Y (Yes).
2. List should include (passive and long-lived SSCs that are not subject to replacement based on a qualified life or specified time period)				The scope of the NPP Y AMP is determined by the US NRC 10 CFR 54 rule. It addresses the aging management of passive, long-lived components, i.e. those that are not subject to routine maintenance or replacement. SSCs listed in the 1st column that are in the list of AMR reports are designated by √. SCs that can be identified from the list of AMPs in Table 2-1 and 2-2 of the Reference are designated x.
• reactor vessel		√		Only some RPV components identified.
• reactor coolant system pressure boundary	√			
• steam generators		√		
• pressurizer			√	Only SG tubes identified.
• piping and tanks (including buried piping and tanks)	√			
• pump casings			√	
• valve bodies			√	
• core shroud	√			
• component supports			√	Identified only as part of RPV internals.
• pressure retaining boundaries			√	
• heat exchangers			√	

AMC	Fully met	Partly met	Not met	Remarks
• ventilation ducts	√			Identified as part of Ventilation air system.
• containment	√			
• containment liner	√			
• electrical and mechanical penetrations			√	
• equipment hatches			√	
• seismic Category I structures			√	
• electrical cables and connections	√			
• cable trays			√	

Summary. The reference provides only a high level, mostly system based listing of in-scope SSCs and a number of SSCs listed in 1st Column are not explicitly identified. For the purpose of the LTO Benchmark example application, NPP Y *Scope of AMP for LTO 'AMC Partly met'*. However, it is known that NPP Y has adequately identified the systems and components within the scope of long term operation, in accordance with 10 CFR 54 rule.

Reference. Technical Review Report on the NPP Y Ageing Management Program, December 2017

NPP: Y **Safety factor: AGEING**

Review area: SSC-specific AMP - Electrical cables

AMC	Fully met	Partly met	Not met	Remarks
1. Scope of AMP based on understanding ageing	√			• Scoping and screening of electrical cables for inclusion in AMP was performed in accordance with US NRC 10 CFR 54. Detailed identification of cable characteristics and routing for each of the generic cable group/commodity (cables with similar materials, environment or operating conditions) was performed. • - The majority of cables, regardless their function (SR or NSR) were purchased as nuclear-qualified cables. Cables outside the Nuclear island are outside of scope of the AMP.
2.Preventive actions to minimize ageing degradation	√			Aging assessment of electrical cables is implemented in accordance with the procedure TD-2D Cable Aging Management Program. Measurement of parameters (temperature, radiation, moisture, vibration, etc.) in building with identification of local adverse ambient – “Hot Spot”. All severe thermal “Hot Spots” identified during the assessment campaign (periodical visual inspections of all buildings) were subsequently eliminated. Approx. 60 cables were rerouted and 20 were replaced due to jacket damage/cracks.
3.Detection of ageing effects	√			The jacket properties are used to determine the condition of the primary cable insulation. Visual inspection and Indenter Modulus testing of cable jackets is the leading indicator of primary insulation condition. If any inacceptable conditions are found, the cable is tested.
4.Monitoring and trending of ageing effects	√			Electrical tests of cables in adverse environment are conducted for each cable with identified degradation mechanism. Aging Management Software Support Platform (COMSY) is used to evaluate all environmental data, cable measurements and to calculate residual life of cable.
5. Mitigating ageing effects	√			The acceptance criteria for testing of cables are categorized in 3 groups: “Good”, “Study” and “Action required”. In the “Action required” range, immediate actions are taken including additional testing, repairing, rerouting or replacing a cable.
6. Acceptance criteria	√			The acceptance criteria are established in accordance with GALL and other applicable standards for any given inspection method. E.g. Insulation resistance acceptance criteria in accordance with ANSI/NETA ATS-2009, Infrared thermography acc. to ASNT Recommended Practice No. SNT-TC-1A, Indenter Modulus for cables in adverse environment (temperature and radiation) – acceptance criteria are in accordance with HS-2011-02-T-IM.
7. Corrective actions	√			Repair or replace damaged cables or to eliminate and prevent aging effects.
8. Opex and R&D results feedback	√			Evaluation of applicable OE is undertaken regularly and all the findings related with aging are addressed within the Corrective Action Program. If applicable, an evaluation is undertaken and results are incorporated in TD-2D program. There is coop. with R&D organizations.

AMC	Fully met	Partly met	Not met	Remarks
9. Quality management	√			"Quality Assurance Plan", fulfils the requirements of 10 CFR 50. Confirmation process is used to verify effective implementation of corrective actions.

Summary. *AMP for electrical cables is consistent with GALL and the relevant AMC are 'Fully met'.*

Reference. Technical Review Report on the NPP Y Ageing Management Program, December 2017

NPP: Y **Safety factor: AGEING**

Review area: SSC-specific AMP - Buried steel piping and tanks

AMC	Fully met	Partly met	Not met	Remarks
1. Scope of AMP based on understanding ageing	√			Buried piping and tanks program includes each underground pipeline or tank that is classified as code class/safety-related or contains hazardous materials and is constructed from a material susceptible to degradation. AMP for concealed components: Essential Service Water System, Diesel Generator Oil System and Fire Protection System.
2.Preventive actions to minimize ageing degradation	√			Surveillance, cathodic protection, maintaining integrity of coating and wrapping.
3.Detection of ageing effects	√			Measuring coating conductance, visual examination. Ultrasonic or magnetic flux leakage methods are used to determine pipeline or tank wall thickness and the presence of corrosion. Fire protection system leakage is monitored by the frequency of starts of the jockey pump.
4.Monitoring and trending of ageing effects	√			Coating conductance vs. time or current requirement vs. time provides indication of coating condition when compared to predetermined values; annual frequency. Opportunistic visual examination when component exposed/ excavated.
5. Mitigating ageing effects	√			Protective coating and cathodic protection systems.
6. Acceptance criteria	√			In accordance with GALL, there should be no evidence of coating degradation or that the type and extent of coating degradation should be insignificant. If degradation detected, wall thickness in the affected area shall be determined to ensure minimum wall thickness is maintained.
7. Corrective actions	√			In accordance with reg. requirements. No relevant indications which would require corrective actions have been found. All inspected pipelines and tanks show very good condition.
8. Op-ex and R&D results feedback	√			Addressed by generic Op-ex program. There is coop. with R&D organizations.
9. Quality management	√			"Quality Assurance Plan", fulfils the requirements of 10 CFR 50. Confirmation process is used to verify effective implementation of corrective actions.

<u>Summary</u>. *AMP for Buried steel piping and tanks is consistent with GALL and the relevant AMC are 'Fully met'.*

<u>Reference</u>. Technical Review Report on NPP Y Ageing Management Program, December 2017

Safety Factor AGEING Report Card

NPP: Y **Safety Factor: AGEING**

Review area	AMC fully met	AMC partly met	AMC not met	Remarks
Overall plant AMP: 1. AMP strategy 2. AMP org. arrangements	√			Overall plant AMP is based on U.S. NRC 10 CFR 54, Gall report and industry guideline NEI-95-10, and meets the relevant AMC.
SSC-specific AMPs	√			AMP for Electrical cables and AMP for Concealed pipework, as described in the reference, meet the AMC.
AMP scope for LTO		√		The reference provides only a high level, mostly system based listing of in-scope SSCs and a number of SSCs listed in 1st Column are not explicitly identified. For the purpose of the LTO Benchmark example application, 'AMC Partly met' in Scope of AMP for LTO. *However, it is known that* NPP Y *has adequately identified the systems and components within the scope of LTO, in accordance with 10 CFR 54 rule and IAEA guidelines.*
TLAA revalidation	N/A			N/A – not in the scope of the Topical Peer Review.

<u>Summary.</u> *Scope*/topics of the Topical Peer Review do not match the AMC scope/review areas for Safety Factor Ageing. Taking this into account, NPP Y *Ageing Management fully meets applicable AMCs - Overall, the AMC for* NPP Y *Ageing Management are 'Fully met'.*

<u>Reference</u>. Technical Review Report on NPP Y Ageing Management Program, December 2017

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Overall Management of external hazards (seismic hazard + flood hazard)

Review topics	Fully met	Partly met	Not met	Remarks
1. EH management program/ function	√ √ √	√		• The Stress Test report, ref. [1] below, addresses those EHs identified by the Fukushima accident as a potential challenge to the safety of the plant. However, it is not clear how the report feeds into the EHC criterion requiring a dedicated program that provides for an effective management of EHs throughout NPP service life. • The Stress Test report is following the ENSREG's guidance provided in stress tests specifications, Ref. [2] below, and US NRC regulations and standards. ENSREG requirements are consistent with IAEA recommendations on this topic (Ref. [3]). • The stress report, Ref. [1], mentions that the experience with EHs and operating practices at NPPs is accounted for and implemented in the design and operating procedures of the NPP. • Experience from managing actual external events are being used/ are being developed to improve existing procedures at the plant. This is highlighted in several places throughout the stress the report.
2. Relevant external hazards	√ √ √ √ √ √ √ √ √ √ √ √			• A list of relevant external hazards that may affect plant safety has been established and reviewed for completeness following the guidance provided in Ref. [2]. • The following external hazards have been considered in stress report: ○ seismic hazard; ○ external flood; ○ maximum and minimum water temperatures of the River Sava; ○ extremely high and low air temperatures; ○ extremely high wind; ○ formation of ice; ○ heavy rainfall; ○ heavy snowfall; ○ external fire; ○ aircraft crashes, external missiles (not an ENSREG requirement);
3. Documentation	√			Documentation of EH analysis and safety analysis was provided as follows: • comprehensive information on EH analysis process and safety analysis (deterministic or probabilistic) • detailed background material for EH analysis and safety analysis

Review topics	Fully met	Partly met	Not met	Remarks
	√ √		√ √	• no detail on raw data (external hazards site records) and processed data (i.e. hazard curves), computer software and input and output files; • sufficient information on the reference documents used for calculation (codes, standards, safety guides, best industry practices, etc.); • no information on the results of intermediate calculations and sensitivity studies.
4. Regulatory review	√			• The national regulatory body, reviewed the stress report drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note

Summary: *EHC for Overall management of external hazards (seismic hazard + flood hazard) are, in general, “Fully met”.*

References

[1] National report on nuclear stress tests, Final Report, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards (draft)

NPP Y: Safety factor: HAZARD ANALYSIS

Review area: Current Site Seismic Characteristics and Seismic Hazard Analysis

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√ √			• The analytical methods, safety standards and information used for the seismic hazard analysis are up to date (for the time when the stress test report Ref. [1] was drafted) are up to date and valid. • The analysis and/or methods take account of the site characteristics and relevant international practice.
2. Site Seismic Characteristics	√ √ √ √ √ √			• Sufficient information is provided on the following site seismic characteristics for design review, chapter 2.1, pgs. 17-39: • evaluation of geological and geotechnical aspects at site area; ○ information on pre-historical, historical and instrumentally recorded earthquakes in the region is collected, documented and used for determination of the seismic hazard. ○ investigation for potential for surface faulting at site. ○ consistency of selected ground motion attenuation relations with the tectonic features of the region. • adequacy of uncertainty analysis of seismic sources parameters, ground motion models and site response for evaluation of site seismic hazard. • Sufficient up to date information on the site seismic characteristics vs. design is provided and accounted for in Seismic Hazard Analysis (see item 3 below).
3. Seismic Hazard Analysis (DSHA method. OR PSHA method.)	√ √			**Deterministic seismic hazard analysis (DSHA)** The design basis earthquake (DBE) was established using the Ref. [2], Accordingly, the DBE was defined as response spectrum scaled to 0.3 g PGA with no direct relation to a specific return period. The response spectrum was conservatively applied at the foundation level of the NPP on both horizontal and vertical directions. A second design earthquake, Operating Basis Earthquake (OBE), was defined with PGA 0.15g. **Probabilistic Seismic Hazard Analysis (SPHA)** Subsequent Probabilistic Seismic Hazard Analyses (PSHA) have been performed in 1994 and 2002-2004 due to increased concerns regarding the possibility of stronger earthquakes with a return period of 10,000 years. The PGA in free field for a return period of 10,000 years resulted in a 0.56g. The corresponding PGA value at the foundation level resulted from a deconvolution process was determined as 0.38g. A

Review topics	Fully met	Partly met	Not met	Remarks
	      √ √ 		 √ √ √ √ √	comparison between the peaks in the floor response spectra between original response spectra and the new response spectra are quite close. "This finding suggested that the NPP Y can accommodate a ground motion of much higher intensity", para 2.1.1.3, pg. 19 of Ref. [1]. The following elements of SPHA have been checked and found sufficiently detailed: • evaluation of the seismo-tectonic model for the site region in terms of the defined seismic sources, including uncertainty in their boundaries and dimensions; • evaluation for each seismic source of the maximum potential magnitude, earthquake occurrence probability and the type of magnitude–frequency relationship, together with the uncertainty associated with each evaluation; The items below have been developed and used by NPP Y in SPHA studies but not included in Ref. [1] • selection of the attenuation relationships for the site region, and assessment of the ground motion uncertainty. • derivation of hazard curves for different confidence level; • uniform hazard response spectrum site-specific response spectra at free field or any specified condition from the hazard. • a family of seismic uniform hazard curves has been developed for various confidence levels.
4. Regulatory review	√			National regulator reviewed the site seismic hazard characteristics and the results of seismic hazard analysis drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note

- hazard curve = a representation of the annual frequency of exceeding a given parameter of the external hazard.
- UHS = uniform hazard response spectrum is the response spectrum with an equal probability of exceedance for each of its spectral ordinates.
- PGA = peak ground acceleration
- DBE = Design Basis Earthquake

- SSE = Safe Shutdown Earthquake. In American terminology, Safe Shutdown Earthquake (SSE) is equivalent to DBE.
- OBE = Operational Basis Earthquake
- DSHA = Deterministic Seismic Hazard Analysis
- PSHA = Probabilistic Seismic Hazard Analysis

Summary: The site characterization with respect to the seismic hazard is sufficiently detailed. Some information has been developed and used in determination of the DBE but not included in stress report.

In the case of NPP Y, the DBE was originally established on deterministic basis using DSHA methodology. The value of DBE (SSE) was established as a ground seismic response spectrum with PGA of 0.3g following the guidance provided in US NRC Regulatory Guide, Ref. [2]. Conservatively, it was applied on horizontal and vertical directions at the foundation level of the NPP.

Later on, NPP Y decided to use the probabilistic methodology, PSHA, to define more accurately the DBE (SSE) and its associated ground response spectrum. Several SPHA studies were performed in 1994 and 2002-2004. The latter resulted in a free field seismic ground response spectrum with PGA of 0.56g (practically 2xSSE) for a return period of 10,000 years. However, the PGA of the seismic response spectrum at the foundation level of NPP was 0.38g. Additional work suggested that the NPP Y can accommodate the new ground motion.

EHC for current Seismic Site Characteristics and Seismic EH analysis are, in general, 'Fully met'.

References

[1] National report on nuclear stress tests, Final Report, December 2011

[2] USNRC, Regulatory Guide 1.60 "Design response spectra for seismic design of nuclear power plants", rev. 1, 1973

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Adequacy of protection against current seismic hazard (safety analysis)

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√ √ √			• The analytical methods, safety standards and information used for the seismic hazard safety analysis are up to date (for the time when the stress test report Ref. [1] was drafted) and valid. • The analysis and methods take account of the current plant design, seismic site characteristics, condition of SSCs important to safety at present (but not for their condition predicted for the end of LTO) and relevant international practice. • Changes in plant design, the potential for earthquakes have been considered.
2. Desired outcomes of seismic safety analysis	√ √			• frequency of occurrence and/or the consequences of the seismic hazard are sufficiently low so that either no specific protective measures are necessary, or • the preventive and mitigating measures in place are adequate, or • deficiencies in the preventive and mitigating measures are properly identified.
3. Seismic Safety Analysis applicable to the current seismic hazard (DSSA method. OR SPSA method.)	√ √ √ √ √ √ √ √ √			**Deterministic Seismic Safety Analysis (DSSA)** (Not applied) OR **Seismic Probabilistic Safety Analysis (SPSA)** • Seismic PSA elements reviewed are properly addressed: ○ Seismic Hazard Analysis; ○ Data collection and plant familiarization; ○ Structural Response Analysis including SSI or Equipment Structure Interaction; ○ Seismic Fragility Evaluation; ○ Systems/Accident Sequence Analysis; ○ Risk Quantification; ○ Peer review Requirements; ○ Documentation Requirements; • Cliff-edge effects have been identified for the current seismic hazard (see item below)

Review topics	Fully met	Partly met	Not met	Remarks
4. Robustness of the plant; (DSSA method. OR SPSA method)	 √ √ √		 √	**Deterministic Seismic Safety Analysis (DSSA)** (not applied) OR **Seismic Probabilistic Safety Analysis (SPSA)** • Core Damage could occur for PGA ≥ 0.8g. Cliff-edge effect expected above this value. • Large Early Radioactive Release could occur for PGA ≥ 1.0g. Cliff-edge effect expected above this value. • Spent Fuel failure (fuel uncovered) could occur for PGA ≥ 0.9g. Cliff-edge effect expected above this value. • **No values** provided for seismic induced Core Damage Frequency or Large Early Release Frequency
5. Regulatory review	√			• National regulator reviewed the site seismic hazard characteristics and the results of seismic hazard analysis drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note. For stress report evaluation, the current seismic hazard is Beyond Design Basis Earthquake (BDBE)

Note:

- High Confidence of Low Probability (HCLPF) = site characteristic level at which there is a high confidence (about 95%) of a low (5%) probability of failure of a component or the plant
- PGA = peak ground acceleration
- DBE = Design Basis Earthquake
- BDBE = Beyond Design Basis Earthquake
- CD= core damage
- LERR = Large Early Radioactive Release
- DSSA = Deterministic Seismic Safety Analysis
- SPSA = Seismic Probabilistic Safety Analysis

Summary: NPP Y performed a comprehensive SPSA for estimation of plant robustness against the current seismic hazard. The conclusion of the stress test report accepted by the regulatory body is that NPP Y can accommodate a ground motion of much higher intensity than the one the plant was originally designed for.
The stress test report shows that no core damage would occur for a free field seismic response spectrum with PGA of 0.8g. This corresponds to an earthquake with a return period of 50,000 years. The occurrence of such event is considered by national regulator as very rare.
Other significant values presented in the stress report:

- the PGA at which LERR could occur is 1.0g,
- the PGA at which the fuel in spent fuel pool could be uncovered is 0.9g

No information has been provided for the core damage frequency and large early radioactive release frequency, the safety goals that are typical outputs of a SPSA.

EHC for the adequacy of protection of NPP against seismic EH (safety analysis) have been, in general, 'Fully met'

References

[1] National report on nuclear stress tests, Final Report, December 2011
[2] USNRC, Regulatory Guide 1.60 "Design response spectra for seismic design of nuclear power plants", rev. 1, 1973

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Implementation of follow-up actions from safety analysis for current seismic hazard

Review topics	Fully met	Partly met	Not met	Remarks
1. Physical design modifications for detection and/or mitigation of consequences of the impact of current seismic hazard on plant safety	√			- Physical design modifications reflect the findings and recommendations from Safety Analysis report.
2. Procedures for mitigating the consequences of the impact current seismic hazard on plant safety	√			- Procedures for mitigating the potential consequences of the impact of the current external hazard reflect the findings and recommendations from Safety Analysis report.
3. Regulatory review	√			- National regulator reviewed the site seismic hazard characteristics and the results of seismic hazard analysis drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note. For stress report evaluation, the current seismic hazard is Beyond Design Basis Earthquake (BDBE)

Note

- BDBE – Beyond Design Basis Earthquake

Summary: The stress test report has been accepted and released by national regulator. It follows the ENSREG , IAEA and the relevant international requirements and recommendations on the topic of seismic hazard. *EHC for Implementation of follow-up actions from safety analysis for current seismic hazard are “Fully met”.*

References:

[1] National report on nuclear stress tests, Final Report, December 2011

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Current site Flood Characteristics and Flood Hazard Analysis

Review topics	**Fully met**	**Partly met**	**Not met**	**Remarks**
1. Attributes of analytical methods used	√ √			- The analytical methods, safety standards and information used for the flood hazard analysis were up to date and valid at the time the stress test report was drafted. - The analysis and/or methods take account of the site characteristics and relevant international practice.
2. Flood Site Characteristics	√ √			- Sufficient information is provided on the following site flood characteristics for design review: height of the water, the height and period of the waves, the warning time for the flood, the duration of the flood and the flow conditions have been included in stress test report, Ref. [1] - Sufficient up to date information on the site flood characteristics vs. design is provided and accounted for in Flood Hazard Analysis.

3. Flood Hazard Analysis
 	
(DFHA
OR
PFHA)
	
	√
	
	√
	
	√
	
	√
	√
	
	√
	
	√
				Deterministic Flood Hazard Analysis (DFHA)
				The following DFHA outputs have been reviewed:
				- evaluation of maximized hypothetical storms taking into account information, knowledge and results from the assessment of the meteorological hazards;
				- evaluation of the hypothetical storms at locations that produce high water effect at the site = 0.46m;
				- validation of the model selected for calculation of storm surge elevation taking into account site characteristics.
				- Evaluation of the maximum flood level.
				- Design Basis Flood parameters correspond to a return period of 10,000 years:
				    o design flow = 4790 m3/sec
				    o elevation = 155.35 m.a.A.s.l.
				- Probable Maximum Flood (PMF) parameters:
				    o - probable maximum flow = 6500 m3/sec
				    o - elevation = 155.89 m.a.A.s.l.
				- PMF for worst case of conservative scenarios (acc. Ref. [2])
				    o - maximum flow = 7081 m3/sec
				    o - elevation = 156.10 m.a.A.s.l.
				
				Probabilistic Flood Hazard Analysis (PFHA) not applied.

Review topics	Fully met	Partly met	Not met	Remarks
4. Regulatory review	√			- National regulator reviewed the site flood hazard characteristics and the results of flood hazard analysis drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note

- DBF = Design Basis Flood
- PMF = Probable Maximum Flood
- m.a.A.s.l.= meters above Adriatic sea level
- DFHA = Deterministic Flood Hazard Analysis
- PFHA = Probabilistic Flood Hazard Analysis

Summary: The Deterministic Flood Hazard Analysis (DFHA) has been used for characterization of the external flood hazard at site. The DBF parameters have been established as: elevation 155,35 m.a.A.s.l. and design flow = 4790m^3/sec. The parameters for PMF have been established as: elevation = 155.89 m.a.A.s.l and design flow = 6500 m^3/sec. *Overall, EHC for current Flood Site Characteristics and Flood Hazard analysis are 'Fully met'.*

References

[1] National report on nuclear stress tests, Final Report, December 2011

[2] ANSI/ANS-2.8 "Determining Design Basis Flooding at Power Reactor Sites", 1992.

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Adequacy of protection against current flood hazard (safety analysis)

Review topics	Fully met	Partly met	Not met	Remarks
1. Attributes of analytical methods used	√ √ √			- The analytical methods, safety standards and information used for the seismic hazard safety analysis were up to date and valid at the time the stress report, Ref. [1], has been drafted. - The analysis and/or methods take account of the current plant design, existing site characteristics, condition of SSCs important to safety at present (but not for their condition predicted for the end of LTO) and relevant international practice such as ENSREG (Ref. [2]) and IAEA (Ref. [3]) on this topic. - Ref. [1] takes into consideration the changes in plant design and the potential for floods.
2. Desired outcomes of Flood Safety Analysis	√			- The frequency of occurrence and/or the consequences of the flood hazard are sufficiently low so that either no specific protective measures are necessary, or - The preventive and mitigating measures in place are adequate or - Deficiencies in the preventive and mitigating measures are properly identified
3. Flood Safety Analysis applicable to current flood hazard (DFSA method. OR FPSA method.)	√ √ √ √ √ √ √ √ √ √ √			**Deterministic Flood Safety Analysis (DFSA)** The DFSA was used to perform the safety analysis for BDBF. The following elements have been checked: o RLF selection; o selection of the assessment team; o plant familiarization and data collection; o selection of success path(s); o systems walk-down to review preliminary success path(s), select success path(s) and SSCs; o flood capability walk-down; o HCLPF calculations (SSCs and plant) not provided o peer review is assumed o enhancements; and o documentation - Cliff-edge effect for BDBF has been identified (see item 4 below) **Flood Probabilistic Safety Analysis (FPSA)** not applied.

Review topics	Fully met	Partly met	Not met	Remarks
4. Robustness of the plant DFSA method. OR FPSA method.)	√		√	- In the worst scenario of PMF, the flow is 7081 m3/sec and the water elevation is 156.41 m.a.A.s.l. The elevation of the around the NPP is 157.10 m.a.A.s.l. There is safety margin of 157.10-156.41 = 0.9 m. - A cliff-edge effect has been identified for an extreme river flow flood of 11,000 m3/sec. This corresponds to a flood with a return period of 1,000,000 years
5. Regulatory review	√			- National regulator reviewed the site flood hazard characteristics and the results of flood hazard analysis drafted by NPP Y, provided comments which were incorporated in the report, and finally accepted it and released it under their LOGO.

Note. For stress report evaluation, the current flood hazard is Beyond Design Basis Flood (BDBF)

Note

- DBF = Design Basis Flood
- BDBF – Beyond Design Basis Flood
- m.a.A.s.l.= meters above Adriatic sea level
- PMF = Probable Maximum Flood
- RLF = Review Level Flood
- HCLP = High Confidence of Low Probability of Failure
- DFSA = Deterministic Flood Safety Analysis
- FPSA = Flood Probabilistic Safety Analysis

Summary: NPP Y performed a “deterministic assessment” for estimation of the flood margins. It concluded that there is sufficient safety margin against BDBF. *Overall, EHC for adequacy of protection against Flood Hazard (safety analysis) are ‘Fully met’.*

References

[1] National report on nuclear stress tests, Final Report, December 2011

NPP: Y **Safety factor: HAZARD ANALYSIS**

Review area: Implementation of follow-up actions from safety analysis for current flood hazard.

Review topics	Fully met	Partly met	Not met	Remarks
1. Physical design modifications for detection and/or mitigation of consequences of the impact of current flood hazard on plant safety	√			- Physical design modifications coping with the impact of BDBF upon the SSCs performing safety functions have been proposed, implemented or scheduled for implementation.
2. Procedures for mitigating the consequences of the impact of the current flood hazard on plant safety	√			- Procedures for mitigating the potential consequences of the impact of the current flood hazard reflect the findings and recommendations from Safety Analysis reports.
3. Regulatory review	√			- Physical design modifications and procedures for mitigation the potential consequences of the impact of the current flood hazard should be reviewed and accepted by the national regulatory body.

Note. For stress report evaluation, the current flood hazard is Beyond Design Basis Flood (BDBF)

Note

- BDBF = Beyond Design Basis Flood

Summary: *Overall, EHC for current Flood Site Characteristics and Flood Hazard analysis are 'Fully met'.*

References

[1] National report on nuclear stress tests, Final Report, December 2011

EXTERNAL HAZARDS REPORT CARD

NPP: Y **Safety factor: EXTERNAL HAZARD ANALYSIS**

Review area	Fully met	Partly met	Not met	Remarks
1. Overall management of external hazards (seismic hazard + flood hazard**)**		√ √		- It is not clear how the stress report, Ref. [1] feeds into the EHC criterion requiring a **dedicated program** that provides for an effective management of EHs throughout NPP service life. - Documentation of seismic hazard analysis and seismic safety analysis is limited.
2. Current site seismic characteristics and Seismic Hazard Analysis	√			- Information provided in stress report, Ref. [1], is up to date and sufficient.
3. Adequacy of protection against current seismic hazard (safety analysis)		√		- Information provided in Ref. [1] on how the deterministic safety analysis was performed is limited.
4. Implementation of follow-up actions from safety analysis for current seismic hazard	√			- Information provided on this review area is sufficient and detailed.
5. Current site flood characteristics and Flood Hazard Analysis	√			- Information provided in the stress report is sufficient and detailed.

Review area	Fully met	Partly met	Not met	Remarks
6. Adequacy of protection against current flood hazard (safety analysis)	√			- Information provided in the stress report is sufficient and detailed.
7. Implementation of follow-up actions from safety analysis for current flood hazard	√			- Information provided in the stress report is sufficient and detailed.

Summary: National regulator has reviewed, accepted and released the NPP Y stress test report. The stress test report follows the recommendations of ENSREG, Ref. [2], IAEA, Ref. [3] and relevant international codes, standards and best practices. The information provided in the stress test report is adequate for evaluating compliance with the EHC. *Overall, the EHC are 'Fully met'.*

References

[1] National report on nuclear stress tests, Final Report, December 2011

[2] ENSREG, EU stress tests specifications, May 13, 2011

[3] IAEA, A Methodology to Assess the Safety Vulnerabilities of Nuclear Power Plants against Site Specific Extreme Natural Hazards (draft)