European Commission

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

FINAL REPORT

EC Contract No. ENER/17/NUCL/SI2.769200

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PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

FINAL STUDY

VOLUME 1

SUMMARY REPORT

Revision R0

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Table of contents

List of acronyms

AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transient without Scram
Bel V	Belgian TSO
BDBA	Beyond Design Basis Accident
BMU	Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
CLI	Criteria for Limited Impact
CVR	Centrum výzkumu Řež, Czech TSO
DBA	Design Basis Accident
DEC	Design Extension Condition
DG	Directorate-General (of the European Commission)
DiD	Defence-in-Depth
EC	European Commission
EESC	European Economic and Social Committee
ENEF	European Nuclear Energy Forum
ENISS	European Nuclear Installations Safety Standards Initiative
ENSREG	European Nuclear Safety Regulators Group
ETSON	European Technical Safety Organisations Network
EU	European Union
EUR	European Utility Requirements
GPR	Groupe permanent d'experts pour les réacteurs nucléaires
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, German TSO
IAEA	International Atomic Energy Agency
IJS	Institute Jozef Stefan, Slovenian TSO
INSAG	International Nuclear Advisory Group
IRSN	Institut de Radioprotection et de Sûreté Nucléaire, French TSO
JRC	Joint Research Centre
LEI	Lithuanian Energy Institute, Lithuanian TSO
MTA EK	Hungarian Academy of Sciences Centre for Energy Research – MTA, Hungarian TSO
NPP	Nuclear Power Plant
NRA SR	Nuclear Regulatory Authority of the Slovak Republic
NSD	Nuclear Safety Directive (i.e. Council Directive 2009/71/Euratom amended by Council Directive 2014/87/Euratom)
PSR	Periodic Safety Review
RATEN ICN	Institute for Nuclear Research Pitesti (ICN) of "Technologies for Nuclear Energy (RATEN)", Romanian TSO
RHWG	Reactor Harmonisation Working Group
SRLs	Safety Reference Levels
STUK	Radiation and Nuclear Safety Authority, Finland

TSO	Technical Safety Organisation
VDNS	Vienna Declaration on Nuclear Safety
VTT	Technical Research Centre of Finland Ltd, Finnish TSO
VUJE	Slovak TSO
WBS	Work Breakdown Structure
WENRA	Western European Nuclear Regulators Association

List of tables

1 EXECUTIVE SUMMARY

Currently, the European Commission is promoting and facilitating an ambitious implementation of the requirements stipulated in Articles 8a to 8c of Council Directive 2014/87/Euratom. The objective of this study was to support Member States in achieving consistent practical implementation of the provisions set out in the new Council Directive 2014/87/Euratom[1]. Within this project, the focus was limited to selected nuclear installations, i.e. nuclear power plants and research reactors with a thermal power of 1 MW_{th} or more.

Within this project, the guidance and standards at the international and European level were reviewed and assessed to analyse if these documents provide sufficient information to support Member States in implementing the provisions of Articles 8a to 8c.

Besides international and European guidance documents, the national practice in EU Member States with nuclear power plants and research reactors was analysed using a technical survey.

In addition to the survey, detailed studies of safety improvements in selected Member States were performed for the following six countries: Finland, France, Germany, Hungary, Romania and Slovenia. These countries were selected to cover a broad range of reactor designs (Western design PWRs, Russian-type reactors as well as the CANDU design) and boundary conditions in the countries (operating NPPs, new-build projects, large fleet, phase-out).

The original objective of this study was to identify gaps, common approaches and areas of good practice and to highlight areas where it might be worthwhile to develop further guidance to reinforce consistent practical implementation of Articles 8a to 8c in the Member States. Due to concerns of the Member States' regulatory authorities expressed before and during the 1st workshop that this study could be misused to derive a ranking of national practices, the objective, originally formulated as identification of good practices and gaps, was slightly reworded to the identification of similarities and differences.

This study was performed in two phases with a workshop held shortly before the end of each phase.

[1] The Euratom Directives establish minimum requirements which shall be transposed by Member States into their national law. Member States may maintain or introduce more stringent measures.

One of the main observations of this study is that there is harmonisation amongst the Member States. However, a closer look at the effective and accurate implementation of Articles 8a to 8c in practice may also show potential differences between practices in the Member States. Consequently, future activities should aim at improving and fostering mutual understanding of these potential differences in the practices applied in the Member States. Given the wide range of technical issues addressed by the practical implementation of Articles 8a to 8c, such future activities need to be prioritised and focused on specific technical issues.

Finally, the following seven suggestions were derived from Tasks 1 to 3 of this study and are summarised below. Details can be found in Section 7 of this report.

- **Suggestion 1:** Increase mutual understanding of approaches to demonstrate avoidance of early releases by sharing the different approaches in the Member States.

- **Suggestion 2:** Increase mutual understanding of quantitative and qualitative approaches to demonstrate avoidance of large releases. This would first require a mutual understanding of approaches for radiological acceptance criteria for design extension conditions.

- **Suggestion 3:** Similarities and differences of these case-by-case analyses for application of the most up-to-date safety objectives could be a relevant scope for future work of ENSREG and WENRA. Exchange on this scope could in particular foster a mutual understanding of regulatory decision making.

  In particular, considering that the consortium supports pursuing the reinforcement of the DEC concept application, the scope could be first focusing on DECs. Within this context, on-going WENRA work related to the implementation of SRLs of issue F (Design Extension Conditions) at the plants appears to be an adequate tool.

- **Suggestion 4:** The consortium highlights the benefit of using a systematic approach for ensuring the continuous improvement of safety, based on:

  - identifying the sequences entailing the most critical safety issues, and

  - adequately reinforcing the requirements on the corresponding safety provisions – either to prevent these sequences or to limit their consequences.

  The consortium suggests that the European nuclear industry, TSOs, ENSREG and WENRA share their practice on this topic to contribute to the identification and development of new improvements.

- **Suggestion 5:** The consortium suggests to the regulatory authorities, TSOs and the European nuclear industry to consider the recent WENRA report on practical elimination as a reference document. If they would be involved in international cooperation on practical elimination, it would be better to focus the discussion on similarities in the application of the concept in order to identify further insights related to its justification. Discrepancies in the theoretical view of practical elimination have already been discussed in the past and do not seem to be a worthwhile topic for harmonisation anymore.

- **Suggestion 6:** Enhance mutual understanding of approaches for periodic safety reviews with particular focus on the implementation of the DEC approach in Member States.

- **Suggestion 7:** Promote discussions on the balance between increasing complexity in plant designs and independence of levels of defence-in-depth to avoid a potential decrease in nuclear safety. Practical examples might include side effects of modifications, shared equipment in multi-unit sites and I&C. Development of guidance may be needed to adequately support decision-making.

In addition, and in parallel to this study, WENRA's Reactor Harmonization Working Group (RHWG) was performing a pilot study on the practical implementation of two selected WENRA Safety Reference Levels (SRLs) related to the management of combustible gases and the supply of adequate power during design extension conditions [13]. RHWG's pilot study is a complementary project to this study, addressing selected technical requirements. Compared to this study, the scope is reduced whereas the level of detail is higher.

After the pilot study, the RHWG will continue with the project to address SRLs F4.8 to F4.18. As this study will cover 16 countries with 150 nuclear power plants and expected around 60 technical solutions to meet the requirements of the SRLs under consideration it is expected that this activity will continue for the next three to four years.

The study dealt with in this report was performed by ten participating TSOs of the ETSON network (namely Bel V, CVR, GRS, IJS, IRSN, LEI, MTA EK, RATEN, VTT and VUJE) on behalf of the European Commission under EC Contract No. ENER/17/NUCL/S12.769200.

The study comprises four volumes:

- Volume 1 – Summary Report
- Volume 2 – Study on International and European Guidance Documents
- Volume 3 – Assessment of Approaches and Methodologies Set in Place at National Levels for the Implementation of the EU Nuclear Safety Directive
- Volume 4 – Report on Selected Examples of Safety Upgrades Performed in Existing Nuclear Reactors

The four volumes present the results, findings and conclusions of the study. Volume 1 summarises the main results from Tasks 1 to 3 performed within this project. Volume 2 describes the analysis of international and European guidance documents. Volume 3 summarises the analysis of a survey performed in the Member States for NPPs in operation, under construction or in planning as well as for research reactors with a thermal power of 1 MW_{th} or more. The analyses of the detailed studies on safety upgrades performed for six selected Member States are described in Volume 4. In addition, Volume 3 and 4 include the collected data (answers received to the questionnaire in Task 2 and detailed studies in Task 3) as appendices.

2 SCOPE OF THE STUDY

After the nuclear accident at the Fukushima Daiichi nuclear power plant in 2011, activities have started or have been reinforced to further strengthen nuclear safety worldwide. At the EU level, Council Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations was amended by Council Directive 2014/87/Euratom of 8 July 2014. In particular, Articles 8a to 8c introduced the nuclear safety objective, requirements for the implementation of the nuclear safety objective and requirements for the initial assessment / periodic safety reviews. The relevant articles of Council Directive 2014/87/Euratom are quoted below:

> *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.

Article 8c

Initial assessment and periodic safety reviews

Member States shall ensure that the national framework requires that:

(a) any grant of a licence to construct a nuclear installation or operate a nuclear installation, is based upon an appropriate site and installation-specific assessment, comprising a nuclear safety demonstration with respect to the national nuclear safety requirements based on the objective set in Article 8a;

(b) the licence holder under the regulatory control of the competent regulatory authority, re-assesses systematically and regularly, at least every 10 years, the safety of the nuclear installation as laid down in Article 6(c). That safety reassessment aims at ensuring compliance with the current design basis and identifies further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards, using as a reference the objective set in Article 8a.

The amendment established “the objective of preventing accidents and, should an accident occur, mitigating its consequences and avoiding early radioactive releases and large radioactive releases”. While for nuclear reactors “for which a construction licence is granted for the first time after 14 August 2014”, the objective directly applies and shall be considered already in the plant design, this objective serves for existing nuclear reactors as a reference “for the timely implementation of reasonably practicable safety improvements”.

To achieve the nuclear safety objective, Article 8b of the Directive requires the application of defence-in-depth as well as the enhancement of an effective safety culture.

Before a construction or operating licence is granted, an appropriate site- and installation-specific assessment must be carried out, based in particular on the achievement of the nuclear safety objective. A reassessment of existing plants must be performed by a decennial periodic safety review as required in Article 8c.

With the adoption of Council Directive 2014/87/Euratom, EU Member States are confronted with the implementation of Articles 8a to 8c, especially those Member States either operating or constructing nuclear power plants or research reactors. A harmonised implementation of the three articles requires a common technical understanding of certain key terms applied in the Council Directive. In particular, the understanding of the terms “preventing”, “avoiding”, “early releases”, “large releases”, and the identification of “reasonably practicable” safety improvements requires further explanation and guidance. As those terms are also discussed within international and European organisations and working groups, some guidance documents are already available. Furthermore, the national nuclear regulatory authorities need to express their expectations towards the licence holders by adequate written information.

Currently, the European Commission is promoting and facilitating an ambitious implementation of the requirements stipulated in Articles 8a to 8c. The objective of this project was to support Member States in achieving consistent practical implementation of the provisions set out in the new Council Directive. Within this project, the focus was limited to selected nuclear installations, i.e. nuclear power plants and research reactors with a thermal power above 1 MW_{th}.

Within this project, the guidance and standards at the international and European level were reviewed. It was assessed if these documents provide sufficient information to support Member States in implementing the provisions of Articles 8a to 8c.

Besides international and European guidance documents, the national practice in EU Member States with nuclear power plants and research reactors was analysed by performing a technical survey. The survey was sent to the following 16 Member States with nuclear power plants / research reactors[2] in planning, under construction or in operation: Belgium, Bulgaria, Czech Republic, Finland, France, Germany, Hungary, Italy, Poland, Romania, Slovakia, Slovenia, Spain, Sweden, The Netherlands and the United Kingdom[3].

For six selected Member States, i.e. Finland, France, Germany, Hungary, Romania and Slovenia, detailed studies on safety improvements were performed, which cover different reactor designs and boundary conditions. In all cases, the studies were performed by the project team and consulted on with the competent national regulatory authority.

Findings of the project were presented during two technical workshops (1st workshop in July 2018, 2nd workshop in November 2019) and discussed with the participants from national regulatory authorities, industry, international organisations as well as representatives from civil society. The results of these discussions were considered by the project team and are reflected in the final results of this study.

[2] Research reactors with a thermal power of 1 MW_{th} or more

[3] The contribution of the following Member States by responding to the questionnaire is acknowledged: Belgium, Czech Republic, Finland, Germany, Hungary, The Netherlands, Slovakia, Slovenia, Spain, Sweden, United Kingdom.

Although the overall objective of the project was to contribute to identifying gaps, common approaches as well as areas of good practice and to emphasise the need for developing further guidance to achieve consistent practical implementation of the ambitious objectives in Articles 8a to 8c, discussions during the 1st workshop revealed that the European regulatory authorities had serious concerns that the study would result in a ranking of the practices amongst the Member States. Therefore, the focus was laid more on the identification of similarities and differences with the objective to achieve a better mutual understanding of national practices and to identify topics where further activities may be necessary to support Member States in implementing Articles 8a to 8c in practice.

The project was organised in four tasks:

- **Task 1:** “Review and assessment of international and European guidance documents”
- **Task 2:** “Assessment of approaches and methodologies set in place at national levels for the implementation of the EU safety directive”
- **Task 3:** “Performing a detailed study on the safety upgrades in existing reactors performed in selected Member States”
- **Task 4:** “Organisation of two workshops”

Chapters 3 to 6 summarise the main results of Tasks 1 to 4 of this study. Detailed results of Tasks 1 to 3 are provided in Volumes 2 to 4 of this study.

This study was performed in two phases with a workshop held shortly before the end of each phase.

- ***Phase 1 (January 2018 to September 2018)***
    - Performing the study on review and assessment of the international and European Guidance documents (**Task 1**);
    - Drafting and finalisation of a questionnaire to analyse the approaches and methodologies set in place at national levels for the implementation of the EU safety directive (**Task 2**);
    - Performing the detailed pilot study on safety upgrades in Slovenia (**Task 3**);
    - Organisation of the 1st workshop with the following topics: presenting the results of Task 1, presenting and discussion of the questionnaire for the survey of Task 2 and presenting the pilot study on safety upgrades in Slovenia (**Task 4**).

- ***Phase 2 (September 2018 to December 2019)***
    - Finalising study on international and European guidance documents (**Task 1**);
    - Analysis of the technical survey and derivation of areas for future activities (**Task 2**);
    - Performing detailed studies on safety upgrades in Finland, France, Germany, Hungary and Romania. Analysis of the six detailed studies and derivation of areas for future activities (**Task 3**);
    - Organisation of the 2nd workshop with the following topics: presenting the results of Task 2 and Task 3, discussing the suggestions for future activities derived from the analyses performed within Task 1, Task 2 and Task 3 (**Task 4**).

The two workshops were an opportunity to share the findings of the project with various stakeholders, such as representatives from competent regulatory authorities, industry, international organisations and civil society. The project benefits from the discussions and feedback from the two workshops to improve clarity on the conclusions but also to be sensitised to the concerns of the stakeholders.

3 REVIEW AND ASSESSMENT OF INTERNATIONAL AND EUROPEAN GUIDANCE DOCUMENTS

Council Directive 2014/87/Euratom introduces several terms, but without definition of the terms or a glossary. Thus, the meaning of the terms leaves room for interpretation by the Member States and consequently is a challenge regarding a harmonised approach in Europe. Within this study, a review of international and European documents was performed to analyse if sufficient guidance is provided to support Member States in implementing the objectives set out in Articles 8a to 8c. The review was strictly performed from a technical point of view rather than considering legal aspects of the terms under consideration. In particular, available documents were analysed regarding the interpretation of the following terms:

- ‘*to prevent*’,
- ‘*to mitigate*’,
- ‘*to avoid*’,
- ‘*early releases*’ and
- ‘*large releases*’.

Regarding the identification of potential safety improvements for existing reactors to meet the nuclear safety objectives, the following terms are discussed in this study:

- ‘*timely implementation*’;
- ‘*reasonably practicable*’.

The results presented here were elaborated within the framework of Task 1 “Review and assessment of international and European guidance documents” of the project.

Within this study, the review primarily focused on the information available from European organisations. Documents published by the Western European Nuclear Regulators Association (WENRA) and its Reactor Harmonisation Working Group (RHWG) were recognised as important documents representing the European consensus of the competent regulatory authorities of the Member States.

The following WENRA documents have been identified as relevant for this study:

- WENRA Safety Reference Levels for Existing Reactors [6];
- WENRA statement on “Safety of new NPP designs” [7];
- WENRA RHWG Report "Safety of new NPP designs" [8];
- WENRA Guidance "Article 8a of the EU Nuclear Safety Directive: “Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants” [9];

- WENRA-RHWG Guidance Document Issue T: Natural Hazards (Head Document) [10];
- WENRA-RHWG Guidance on Issue F (Design Extension Conditions) [11];
- RHWG – Position paper on Periodic Safety Reviews [12].

In addition, supplementary information was gathered from the European Utility Requirements (EUR) [3,4], a set of specifications for new nuclear power plants published by the European utilities, which are finally in charge of implementing the requirements in practice at their plants in the design phase, during construction or in operation. However, preference was given to the documents published by WENRA because these publications represent a common position of the European regulatory authorities whereas the EUR represent the position of the licence holders.

Further documents representing international high-level recommendations are the publications of IAEA's advisory body and the International Nuclear Safety Advisory Group (INSAG). INSAG publications do not represent a common view of IAEA Members States but express the views of renowned experts in the field of nuclear safety. Recommendations of INSAG are published by the IAEA. In the context of this project, the following INSAG reports were referred to:

- INSAG-10 "Defence in Depth in Nuclear Safety" [14];
- INSAG-12 "Basic Safety Principles for NPP" [15].

In practice, these documents have a direct impact on the development of IAEA safety standards.

The IAEA safety standards represent the global consensus with respect to nuclear safety. During the development of these standards, relevant INSAG reports and expertise from IAEA Member States were taken into account. In the framework of this project, the most relevant IAEA publications are

- IAEA Safety Standard Series SSR 2/1 Rev. 1 "Safety of Nuclear Power Plants: Design" [16];
- IAEA Safety Standard Series SSR 3 "Safety of Research Reactors" [17];
- IAEA Safety Standard Series SSG-25 "Periodic Safety Review for Nuclear Power Plants" [18];
- IAEA Safety Standard Series SSG-2 "Deterministic Safety Analysis for Nuclear Power Plants" [19] as well as DS491 "Deterministic Safety Analysis for Nuclear Power Plants" [21], now published as SSG-2 Rev. 1 "Deterministic Safety Analysis for Nuclear Power Plants" [20].

Furthermore, the IAEA Safety Glossary [22] and its updated draft from 2016[4] [23] were used.

Latest discussions on the further development of the defence-in-depth concept and the design of nuclear power plants are reported in IAEA TECDOC 1791 "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants" [24]. This TECDOC reflects only partially the European position as published by WENRA, and also discusses other approaches applied in IAEA Member States.

After analysing the identified international and European documents with respect to Articles 8a to 8c of Council Directive 2014/87/Euratom, a gap analysis was performed. The observations were categorised as follows:

- No guidance available – if no specific guidance can be found in the analysed documents or if only a few organisations address the issue under consideration.
- Ambiguous guidance – if the provided guidance differs amongst the analysed documents and leaves room for interpretation.
- Harmonised guidance – if the provided guidance is consistent in all analysed documents.

Based on the identified gaps, advice for technical areas requiring future work was proposed. The focus was placed on ensuring a harmonised application of the approaches and a common understanding of the definitions of terms in Europe.

[4] In 2019, the draft was published as “IAEA Safety Glossary: 2018 Edition”.

Table 1 summarises the main findings from the gap analysis.

Table 1: Summary of gap analysis performed in Task 1

Term	Category
'to avoid'	Ambiguous guidance
'to prevent'	Harmonised guidance
'to mitigate'	Harmonised guidance
'to control'	Harmonised guidance
'early release'	Harmonised guidance (qualitatively)
'large release'	Harmonised guidance (qualitatively)
'reasonably practicable'	No guidance available[5] (from technical point of view)
'timely implementation'	No guidance available[5] (from technical point of view)

3.1 Summary of findings

After reviewing international and European guidance documents, the following findings can be summarised:

- Article 8b of Council Directive 2014/87/Euratom repeats the objectives for the first four levels of defence-in-depth as described in INSAG-10.
- With respect to new nuclear power plants, the defence-in-depth concept addressed in Article 8b of Council Directive 2014/87/Euratom is less detailed than the defence-in-depth concept agreed within WENRA. However, it is fully in line with the expectations expressed in INSAG 10 and IAEA SSR 2/1. Council Directive 2014/87/Euratom requires that where defence-in-depth applies, it shall be applied to ensure that severe conditions are controlled, including prevention of accidents progression and mitigation of the consequences of severe accidents. In recital 17, the expectations with regard to the defence-in-depth concept is described explicitly referring to international standards and guidance and to WENRA. In addition, recital 15 expresses the expectation to take into account the latest international safety requirements. The authors of this study consider the defence-in-depth concept developed and published by WENRA (see [8])

[5] In June 2017, WENRA published the WENRA Guidance "Article 8a of the EU Nuclear Safety Directive: "Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants"" [9].

as the most advanced description of the defence-in-depth concept for new nuclear power plants and recommend this concept as international good practice.

- Sufficient guidance is available on the implementation of the defence-in-depth concept. It was emphasised that a common understanding of ‘prevention’ exists, and that ‘mitigation’ is used in the context of minimising or reducing radiological consequences in case of accident conditions.

- The term ‘to avoid’ is frequently used to express a radiological safety objective.

- The term ‘to control’ is frequently used to express that a certain plant state can be dealt with by safety-relevant features provided on the respective level of defence-in-depth. This term is internationally used in a partially harmonised manner.

- In the internationally available documents there is a common understanding that in the case of an ‘early release’, adequate off-site countermeasure will not be effective in due time. It can be concluded that no quantitative values are provided in either international or European documents to provide further guidance for the national regulatory authorities in implementing a harmonised meaning of this term.

  There is a common understanding that ‘large releases’ would require off-site countermeasures, which are not limited in area and time. Most internationally available documents do not further discuss this point. It can be concluded that no quantitative values are provided in either international or European documents to provide further guidance for the national regulatory authorities in implementing a harmonised meaning of this term. However, WENRA has published an overview of the acceptable off-site countermeasures as function of distance [8]. Furthermore, EUR [3, 4] describes an approach to determine quantitative radiological acceptance criteria taking intervention levels for emergency response countermeasures into account. However, it is important to emphasise that methods, models and parameters used for safety analysis and emergency preparedness are different. As regards doses to the public or level of contamination to foodstuff, the definition of harmonised quantitative radiological acceptance criteria linked to qualitative goals of WENRA Safety Objective O3 is difficult since the methods for analysing radiological consequences might be based on different national regulations and practices including calculation models and hypothesis.

  It was also recognised that some countries (e.g. Finland, Sweden) have defined quantitative radiological acceptance criteria. In Hungary, a new guideline is under publication on requirements regarding large and early releases [5]. In addition, the recently published IAEA Safety Standard SSG-2 Rev. 1 [20] recommends radiological acceptance criteria for all plant states in terms of effective dose, equivalent dose or dose rate to workers at the nuclear power plants, members of the public or the environment[6].

The study as well as the discussions during the 1st workshop showed that further exchange on the demonstration and on the assessment of avoiding 'large releases' as well as 'early release' would be desirable. Here, the objective should be to foster the mutual understanding of the different approaches in the Member States.

Further exchange of information and practices would be recommended on the necessary technical background information underlying the regulatory decision-making if a potential safety improvement is 'reasonably practicable' or not. In a similar manner, the approach to ensure the 'timely implementation' should be discussed from a technical point of view.

[6] In addition to radiological acceptance criteria, SSG-2 Rev.1 also recommends technical acceptance criteria.

3.2 Proposals for future activities

As discussed in the previous sections, the following topics were identified for future activities and are discussed below.

Acceptance criteria for large releases

In the aftermath of the accident at the Fukushima Daiichi NPP, discussions have been started on recommendations to protect the public in the event of a severe accident in Europe. In 2014, HERCA and WENRA published a common approach to harmonise the off-site protective actions during the early phase of a nuclear accident [25]. It was recognised that each European state has developed arrangements to protect the public in the event of a severe nuclear accident. However, due to the freedom to define intervention values and countermeasure on a national basis, sometimes significant differences exist. Especially in Europe, a nuclear accident could lead to cross-border issues. Thus, a necessity was seen to align protective actions along adjacent national borders to issue harmonised recommendations to the public on both sides of the border. In contrast to the field of emergency preparedness and response as addressed in the common HERCA / WENRA paper, the situation with respect to the safety demonstration for new reactors is somewhat different. Here, the applicant must demonstrate that in the case of a postulated core melt accident only accepted protective actions may be necessary. Radiological quantitative acceptance criteria are recommended in IAEA Safety Standard Series No. SSG-2 Rev. 1 [20] amongst other criteria. However, quantitative radiological acceptance criteria, in particular in terms of maximum activity released, have to be aligned to the chosen approach and assumptions made in the radiological consequence analysis. This would require a mutual understanding of the applied methodology of the radiological consequence analysis and how quantitative acceptance criteria have been defined by the regulatory authority.

Acceptance criteria for ‘early releases’

‘Early releases’ are commonly defined as releases which will not allow for a proper implementation of mitigative off-site countermeasures before a potential release occurs. For the safety demonstration, the site-specific boundary conditions for implementing off-site countermeasures (like e.g. sheltering, iodine prophylaxis, evacuation, etc.) must be taken into account as well as the installation-specific accident scenarios, in particular the timing of the accidental releases. In the safety demonstration, it should be demonstrated that an accidental release will not occur before the off-site countermeasures are effectively implemented.

‘reasonably practicable’ and ‘timely implementation’

The most recent guidance on this topic can be found in the WENRA Guidance “Article 8a of the EU Nuclear Safety Directive: ‘’Timely Implementation of reasonably Practicable Safety Improvements to Existing Nuclear Power Plants’’”. In this paper, it is clearly expressed that ‘reasonably practicable’ as well as ‘timely implementation’ is judged on a case-by-case basis.

It is proposed to discuss the matter in more detail. In general, ‘reasonably practicable’ is the result of a decision-making process involving the licence holder as well as the competent regulatory authority. This process should start with the safety objective to be achieved, a thorough investigation of the technical solutions available, the impact on the design of the existing facility, further implications of the implementation of the identified safety improvement and the final judgement of the competent regulatory authority. A process may be developed which ensures a harmonised decision-making process in the Member States.

The term ‘timely implementation’ first describes a requirement with respect to the attitude of the licence holder and the competent regulatory authority. It expresses the expectation that all involved parties will not unduly delay the implementation of a reasonably practicable safety improvement. It is therefore first of all a prerequisite for the safety culture of all parties involved. Secondly, it can be discussed from the technical point of view. Here, the licence holder may propose a time schedule for the implementation of a safety improvement, which can be reviewed and assessed by the competent regulatory authority. Based on the review and assessment of the proposed timeline, the timely implementation can be judged. After the licence holder and the competent regulatory authority have agreed on the time line, it is very important to follow up the activities to ensure the timely implementation in accordance with the proposed time schedule.

From the observations made within Task 1, the following three suggestions were proposed as well as presented during the 1st workshop and discussed with the participants.

- **Proposed suggestion 1.1:**
  “*Sharing the different approaches in the Member States would be desirable to foster a mutual understanding of the different approaches applied to demonstrate avoidance of “early releases*.”

- **Proposed suggestion 1.2:**
  “*Foster mutual understanding of demonstrating avoidance of “large releases” by applying different (quantitative/qualitative) approaches.*”

- **Proposed suggestion 1.3:**
  "*Sharing approaches to enhance mutual understanding of regulatory decision-making on 'reasonably practicable'.*"

Based on the discussions during the 2nd workshop and regarding the proposed suggestions from Task 2 and Task 3 all proposed suggestions were finalised and consolidated (see Chapter 7).

4 ASSESSMENT OF APPROACHES AND METHODOLOGIES SET IN PLACE AT NATIONAL LEVELS FOR THE IMPLEMENTATION OF THE EU SAFETY DIRECTIVE

The consortium has analysed the answers to the questionnaire provided by 11 Member States and used its experience as a collection of ETSON members when no answer was provided. On this basis, the consortium performed a technical evaluation of how the requirements of Articles 8a to 8c of Council Directive 2014/87/Euratom are interpreted and applied in practice in the EU Member States. As this section summarises this task, more detailed information can be found in Volume 3 “Assessment of approaches and methodologies set in place at national levels for the implementation of the EU Safety Directive – Report based on the questionnaire on national approaches” of this study.

The consortium devised its questionnaire structure and content to collect detailed information on the implementation of the relevant Articles 8a to 8c as efficiently as possible. It could be highlighted that Article 8a is mainly related to the topic “safety objectives for new reactors and for existing ones”, Article 8b(1) mainly to the topics “design and operational aspects of the defence-in-depth”, “probabilistic safety assessments, “design extension conditions without fuel melt”, “design extension conditions with fuel melt”, “design of the containment” or ”practical elimination”, Article 8b(2) mainly to the topic “safety culture and operational experience”, and Article 8c mainly to the topic “periodic safety review”.

Regarding Article 8a of Council Directive 2014/87/Euratom, the consortium described how achievement of WENRA safety objectives could be considered as a prerequisite to fulfil the general principle of the objective stated in Article 8a(1) of the Council Directive. From the received replies to the questionnaire it can be seen that those countries who have answered the questionnaire have systematically taken into account the relevant safety objectives for new nuclear power plant designs established by WENRA in their national regulatory frameworks along with other international standards,

- considering the most up-to-date safety objectives for any reactor, irrespective of whether it is in operation (in particular for the periodic safety review), under construction, or in the design phase;
- for operating reactors, applying these objectives on the basis of an analysis taking into account both the characteristics of the respective installation and the promotion of a continuous improvement of safety.

This leads to the first suggestion of the consortium:

- **Proposed suggestion 2.1:**

  "*Similarities and differences of these case-by-case analyses for application of the most up-to-date safety objectives could be a relevant scope for future work of ENSREG and WENRA. Exchange on this scope could in particular foster a mutual understanding of regulatory decision making.*

  Being aware that pursuing the reinforcement of the DEC concept application is an important issue, the scope of such an exchange amongst Member States should be prioritised to (selected) DECs as a first step. Within this context, on-going WENRA work related to the implementation of SRLs of issue F (Design Extension Conditions) at the installations may contribute to such an activity."

Indeed, when performing its evaluation related to Article 8b(1) of Council Directive 2014/87/Euratom, the consortium concluded that pursuing the reinforcement of the DEC concept must be supported.

Regarding this evaluation, it should first be reminded that defence-in-depth remains a fundamental concept that is systematically underlined in international literature as a basis for safety and considered as a "key concept" in the context of harmonised improved safety vis-à-vis Europe: answers to the questionnaire confirm a well-established concept both in the regulations and in actual practices, applied equally to new reactors and existing reactors. Moreover, many safety improvements had already been performed prior to the formal reinforcement of defence-in-depth that came along with Council Directive 2014/87/Euratom and the post Fukushima updates.

Henceforth, the consortium would like to highlight that the potential strengthening of defence-in-depth and associated added value should not be sought either through its basic application or through a too straightforward and never-ending undifferentiated expectation of reinforcement of each level. Within this context, the concept of "design extension conditions" (DECs) contributes to the deepening of the considerations related to reinforcement of defence-in-depth. This concept has been developed to reflect that the consideration of more challenging events than those considered in the design basis is an integral part of the safety approach. The DEC concept is a major step as its goal is a comprehensive consideration of events associated with safety objectives whose achievement is expected to be demonstrated: answers to the questionnaire showed that Member States have progressively sought more than a simple confidence regarding the resilience of reactors to challenging events but an actual demonstration that ambitious objectives are achieved.

The consortium highlights that the answers clearly show that significant results have already been achieved with regard to the application of the DEC concept, leading to many safety improvements, some of which are common to several Member States. As a consequence, and as mentioned above, the consortium supports pursuing the reinforcement of the DEC concept application, thus proposing to focus on this topic for its first suggestion. More details on this topic are provided in Volume 3 of this study.

Whilst the first suggestion of the consortium is related to the regulatory decision-making and proposes to use the feedback of past improvements, all the considerations detailed above, backed up with several examples extracted from the answers gathered, have led the consortium to formulate the second proposed suggestion, which is related to Article 8b(1) of Council Directive 2014/87/Euratom and which is applicable to the whole nuclear community when identifying and designing new improvements.

- **Proposed suggestion 2.2:**

  "*The consortium highlights the benefit of using a systematic approach for ensuring the continuous improvement of safety, based on:*

  - *identifying the sequences entailing the most critical safety issues, and*
  - *adequately reinforcing the requirements on the corresponding safety provisions – either to prevent these sequences or to limit their consequences.*

  The consortium suggests that the European nuclear industry, TSOs, ENSREG and WENRA share their practices on this topic to contribute to the identification and development of new improvements."

The third suggestion of the consortium is specific to a particular approach: practical elimination.

- **Proposed suggestion 2.3:**

  "*The consortium suggests to the regulatory authorities, TSOs and European nuclear industry to consider the recent WENRA paper on practical elimination as a reference document. If they would be involved in international cooperation on practical elimination, it might be better to focus the discussion on similarities in the technical application of the concept in order to identify further insights related to its justification. Although discrepancies in the theoretical notion of practical elimination exists, this is a topic that is of less importance than the aforementioned technical one.*"

Regarding Article 8b(2) of Council Directive 2014/87/Euratom and safety culture, all the answering countries agree that promoting and encouraging a strong safety culture is crucial to ensuring the safety of nuclear installations. However, the answers were at a very high level, since this topic is hard to apprehend through documents only, in particular through a questionnaire. Therefore, safety culture is not addressed in this report. To better understand how safety culture is promoted and encouraged in practice, some dedicated work would be necessary, involving case studies and on-site immersions.

Finally, regarding Article 8c of Council Directive 2014/87/Euratom, common expectations regarding PSRs are identified and are an essential element for ensuring a satisfactory safety of these installations throughout their lifetime through:

- compliance with design expectations;
- implementation of a continuous improvement approach.

Even if the wording may differ, answers reflect consideration of ageing issues, operational experience, most recent research results and developments in international standards.

The range of topics covered by PSRs may vary slightly from one country to another (physical protection/security, exports control, transport safety, radioactive waste, spent nuclear fuel, safeguards, radiation protection …). Nevertheless, the answers provided by the Member States safety authorities tend to show harmonisation, in particular WENRA SRLs requirements are adequately considered in PSRs.

5 PERFORMING A DETAILED STUDY ON THE SAFETY UPGRADES IN EXISTING REACTORS PERFORMED IN SELECTED MEMBER STATES

The main objective of Task 3 was to gain a deeper insight into the technical background of certain safety improvements implemented in selected Member States. The countries were selected to cover a wide range in terms of nuclear strategies and reactor designs in Europe. The following six countries were selected:

- **Finland**, existing reactors as well as new-build projects (EPR) and VVER in planning;
- **France**, largest fleet of NPPs, existing reactors as well as new-build projects (EPR);
- **Germany**, phase out of nuclear energy, PWRs and BWRs in operation;
- **Hungary**, new-build projects and existing reactors with life time extension of the VVER-type;
- **Romania**, the only country in Europe operating CANDU-type reactors;
- **Slovenia**, small country with only one NPP shared with Croatia, prolonged operation.

The detailed studies for the selected countries were analysed to propose suggestions for future activities. The analyses were aimed at gaining insights on the similarities and differences between national practices in the national implementation of specific safety upgrades in existing nuclear reactors. During the first phase of the project, the detailed study for Slovenia was performed as a pilot study. In the second phase, the five remaining detailed studies on safety improvements performed in Finland, France, Germany, Hungary and Romania were completed. The detailed studies were prepared by the national TSOs. The competent authorities of all six countries were consulted on the contents of the country reports.

One result of analysing the detailed studies was that the national reports of the six countries under consideration showed a fairly high degree of harmonisation in technical rationales for safety improvements of nuclear installations. The following main drivers for safety improvements have been identified:

- implementation of the defence-in-depth concept, including DECs;
- implementation of the WENRA SRLs published in 2014 [6];
- requirements for new reactors, as also applied to existing reactors (in some cases with some modifications);

- findings from periodic safety reviews (PSRs);
- insights from probabilistic safety analyses (PSAs);
- insights from operating experience.

In all countries, the defence-in-depth concept is a fundamental technical rationale to assess nuclear safety. It is an important driver to identify any further proposals for safety improvements. In the context of defence-in-depth, the analyses of all six country reports revealed that design extension conditions (DECs) have been integrated in the defence-in-depth concepts. Identifying and analysing DECs are regarded as essential to implement further safety improvements in the NPPs. In all six countries, the preference given to preventive measures, i.e. avoiding the escalation to more severe plant states, in particular accidents with core melt, does not challenge the importance given to complement prevention by implementation of provisions to manage severe accidents.

The analyses of the six country reports showed a high degree of harmonisation in the country-specific approaches. This is probably due to the European harmonisation process expedited by WENRA. In 2014, WENRA published the revised set of SRLs [6] considering the lessons learnt from the accident at Fukushima Daiichi NPP. All six countries are implementing the new set of SRLs in their national regulatory frameworks.

All countries apply the requirements for new reactors as reference targets for existing reactors to identify potential safety improvements. A similar observation was made within Task 2 when evaluating the technical survey.

All six countries require the performance of decennial periodic safety reviews (PSRs). Based on the answers provided, it can be stated that differences in practice exist amongst Member States. For example, France performs the PSR in a two-step approach. In the first step, a specific construction line is considered, and in the second step. individual reactors are analysed in addition to the generic review performed in the first step. Such an approach has advantages in the case of a fleet of similar and standardised reactors with only small differences due to site-specific boundary conditions or differences in the operating history.

PSAs are also considered as an important tool to evaluate safety improvements in NPPs. The objectives of PSAs are twofold. At first, insights from PSAs contribute to the identification of accident sequences with a dominant contribution to the core damage frequency and the frequency and qualification of corresponding releases. Safety improvements can be developed and implemented to reduce the frequency of such accident sequences, thus reducing the overall core damage frequency. Another aspect is the analysis of safety improvements. Here, PSAs help to quantify the influence of safety improvements on the reduction of the core damage frequency or release frequency.

The detailed studies emphasised that the analysis of operating experience is an important rational for safety improvements. The most recent and significant operating experience in the last decades was in connection with the accident at Fukushima Daiichi NPP leading to safety improvements in many European NPPs. An extensive exercise was the ENSREG stress tests which led to National Action Plans for the implementation of further safety improvements. Not only the accident at the Fukushima Daiichi NPP triggered safety improvements but also major or minor events in the past as it was highlighted in the country reports of Finland and Germany.

In addition to the similarities described above, diversity was also identified in the national reports, which is discussed in more detail below.

Feedback from external advisors, research, operating experience

With respect to the feedback from external advisors, the German report describes the Reactor Safety Commission (Reaktor-Sicherheitskommission – RSK) as an external independent advisory body to the German regulator. In contrast to other countries, the RSK has a proactive role in identifying safety improvements. Germany strongly emphasises the “state of the art in science and technology”, “safety research” and “operating experience” as important drivers for identifying further safety improvements. The Romanian report puts a comparatively strong emphasis on operating experience. The analyses of the six country reports revealed that safety research and operating experience are useful drivers for safety. However, it was obvious that the driver “operating experience” has a much more direct influence on safety improvements than the driver “safety research”.

Probabilistic safety targets

Quantitative probabilistic safety limits or targets are applied in three countries (i.e. Finland, Romania and Slovenia). These could be used as “drivers” among other types of drivers for safety improvements, provided that the assumptions and uncertainties in the input data and models of PSAs, which inevitably lead to uncertainties in the absolute results of PSAs, are well understood and documented.

Diversity in drivers and status of regulations

Diversity was also observed in the applied processes for the development of regulations. The involvement of TSOs in the conceptualisation and drafting of regulations can be noted. For example, Slovenia does not formally involve TSOs. However, everybody within the EU is entitled to comment on the Slovenian draft regulations. The regulations are available in national languages. Some countries provide also translations in foreign languages for information. All six countries reported that WENRA SRLs are not directly endorsed (i.e. without transposition in the national regulations). In practice, WENRA SRLs are considered during the process of establishing or revising national regulations. The implementation of WENRA SRLs is followed up by WENRA RHWG benchmark processes [26,27] as well as by self-assessment and annual reporting [28]. It could be concluded that the diversity in the development and issuance of the regulations appears to lead to similar results.

Documenting the technical rationales for exceptions

Another observation made was the documentation of the technical rationales for exceptions and exemptions. For some exemptions noted in the country reports (e.g. Slovenia, PSA Levels 1, 2 and 3 are required for new reactors and PSA Levels 1 and 2 for existing reactor), the technical rationale is not always given in the regulations and might not be easily understood by the public domain. It appears that a lot of attention was paid to the “technical rationale” for safety improvements that are implemented. It also appears to be worthwhile to focus on adequately documenting the “technical rationale” for exceptions or exemptions in the future.

Establishing a nuclear rapid action force (FARN)

The French report described the creation of a nuclear rapid action force (FARN). Similar actions were not described in other country reports. While such an approach may be very valuable in a country with many nuclear sites, it may also be not very useful for countries with only a few sites. Building a consensus on the technical rationale behind nuclear rapid action forces might potentially bring an opportunity for cross-border support at the EU-level.

Design extension conditions

As already mentioned, design extension conditions (DECa) are an important driver for safety improvements in NPPs. It was found that the objectives of implementing design extension conditions in the national regulatory frameworks are clearly similar. However, it was recognised that the applied terminology is different and aligned to the terminology applied in the national regulatory frameworks and possibly also to the native language in the country. For example, the wording for the Finnish classification of DECs includes accident conditions without core melt (in Finland, severe accidents do formally not belong to DECs but to an own class of accidents), but more severe than design basis accidents. An interesting topic for future discussions might be the role of hazards being more severe than those considered for the design basis of an NPP and its relation to DECs.

Safety culture and increasing technical complexity

From the analyses of the six country reports it could be concluded that implementing safety improvements might also increase the technical complexity. The increase in technical complexity also has an impact on the organisations involved in design and construction as well as the staff operating the modified plants. In the future, this may be considered in education and training of the personnel involved in implementing safety improvements. Safety culture and knowledge management in the involved organisations may need appropriate attention in the future. Another aspect in connection with the increased technical complexity is related to the required independence of levels of defence-in-depth. One design approach is to simplify the design, which may be in contradiction to the increasing complexity when trying to address the independence of defence-in-depth. In the future, discussions on the balance between increasing complexity in plant design and the independence of levels of defence-in-depth to avoid a potential decrease in nuclear safety should be promoted.

5.1 Proposals for future activities

Based on the analyses described above, the following topics for future activities are proposed:

- **Proposed suggestion 3.1:** Enhance the mutual understanding of approaches for periodic safety reviews in Member States
- **Proposed suggestion 3.2:** Sharing information on implementing the DEC approach
- **Proposed suggestion 3.3:** Addressing hazards more severe than those hazards considered in the design basis in national regulations and approach to demonstrate robustness of NPPs to avoid cliff-edge effects leading to large or early releases.
- **Proposed suggestion 3.4:** Promote discussion of the increasing complexity in plant designs to fulfil the independency requirement for the levels of defence-in-depth and how to find an adequate balance of both approaches to avoid a decrease in nuclear safety.

The four proposed suggestions were discussed during the 2nd workshop and, considering the proposed suggestions resulting from Task 1 and Task 2, all proposed suggestions were finalised and consolidated (see Chapter 7).

6 ORGANISATION OF TWO WORKSHOPS

The objectives of Task 4 are the technical and organisational management of two workshops, and the drafting and compilation of the final project report. In the first phase of the project, the 1st workshop was organised including the preparation and distribution of the invitation letters and the agenda. The workshop was held from 4 to 5 July 2018 on the premises of the European Commission in Luxemburg. 29 participants from the competent regulatory authorities, the EC, international organisations and TSOs joined the workshop. The participants discussed the first results of the project: the findings of the study on international and European guidance documents prepared under Task 1, the idea and objective of the questionnaire for the technical survey as well as the draft questionnaire prepared under Task 2, and the results of the detailed study on safety upgrades in Slovenia performed under Task 3. In addition, JRC and ENISS presented their recent work. The workshop proceedings were drafted by the project team and approved by the EC. The presentations together with the proceedings were sent to the workshop participants.

In the second phase of the project, the 2nd workshop was organised including the preparation and distribution of the invitation letters and the agenda. It was was held from 12 to 13 November 2019 on the premises of the European Commission in Luxemburg. 30 participants from ten competent regulatory authorities, the EC, ENISS and six TSOs joined the workshop. During the 2nd workshop, the results of the technical survey as well as the evaluation of the detailed studies were presented. Findings and future activities were shared and discussed with the participants. It was agreed that the proposals for future activities will be referred to as “suggestions”. In addition, ENISS and RHWG presented their recent work. ENISS presented its most recent publication “ENISS Position Paper on identification of safety improvements. DiD implementation, Practical Elimination” [29,30,31]. The six principles listed in the Position Paper were introduced and explained to the participants. RHWG presented the first results of the “WENRA-RHWG Pilot Study. Safety Reference Levels 2014 – Implementation at the Nuclear Power Plants, Reasonably Practicable Safety Improvements and Benchmarking” [32]. It could be concluded that the objectives of the 2nd workshop had been fulfilled. The workshop proceedings were drafted by the project team and approved by the EC. The presentations together with the proceedings were sent to the workshop participants.

7 IDENTIFICATION OF TOPICS FOR FUTURE ACTIVITIES TO SUPPORT MEMBER STATES IN IMPLEMENTING ARTICLES 8a – 8c IN PRACTICE

Based on the results of the studies and analyses performed in Task 1, Task 2 and Task 3 and the discussions during the two workshops held in July 2018 and November 2019, the consortium summarises the following suggestions for future activities to support Member States in implementing Articles 8a to 8c in practice and to enhance the mutual understanding of the different approaches observed in the Member States to fulfil the obligations of Articles 8a to 8c on a technical level.

Particularly during the 2nd workshop it was discussed that it should be clearly identified to which organisation or forum the suggestions should be addressed. It was also emphasised that the suggestions should be precisely formulated to be as concrete and specific as possible. To increase the understanding of each suggestion, a short background information should be provided to briefly explain in more detail the idea of the suggestion to the reader.

A total of seven suggestions were derived, which are described below taking into account the feedback and recommendations discussed during the 2nd workshop in November 2019.

Suggestion 1

Increase mutual understanding of approaches to demonstrate avoidance of early releases by sharing the different approaches in the Member States.

Addressees:

- ENSREG / WENRA;
- Industry;
- TSOs.

Additional references:

- IAEA SSG-2 Rev. 1 “Deterministic Safety Analysis for NPPs” [20]

Background:

Suggestion 1 is based on proposed suggestion 1.1 (see Section 3.2) considering the discussions during the 2nd workshop. The decisions to be taken in the event of an early release strongly depends on the site-specific boundary conditions to implement emergency response measures. In addition, the resulting dose rate from an accidental release in relation to off-site countermeasures need to be considered. The first item to be discussed is the point in time of a potential release after the onset of an accident and the time necessary to implement emergency response measures in the vicinity of the site. The crucial point here is the estimation of the period to implement the off-site countermeasures. The second item to be discussed is the link to the intervention levels usually defined in the national regulatory framework. These levels are slightly different across the Member States since their definition is in the responsibility of the national competent authority.

In addition, the para 4.6 of the new IAEA safety guide SSG-2 Rev. 1 "Deterministic Safety Analysis for NPPs" [20] requires that radiological acceptance criteria should be expressed in terms of effective dose, equivalent dose or dose rate to workers at the nuclear power plant, members of the public or the environment, including non-human biota, as appropriate. For accident conditions more severe than traditional design basis accidents, i.e. design extension conditions, para. 4.11 recommends that radiological acceptance criteria should ensure that protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures.

Suggestion 2

Increase mutual understanding of quantitative and qualitative approaches to demonstrate avoidance of large releases. This would first require a mutual understanding of approaches for radiological acceptance criteria for design extension conditions.

Addressees:

- ENSREG / WENRA;
- Industry;
- TSOs.

Additional references:

- IAEA SSG-2 Rev. 1 "Deterministic Safety Analysis for NPPs" [20]

Background:

Suggestion 2 is based on the proposed suggestion 1.2 (see Section 3.2) considering the discussions during the 2nd workshop. The analyses performed in Task 1 and Task 2 reveal that high level qualitative radiological acceptance criteria as well as quantitative radiological acceptance criteria are applied. In addition, the new IAEA safety guide SSG-2 Rev. 1 "Deterministic Safety Analysis for NPPs" [20] requires in para 4.6 that radiological acceptance criteria should be expressed in terms of effective dose, equivalent dose or dose rate to workers at the nuclear power plant, members of the public or the environment, including non-human biota, as appropriate. For accident conditions more severe than design basis accidents, i.e. design extension conditions, para. 4.11 recommends that radiological acceptance criteria for design extension conditions should ensure that protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures. The discussions should focus on the mutual understanding of the background of quantitative radiological acceptance criteria and the methodology to derive quantitative acceptance criteria from high-level qualitative acceptance criteria.

Suggestion 3:

Similarities and differences of these case-by-case analyses for application of the most up-to-date safety objectives could be a relevant scope for future work of ENSREG and WENRA. Exchange on this scope could in particular foster a mutual understanding of regulatory decision making.

In particular, considering that the consortium supports pursuing the reinforcement of the DEC concept application, the scope could be first focusing on DECs. Within this context, on-going WENRA work related to the implementation of SRLs of issue F (Design Extension Conditions) at the plants appears to be an adequate tool.

Addressees:

- ENSREG / WENRA.

The ongoing WENRA / RHWG study on “WENRA Safety Reference Levels 2014 – Implementation at the Nuclear Power Plants, Reasonably Practicable Safety Improvements and Benchmarking” is a relevant activity with respect to this suggestion.

Additional references:

- WENRA RHWG project on “WENRA Safety Reference Levels 2014 – Implementation at the NPPs, reasonably practicable safety improvements” published on 11 November 2019

A potentially interesting document with respect to suggestion 3 is the IAEA TECDOC on “Experiences on implementing safety improvements at existing nuclear power plant” which is currently under preparation.

Background:

Suggestion 3 is mainly based on proposed suggestion 2.1 (see Section 4) as well as suggestion 1.3 (see Section 3.2).

The consortium highlights that there is a remarkable similarity that contributes to the fulfilment of the safety objective of Council Directive 2014/87/Euratom:

- considering the most up-to-date safety objectives for any reactor irrespective of whether it is in operation (in particular for the periodic safety review), under construction or in the design phase;
- for operating reactors, applying these objectives on the basis of an analysis taking into account both the characteristics of the respective installation and the promotion of a continuous improvement of safety;

In addition, the analysis performed in Task 2 and Task 3 indicates that a common approach is applied in the Member States to meet the nuclear safety objective required in Article 8a for new NPPs and that for existing NPPs, the nuclear safety objectives serve as a reference to identify potential safety improvements. However, the decision to implement a specific safety improvement is always a case-by-case decision for each individual NPP under reassessment. To increase the mutual understanding of the regulatory decision-making processes either to require the implementation of a safety improvement or to conclude that an implementation is not necessary should be discussed in more detail. The ongoing work within WENRA / RHWG could be considered as a starting initiative for such an activity. These discussions should focus on the technical rationale, e.g. how to consider safety features already implemented either by design or by earlier backfitting, the role of deterministic and probabilistic safety analyses to support decision-making, etc.

Suggestion 4

The consortium highlights the benefit of using a systematic approach for ensuring the continuous improvement of safety, based on:

- identifying the sequences entailing the most critical safety issues, and
- adequately reinforcing the requirements on the corresponding safety provisions – either to prevent these sequences or to limit their consequences.

The consortium suggests that the European nuclear industry, TSOs, ENSREG and WENRA share their practices on this topic to contribute to the identification and development of new improvements.

Addressees:

- ENSREG / WENRA;
- Industry;
- TSOs.

Additional references:

- WENRA Guidance "Article 8a of the EU Nuclear Safety Directive: ''Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants''"

Background:

Suggestion 4 is based on proposed suggestion 2.2 (see Section 4). This suggestion results from the analysis of the questionnaire related to Article 8b of Council Directive 2014/87/Euratom. An important aspect was the reinforcement of the defence-in-depth concept. Addressing design extension conditions (DECs) led to several safety improvements. Although differences exist in the applied terminology, similar safety improvements have been achieved.

Suggestion 5

The consortium suggests to the regulatory authorities, TSOs and the European nuclear industry to consider the recent WENRA report on practical elimination as a reference document. If they would be involved in international cooperation on practical elimination, it would be better to focus the discussion on similarities in the application of the concept in order to identify further insights related to its justification. Discrepancies in the theoretical view of practical elimination have already been discussed in the past and do not seem to be a worthwhile topic for harmonisation anymore.

Addressees:

- National regulatory authority;
- National TSOs;
- Industry.

Additional references:

- WENRA report "Practical Elimination Applied to New NPP – Designs – Key Elements and Expectations Guidance", published in 2019 as a relevant reference document

- IAEA on-going work (Draft Safety Standard DS508 and development of a TECDOC) notably on practical elimination (as a relevant framework for applying the recommendation (potentially)

Background:

Suggestion 5 is based on proposed suggestion 2.3 (see Section 4). From the analyses performed within this study it could be concluded that a consensus amongst the Member States exists on the requirement to practically eliminate early and large releases in case of accident conditions, in particular severe accidents with core melt. Different views exist on the safety demonstration with regard to practical elimination. At present, this discussion is not limited to Member States in Europe but also internationally discussed such as at the IAEA.

Suggestion 6

Enhance mutual understanding of approaches for periodic safety reviews with particular focus on the implementation of the DEC approach in Member States.

Addressees:

- WENRA / ENSREG

Additional references:

- WENRA "Position paper on Periodic Safety Reviews (PSRs) taking into account the lessons learnt from the TEPCO Fukushima Dai-ichi NPP accident"

A potentially interesting document with respect to suggestion 6 is the IAEA TECDOC on "Experiences on implementing safety improvements at existing nuclear power plant", which is currently under preparation.

Background:

Suggestion 6 is mainly based on proposed Suggestions 3.1 and 3.2 (see Section 5.1). In all Member States, decennial periodic safety reviews are performed. However, differences exist in the applied approaches to perform PSRs. Within this activity, the differences in the national approaches should be elaborated in more detail. An important aspect should be the analysis of potential accident sequences belonging to the DEC regime. During the 2nd workshop it was emphasised that DEC sequences for which currently no technical solutions are available, need a reassessment in a later PSR. The progressing state of the art in science and technology as well as in safety analysis methodology or insights from experimental studies may open technical solutions which may be implemented at the NPPs.

Suggestion 7

Promote discussions on the balance between the increasing complexity in plant designs and the independence of levels of defence-in-depth to avoid a potential decrease in nuclear safety. Practical examples might include side effects of modifications, shared equipment in multi-unit sites, and I&C. A development of guidance may be needed to adequately support decision-making.

Addressees:

- WENRA / ENSREG

Additional references:

None.

Background:

Suggestion 7 is based on proposed suggestion 3.4 (see Section 5.1). To implement the requirement of independence of levels of defence-in-depth, independent and usually diverse safety features must be implemented, which increases the complexity of the plant systems. On the other hand, simplification is an approach to reduce vulnerability of technical systems by reducing the sources of failures. Within this activity, the approaches to assess the balance between complexity and simplicity to avoid a decrease in nuclear safety and meeting the independence requirement should be elaborated in more detail. The result of this activity may be a guidance to support the regulatory decision-making process with a focus on plant modifications.

8 REFERENCES

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[10] WENRA-RHWG “Guidance Document Issue T: Natural Hazards (Head Document)”, WENRA, April 2015

[11] WENRA-RHWG “Guidance on Issue F Design Extension Conditions”, WENRA, September 2014

[12] RHWG “Position paper on Periodic Safety Reviews”; WENRA, March 2013

[13] RHWG Report “WENRA Safety Reference Levels 2014 – Implementation at the nuclear power plants, reasonably practicable safety improvements and benchmarking – Pilot Study”, WENRA, November 2019

[14] INSAG-10 “Defence in Depth in Nuclear Safety”, International Nuclear Safety Advisory Group, Vienna, 1996

[15] INSAG-12 “Basic Safety Principles for NPP”, International Nuclear Safety Advisory Group, Vienna, 1999

[16] IAEA, Safety Standard Series No. SSR 2/1 “Safety of Nuclear Power Plants: Design” Rev. 1, Vienna, 2016

[17] IAEA, Safety Standard Series No. SSR 3 “Safety of Research Reactors”, Vienna, 2016

[18] IAEA, Safety Standard Series No. SSG-25 “Periodic Safety Review for Nuclear Power Plants”, Vienna, 2013

[19] IAEA, Safety Standard Series No. SSG-2 “Deterministic Safety Analysis for Nuclear Power Plants”, Vienna, 2010

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EUROPEAN TECHNICAL SAFETY ORGANISATIONS NETWORK

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

FINAL STUDY

VOLUME 2

STUDY ON INTERNATIONAL AND EUROPEAN GUIDANCE DOCUMENTS

August 2020

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Table of contents

List of acronyms

AOO	Anticipated Operational Occurrence
ATWS	Anticipated Transient without Scram
Bel V	Belgian TSO
BMU	Federal Ministry for the Environment, Nature Conservation and Nuclear Safety
CLI	Criteria for Limited Impact
DBA	Design Basis Accident
DEC	Design Extension Condition
DiD	Defence-in-Depth
EC	European Commission
ETSON	European Technical Safety Organisations Network
EU	European Union
EUR	European Utility Requirements
GPR	Groupe permanent d'experts pour les réacteurs nucléaires
IAEA	International Atomic Energy Agency
INSAG	International Nuclear Advisory Group
JRC	Joint Research Centre
NPP	Nuclear Power Plant
NRA SR	Nuclear Regulatory Authority of the Slovak Republic
PSR	Periodic Safety Review
RHWG	Reactor Harmonisation Working Group
SSCs	Structures, Systems and Components
TSO	Technical Safety Organisation
WENRA	Western European Nuclear Regulators Association

List of tables

List of figures

1 Introduction

1.1 Starting point

With the adoption of Council Directive 2009/71/Euratom, Member States are obliged to establish a common framework for nuclear safety. In the aftermath of the accident at the Japanese nuclear power plant Fukushima Daiichi, this Council Directive was amended by Council Directive 2014/87/Euratom. By Article 8a of this Council Directive, a nuclear safety objective common for all Member States has been established.

Article 8a reads as follows

> "*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.*"

It is further specified that the above-mentioned nuclear safety objective has to be applied to all

> "*nuclear installations for which a construction licence is granted for the first time after 14 August 2014*".

In addition, it is required that the nuclear safety objective serves as a

> "*reference for the timely implementation of reasonably practicable safety improvements to existing nuclear installations*".

Since Council Directive 2014/87/Euratom introduces several terms, but without definition or glossary, the meaning of the terms leaves room for interpretation by the Member States and thus challenges a harmonised approach in Europe.

The terms under consideration are

- “to prevent”,
- “to mitigate”,
- “to avoid”,
- “early releases” and
- “large releases”.

Regarding the identification of potential safety improvements for existing reactors to meet the nuclear safety objectives, the following terms will be thoroughly discussed in this study:

- “timely implementation”;
- “reasonably practicable”.

To achieve the nuclear safety objective according to Article 8a, Member States are obliged to implement requirements for applying a defence-in-depth concept. The main elements of the defence-in-depth concept required according to Article 8b of Council Directive 2014/87/Euratom are as follows:

- “*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*”.

In addition, it is required that

> “*the impact of extreme external natural and unintended man-made hazards is minimised*”

and that

> “*organisational structures according to Article 8d (1) are in place*”.

Furthermore, Article 8b contains the obligation for the licence holder to achieve the nuclear safety objective set out in Article 8a. These requirements are of a more organisational nature and concretise the requirements set out in Article 6 of Council Directive 2009/71/Euratom.

For the required management system according to Article 6(d), Article 8b stipulates the following:

> "*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*".

Further requirements deal with the operational experience feedback and the reporting obligations of safety-relevant events to the regulatory body. With the requirements stated in Article 8b

> "*arrangements by the licence holder to register, evaluate and document internal and external safety significant operating experience*"

and

> "*the obligation of the licence holder to report events with a potential impact on nuclear safety to the competent regulatory authority*"

the licence holder is forced to thoroughly evaluate and analyse operational experience feedback, which will contribute to further improve nuclear safety and serves as one of the inputs for the periodic safety reviews (PSRs) required in Article 8c.

Article 7 of Council Directive 2009/71/Euratom stipulates the following requirement for the licence holder

> "*regularly assess and verify, and continuously improve, as far as reasonably achievable, the nuclear safety of their nuclear installations in a systematic and verifiable manner*".

This general requirement of performing a PSR is more substantiated in Article 8c. In addition to the initial review before granting a construction licence, it is now legally binding for all Member States to perform a decennial PSR. According to Article 8c, this PSR shall ensure the following:

> "*(…) compliance with the current design basis and identifies further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards, using as a reference the objective set in Article 8a.*"

Within this study, international and European documents thoroughly reviewed to identify how these documents can support Members States in implementing the provisions set out in

Articles 8a to 8c of Council Directive 2014/87/Euratom. In addition to emphasising good practices in the international, European and national practices, potential areas for improvement will be identified. Discussions during the 1st workshop held on 4 and 5 July 2018 in Luxemburg are included in this report. The objective of this study is to contribute to a further harmonisation of nuclear safety in the Member States and support a common implementation of the nuclear safety objective and the related requirements for continuous improvement of nuclear safety in existing nuclear power plants in Europe.

1.2 Identification of international and European guidance documents

As this study strongly focuses on the European framework, the information available from European organisations is most important. In particular, documents published by the Western European Nuclear Regulators Association (WENRA) and its Reactor Harmonisation Working Group (RHWG) are the most important documents. In addition, supplementary information can be obtained from the European Utility Requirements (EUR), a set of requirements for new nuclear power plants published by the European operators. However, preference is given to the documents published by WENRA because these publications represent a common position of the European regulators whereas the EUR represent the position of the licence holders.

The following WENRA documents have been identified as relevant for this study:

- WENRA Safety Reference Levels for Existing Reactors (2014);
- WENRA statement – “Safety of new NPP designs”;
- WENRA RHWG Report "Safety of new NPP designs";
- WENRA Guidance "Article 8a of the EU Nuclear Safety Directive: “Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants””;
- WENRA-RHWG Guidance Document Issue T: Natural Hazards (Head Document);
- WENRA-RHWG Guidance on Issue F (Design Extension Conditions);
- RHWG - Position paper on Periodic Safety Reviews 2013.

These documents have been analysed and the results are summarised in section 4.

Publications of IAEA’s advisory body, the International Nuclear Safety Advisory Group (INSAG), can be regarded as international high-level recommendation. INSAG publications do not present a common view of IAEA Members States but express the views of renowned experts in the field of nuclear safety. Recommendations of INSAG are published by the IAEA.

In the context of this project, the following INSAG reports were referred to:

- INSAG-10 “Defence in Depth in Nuclear Safety”;
- INSAG-12 “Basic Safety Principles for NPP”.

These documents have an impact on the development of IAEA safety standards. For that reason, a short summary of the main conclusions from these documents is provided in section 2.

IAEA safety standards represent the global consensus with respect to nuclear safety. During the development of these standards, relevant INSAG reports and expertise from IAEA Member States are considered. In the framework of this project, the most relevant IAEA publications are

- IAEA Safety Standard Series SSR 2/1 Rev. 1“Safety of Nuclear Power Plants: Design”;
- IAEA Safety Standard Series SSR 3 “Safety of Research Reactors”;
- IAEA Safety Standard Series SSG-25 “Periodic Safety Review for Nuclear Power Plants”;
- IAEA Safety Standard Series SSG-2 “Deterministic Safety Analysis for Nuclear Power Plants” as well as DS491 “Deterministic Safety Analysis for Nuclear Power Plants”[1].

Furthermore, the IAEA Safety Glossary [18] and its updated draft from 2016 [19] were used.

Latest discussions on the further development of the defence-in-depth concept and the design of nuclear power plants are reported in IAEA TECDOC 1791 "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants". This TECDOC reflects the European position as published by WENRA, but also discusses further approaches applied in IAEA Member States. A summary of the IAEA concept and the latest discussions is given in section 3.

[1] In i2019, DS491 was published as SSG-2 Rev.1.

1.3 Objectives of Task 1 and interfaces with other tasks of the project

The results of Task 1 were used to support the development of the technical survey to be performed in Task 2 and for the detailed studies to be performed within Task 3.

After analysing the identified international and European documents with respect to Articles 8a to 8c of Council Directive 2014/87Euratom, a gap analysis was performed. The observations were categorised as follows:

- "*No guidance available*" – if no specific guidance can be found in the analysed documents or if only a few organisations address the issue under consideration.
- "*Ambiguous guidance*" – if the provided guidance differs amongst the analysed documents and leaves room for interpretation.
- "*Harmonised guidance*" – if the provided guidance is consistent in all analysed documents.

Based on the identified gaps, advice for technical areas requiring future work was proposed. The focus was placed on ensuring a harmonised application of the approaches and a common understanding of the definitions of terms in Europe.

The results of Task 1 were presented during the 1st workshop and discussed with the participants. The discussions were taken into account for the drafting of this report.

2 Review and assessment of INSAG documents

In 1996, the International Nuclear Safety Advisory Group (INSAG), an advisory group to IAEA, published the report “*Defence in Depth in Nuclear Safety*” [4]. This document discusses the general idea of the defence-in-depth concept and introduces five levels of defence-in-depth. As this document is the basis for most of the international discussions and developments, it is worthwhile summarising the main statements from this publication.

Table 1: Levels of defence-in-depth according to INSAG-10 [4].

Level of defence in depth	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

The comparison of Article 8b of Council Directive 2014/87/Euratom with the objectives for the first four levels shows that the Council Directive reflects the objectives formulated in INSAG-10. It should be noted that the in-depth defence concept discussed in INSAG 10 is mainly written for existing reactors and reflects their situation. In addition, section 5.2 of INSAG-10 [4] discusses the defence-in-depth concept for the next generation of nuclear reactors, thus introducing the group of accidents with multiple failures. It is stated that some of the multiple failure events could be assigned to level 3 of defence in depth, but others to level 4 of defence in depth. It can be concluded that multiple failure events are not clearly assigned to a certain level of defence in depth in INSAG-10.

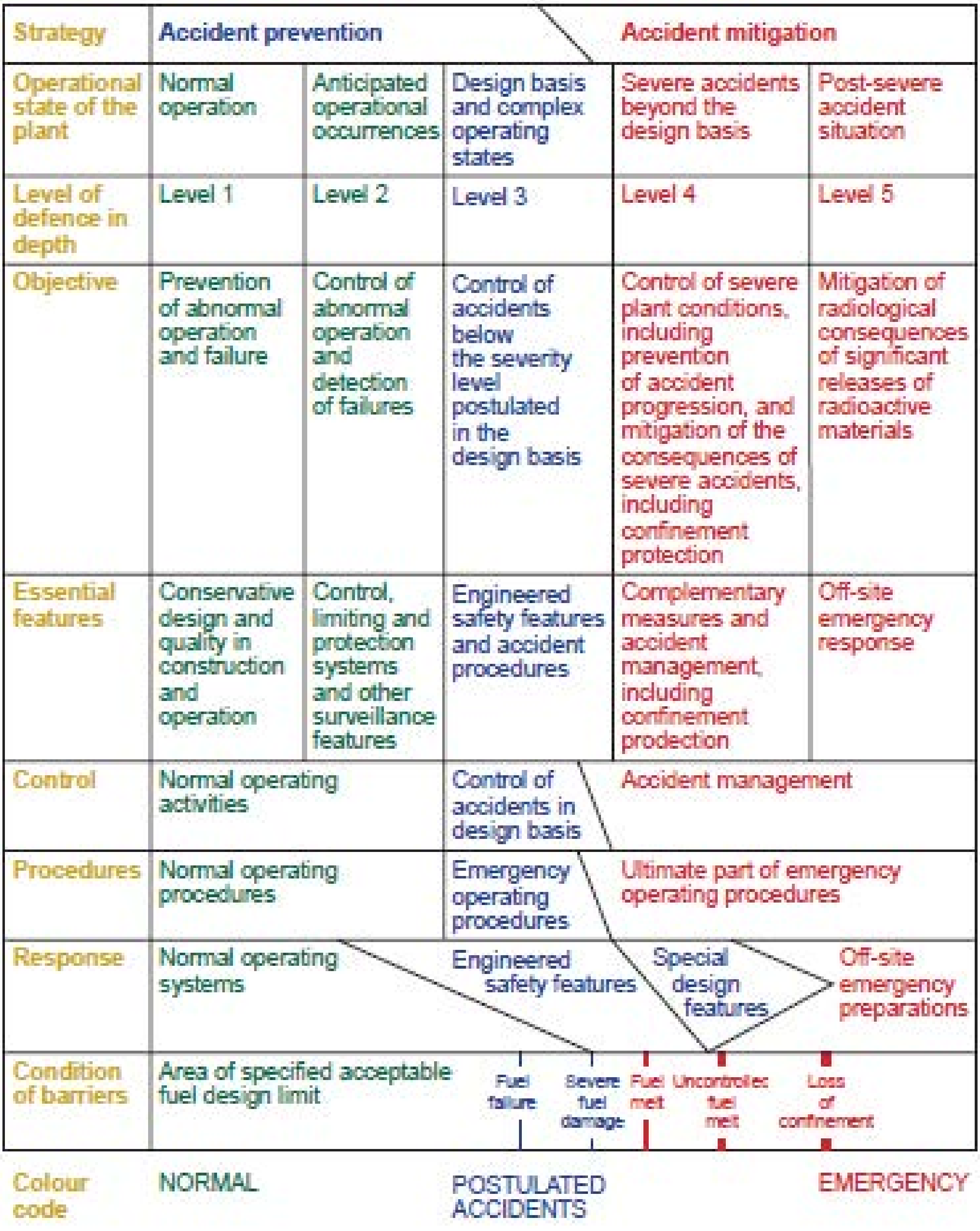
Within INSAG-10, the term ‘to avoid’ is not applied. In the context of the defence-in-depth concept, the terms ‘to control’, ‘to prevent’ and ‘to mitigate’ are used. ‘To control’ means any action necessary to deal with a certain plant state, like AOOs, DBAs or severe accidents, by crediting the equipment provided on the specific level of defence in depth. The meaning of ‘to prevent’ is ambiguous. On the one hand, it means to prevent the escalation to a more severe plant state and, on the other hand, to prevent severe accidents (i.e. accidents with core melt). The term ‘to mitigate’ is applied in the context of radiological consequences. Mitigation aims at the minimisation of radiological consequences in case of severe accidents by applying the ALARA principle, taking into account economic and social factors.

INSAG-10 does not provide a clear definition for “early releases” or “large releases”. For new NPP designs it is expected

> *“that hypothetical severe accident sequences that could lead to large radioactive releases due to early containment failure”*

have to be practically eliminated. From this statement it can be interpreted that an early release is always a large release. However, neither a clear qualitative nor a quantitative definition of an early or large release is given in INSAG-10.

The design of nuclear power plants is further discussed in the INSAG-12 report “Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1” [5]. The described defence-in-depth concept is identical with the description in INSAG-10 (see Table 1). In contrast to INSAG-10, the appendix in INSAG-12 emphasises that ‘to mitigate’ aims at minimising the consequences of severe accidents. It is clearly described that accident prevention is the objective of levels 1 to 3 of defence in depth and that accident mitigation is the objective of levels 4 and 5 of defence in depth, as shown in Figure 1.



Strategy	Accident prevention			Accident mitigation	
Operational state of the plant	Normal operation	Anticipated operational occurrences	Design basis and complex operating states	Severe accidents beyond the design basis	Post-severe accident situation
Level of defence in depth	Level 1	Level 2	Level 3	Level 4	Level 5
Objective	Prevention of abnormal operation and failure	Control of abnormal operation and detection of failures	Control of accidents below the severity level postulated in the design basis	Control of severe plant conditions, including prevention of accident progression, and mitigation of the consequences of severe accidents, including confinement protection	Mitigation of radiological consequences of significant releases of radioactive materials
Essential features	Conservative design and quality in construction and operation	Control, limiting and protection systems and other surveillance features	Engineered safety features and accident procedures	Complementary measures and accident management, including confinement prodection	Off-site emergency response
Control	Normal operating activities		Control of accidents in design basis	Accident management	
Procedures	Normal operating procedures		Emergency operating procedures	Ultimate part of emergency operating procedures	
Response	Normal operating systems		Engineered safety features	Special design features	Off-site emergency preparations
Condition of barriers	Area of specified acceptable fuel design limit		Fuel failure; Severe fuel damage	Fuel melt; Uncontrolled fuel melt	Loss of confinement
Colour code	NORMAL		POSTULATED ACCIDENTS		EMERGENCY

Figure 1: Overview of defence in depth as described in the appendix of INSAG-12 [5].

3 Review and assessment of IAEA documents

The nuclear safety objective for nuclear installations as well as its implementation at nuclear installations as addressed in Articles 8a and 8b of [16] are outlined in various IAEA safety standards. In particular, requirements regarding Article 8b are contained in the IAEA Safety Requirements SSR 2/1 “Safety of Nuclear Power Plants: Design” [12] and SSR 3 “Safety of Research Reactors” [17]. Requirements with respect to Article 8c, i.e. periodic safety reviews, are detailed in the IAEA Safety Guide SSG-25 “Periodic Safety Review for Nuclear Power Plants” [10]. The draft safety guide DS491 [14] (revision of Safety Standard SSG-2 “Deterministic Safety Analysis for Nuclear Power Plants” [13]) provides a guidance on the approach of deterministic safety analyses and the use of technical and radiological acceptance criteria to demonstrate compliance with safety objectives for the different plant states.

To harmonise the terminology applied in IAEA Safety Standards, a glossary was published by the IAEA [18]. In 2016, an updated draft of the IAEA Safety Glossary was published [19][2]. In the following, the definitions of the applied terms are taken from this latest version of the Safety Glossary. The Safety Glossary provides a brief definition of the term ‘defence in depth’:

> “ ***defence in depth*** *- A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.*”

The entry ‘defence in depth’ in the new draft IAEA Safety Glossary provides further information on defence in depth as required in further IAEA Safety Standards, namely

- IAEA Safety Standard Series No. SF-1 “Fundamental Safety Principles” [10] and
- IAEA Safety Standards Series No. SSR 2/1 “Safety of Nuclear Power Plants: Design” [12].

For completeness, information from the report INSAG No. 10 [4] is mentioned in the definition of defence in depth.

[2] In 2019, the IAEA published the “IAEA Safety Glossary: 2018 Edition”.

The IAEA Safety Glossary as such does not contain specific entries for the terms

- “to prevent”,
- “to mitigate” and
- “to avoid”.

However, the meaning of these terms results from the context of the glossary. In the introduction to the glossary it is stated that

> “*’Safety measures’ include actions to prevent incidents and arrangements put in place to mitigate their consequences if they were to occur.*”

The term “to avoid” is mainly used in connection with response to emergency situations in order to protect people and the environment, i.e. avoid means “to prevent” health consequences after an accident. For example, see the following definitions in [19]: “emergency planning zone”, “evacuation”, “exposure situations”, “protective action” and “relocation”.

The terms “early releases” and “large releases” are defined in a qualitative way as part of the definition of “defence in depth” in the IAEA Safety Glossary [19]:

> “*early release of radioactive material. A release of radioactive material for which off-site protective actions are necessary but are unlikely to be fully effective in due time.”*

> “ *large release of radioactive material. A release of radioactive material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment.”*

According to IAEA DS491 [14], technical and radiological acceptance criteria should be defined for all plant states including accident conditions. These acceptance criteria can be general, qualitative or quantitative. In particular safety criteria should be related to radiological criteria or to the integrity of barriers with due consideration of maintaining the safety functions (technical criteria). Using deterministic safety analyses, compliance with regulatory safety objectives can be demonstrated by meeting the defined radiological and technical safety criteria. It is further stated that radiological criteria are usually expressed in terms of activity levels, doses or dose rates. A more detailed overview of the radiological criteria for the different plant states are shown in Table 2.

Table 2: Radiological acceptance criteria for the deterministic safety analysis according to DS491 [14].

<table>
<tr><th>Plant state</th><th>Radiological criteria</th></tr>
<tr><td>Normal operation</td><td>Effective dose limits for workers
Effective dose limits for members of the public in the vicinity of the plant
Authorized limits on the activity in planned discharges from the plant</td></tr>
<tr><td>Anticipated operational occurrences</td><td>Should be more restrictive than for design basis accidents, as such plant states are more frequent.</td></tr>
<tr><td>Design basis accidents</td><td>Acceptable limits for radiation protection shall not be exceeded
have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective actions</td></tr>
<tr><td>DEC without fuel melt</td><td rowspan="2">No unacceptable radiological consequence
Protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures.</td></tr>
<tr><td>DEC with fuel melt</td></tr>
</table>

As usual practice, IAEA safety standards do not provide recommendations for quantitative values. The definition of quantitative values is the responsibility of the Member States.

It can be summarised that Council Directive 2014/87/Euratom [25] and IAEA Safety Standards use the terms “to prevent”, “to mitigate”, “to avoid”, “early releases” and “large releases” in a consistent manner.

As already mentioned, references [12] and [17] are mainly dedicated to design requirements. SSR 2/1 [12] already uses the terms “to prevent” and “to mitigate” in the introductory part. The objective of SSR 2/1 reads as follows:

> “ *1.4. This publication establishes design requirements for the structures, systems and components of a nuclear power plant, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur.*”

> “ *1.5. This publication is intended for use by organizations involved in design, manufacture, construction, modification, maintenance, operation and decommissioning for nuclear power plants, in analysis, verification and review, and in the provision of technical support, as well as by regulatory bodies.*”

It should be noted that the declared objective of IAEA SSR 2/1 [12] relates to design requirements and does not distinguish between commissioned and existing plants. As mentioned above, IAEA SSR 2/1 [12] uses standard terminology as published in the IAEA Safety Glossary [18][19] with exception of the terms that are defined in the part “Definitions” of SSR 2/1 [12].

It can be concluded that there is a quite straightforward link between requirements in [12] and obligations in [25]:

- Article 8a (Nuclear safety objective for nuclear installations) in [25] is related to section 2 of [12], Applying the safety principles and concepts. Similarly, the IAEA safety requirement for research reactors SSR 3 [17] contains a section “Applying the safety objective, concepts and principles for research reactor facilities”.
  Obligation under Part (b) of Article 8a “*the timely implementation of reasonably practicable safety improvements*” is covered by paragraph 1.3 of [12]:

  > “ *It might not be practicable to apply all the requirements of this Safety Requirements publication to nuclear power plants that are already in operation or under construction. In addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further*

enhanced by means of reasonably practicable safety improvements."

Regarding research reactors, "the timely implementation of reasonably practicable safety improvements" requirement is more freely formulated in paragraph 1.6 of [17].

- Article 8b (Implementation of the nuclear safety objective for nuclear installations) is related to sections 3, 4 and 6 of SSR 2/1 [12] that cover Management of safety in design, Principal technical requirements and General plant design. IAEA SSR 3 [17] follows a similar philosophy as SSR 2/1 [12].

- Part (a) of Article 8c [25] requiring an initial assessment is more related to the issues discussed in the above paragraph.

- Part (b) of Article 8c [25] requiring periodic safety reviews is discussed in more detail in IAEA Safety Guide SSG-25 [10]. This safety guide provides recommendations and guidance to conduct a periodic safety review (PSR) for an existing nuclear power plant. It serves as a guidance for operating organisations, regulatory bodies and their technical support organisations (TSO) etc. to manage such review in a systematic manner. It is obvious that the PSR as such is equivalent to the obligation "re-assesses systematically and regularly, at least every 10 years". It should be noted that IAEA SSG-25 [11] is intended to be used for an existing nuclear power plant. Obligations of Part b) of Article 8c regarding research reactors are covered by paragraphs 4.25, 7.121 and 7.122 of SSR-3 [17]. However, requirements of quoted paragraphs are more general than requirements stated in [11].

Finally, any design and operation of plants performed in compliance with the requirements of IAEA standards [10], [12] and [17] meets the obligations of Articles 8a to 8c of Council Directive 2014/87/Euratom [25]. Regarding operating plants, it can be stated that the obligation under Article 8a "*the timely implementation of reasonably practicable safety improvements*" means harmonisation of the plant status with current safety requirements at least as part of the decennial periodic safety review process.

4 Review and assessment of WENRA documents

In Europe, the documents published by the Western European Regulators Association (WENRA) play an important role since they represent a consensual view with respect to nuclear safety of all national regulatory authorities regulating nuclear power plants in Europe.

With respect to Article 8a of Council Directive 2014/87/Euratom, the following information can be extracted from published WENRA documents.

WENRA published safety objectives for new reactors. These are formulated in a qualitative manner to drive design enhancements for new plants with the aim of obtaining a higher safety level compared to existing plants.

According to WENRA, the new reactors are expected to be designed, sited, constructed, commissioned and operated in such manner that in case of […] "*Accidents with core melt*" (Objective O3), the potential radioactive releases to the environment will be reduced, also in the long term, by following the qualitative criteria below:

> " *accidents with core melt which would lead to early or large releases have to be practically eliminated;"*
>
> *" for accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.*"

WENRA defines an "early release" as a release that would require off-site emergency measures but with insufficient time to implement them. A "large release" is defined as situations that would require protective measures for the public that could not be considered as limited in area or time. The above specified definitions and objectives are in concordance with Article 8a of the Council Directive 2014/87/Euratom statement.

To achieve Objective O3, it is expected that the off-site radiological impact of accidents with core melt which are not practically eliminated will require only limited protective measures in area and time. In the view of WENRA, this means no permanent relocation, no long-term restrictions in food consumption, no need for emergency evacuation outside the immediate vicinity of the plant and only limited sheltering will be allowed [8]. As design goals for new nuclear power plants, WENRA proposes an evacuation zone (comparable to a precautionary action zone – PAZ) with a radius of 3 km (no need for emergency evacuation beyond the immediate vicinity of the plant) and a radius of 5 km for the sheltering zone (comparable to an urgent protective action planning zone – UPZ). Details are given in Table 3. It is expected that no sheltering and iodine prophylaxis beyond the zone will be necessary. It should be noted that the specified zones are not meant to be used for emergency preparedness planning but for the safety demonstration of the radiological safety objective. It is considered that agricultural products beyond the sheltering zone should generally be consumable after the first year following the accident.

Table 3: Design goals for areas where limited protective measures may be needed [8]

Measure	Evacuation zone	Sheltering zone	Beyond sheltering zone
Permanent relocation	No	No	No
Evacuation	May be needed	No	No
Sheltering	May be needed	May be needed	No
Iodine Prophylaxis	May be needed	May be needed	No

In addition to the design goals related to limited protective measures, the ALARA principle shall be applied and any reasonably achievable measure should be implemented which would further reduce the radiation doses of workers or the population or environmental consequences. There must also be sufficient time available to implement these measures (protective measures should be initiated early enough).

WENRA does not publish probabilistic target values in terms of core damage frequency (CDF) or large early release frequency (LERF).

WENRA emphasises in its Safety Reference Level A2.3 that the safety policy shall ask for continuous improvement of nuclear safety through timely implementation of the reasonably practicable safety improvements identified, taking into account operating experience, safety research, and advances in science and technology [21].

The term “reasonably practicable” is used in the context of reducing risks as low as reasonably practicable or improving safety as far as reasonably practicable. The concept of reasonable practicability may be considered analogous to the ALARA principle, but with a broader scope, since it applies to all aspects of nuclear safety. The WENRA interpretation of reasonably practicable means that in many cases adopting modern safety standards and practices (sometimes referred to as good practices) in the nuclear field will be sufficient to show achievement of what is “reasonably practicable” [9].

For the existing reactors, where a relevant modern standard or good practice associated with new reactors is not directly applicable or cannot be fully implemented, alternative safety or risk reduction measures (design and/or operation) to prevent or mitigate radioactive releases should be identified and implemented unless the utility is able to demonstrate that the efforts to implement them are disproportionate to the safety benefit they would confer.

The test of reasonable practicability only leads to no implementation of improvements if the effort involved is disproportionate to the safety benefit. Justifications based on lack of financial resources will not be accepted, even if the costs are reasonable. Being “proportionate” is a strong element in deciding what is or is not reasonably practicable, and the decision should consider the following aspects:

- the specific attributes of a facility and its future lifetime;
- the greater the shortfall, the more needs to be done to identify and implement measures to remove or reduce it.

Attention should be paid to the fact that certain safety improvements that may be reasonable in one case may not be necessary for another, or conversely may be insufficient, so better or more measures might be required.

In a particular case, the process supposes that the licence holder will demonstrate that all significant shortfalls have been identified, all possibilities to address them have been considered and the reasonably practicable option selected, and in case where no options are selected, it is shown that they are disproportionate with sufficient rigor and confidence.

Consideration of the options concerning proposed measures to minimise the shortfall needs to be carried out in a holistic way, taking into account all potential impacts on the safe operation of the plant.

Time is another important factor in determining reasonably practicable improvements for existing reactors. Sometimes, a modest mobile system that provides less benefit for longer might be the better option, and sometimes a more expensive solution that can be implemented earlier may be better than less expensive solutions that take a long time to implement.

Article 8b of Council Directive 2014/87/Euratom recalls the main elements of the defence-in-depth concept. The WENRA safety objectives call for an extension of the safety demonstration for new plants, consistent with reinforcement of **defence-in-depth** (DiD), in order to prevent, or if prevention fails, to mitigate any harmful radioactive releases. In addition to the reinforcement of each level of the DiD concept and the improvement of the independence between the levels of DiD (as stated in the WENRA safety objectives), this means also that the principle of multiple and independent barriers should be applied for each significant source of radioactive material.

The scope of the related safety demonstration has to cover all risks induced by the nuclear fuel, including all fuel storage locations, as well as the risks induced by other relevant radioactive materials.

Based on the fact that phenomena involved in accidents with core/fuel melt differ radically from those which do not involve a core melt, WENRA considers that for new reactors, design features that aim at preventing a core melt condition and that are credited in the safety demonstration should not belong to the same level of defence as the design features that aim at controlling a core melt accident that was not prevented. WENRA claims that a refined structure of the DiD levels is necessary, as it is shown in Table 4.

Even though no new safety level of defence is suggested, a subdivision of level 3 DiD was created, and therefore a clear distinction between means and conditions for sub-levels 3a and 3b was proposed.

For new plants, on DiD level 3, associated plant conditions are now broader than those for existing reactors as they now include multiple failure events which were previously considered as "beyond design". Design provisions considered in level 3b for postulated multiple failures shall further decrease the frequency and/or mitigate consequences of sequences beyond those considered in the design basis for existing reactors so far, such as anticipated transients without scram (ATWS) or station black out (SBO) scenarios. For level 3b, analysis methods and boundary conditions, design and safety assessment rules may be developed according to a graded approach, also based on probabilistic insights.

Table 4: The refined structure of the levels of DiD as published by WENRA[8]

Levels of DiD		Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1		Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)	Normal operation
Level 2		Control of abnormal operation and failures	Control and limiting systems and other surveillance features		Anticipated operational occurrences
Level 3	3.a	Control of accident to limit radiological releases and prevent escalation to core melt conditions	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact	Postulated single initiating events
	3.b		Additional safety features, accident procedures		Postulated multiple failure events
Level 4		Control of accidents with core melt to limit off-site releases	Complementary safety features to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time	Postulated core melt accidents (short and long term)
Level 5		Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off-site radiological impact necessitating protective measures	-/-

The term "practical elimination" is applied almost exclusively in the context of situations leading to large or early releases (applicable also for spent fuel pool).

Safety objective O3 requires implementation of provisions to prevent accidents which would lead to early or large releases, either to make such accidents physically impossible to occur, or to demonstrate with a high degree of confidence that they are extremely unlikely to occur.

Accident sequences that are practically eliminated have a very special position in the DiD approach, because they are either physically impossible or provisions are taken to ensure that they are extremely unlikely to arise, and so the mitigation of their consequences does not need to be included in the design. The justification of the "practical elimination" should be primarily based on design provisions, where possible, strengthened by operational provisions (e.g. adequately frequent inspections). There is also consensus that practical elimination shall not be misused for not fully applying the defence-in-depth concept.

A specific accident sequence can be considered as practically eliminated:

- if it is demonstrated as physically impossible for the accident sequence to occur or
- if the accident sequence can be considered with a high degree of confidence to be extremely unlikely to occur.

This definition has been adapted from IAEA Safety Standard No. SSR 2/1 [12]. Consequently, WENRA is in line with the IAEA Safety Standards with respect to the general understanding of practical elimination. Currently, WENRA's RWHG develops a document discussing the regulators expectations for the demonstration of practical elimination. To minimise uncertainties and to increase the robustness of a plant's safety case, demonstration of practical elimination should preferably rely on the criterion of physical impossibility.

Probabilistic and deterministic elements both are required to demonstrate that an accident is extremely unlikely with a high degree of confidence. The compliance with a general cut-off probabilistic value should not be considered as a demonstration of the extreme unlikeliness and as a justification for not implementing reasonable design or operational measures.

It should be ensured that the provisions relied upon to demonstrate the extreme unlikeliness remain in place and valid throughout the plant lifetime.

Following WENRA recommendations, the "high degree of confidence" is translated into the necessity to evaluate the uncertainties associated with the data and methods in order to underwrite the degree of confidence claimed. Furthermore, appropriate sensitivity studies should be included to confirm that sufficient margins to cliff edge effects exist.

Measures to increase the reliability of engineered provisions contribute to the high degree of confidence.

The most stringent requirements regarding the demonstration of practical elimination should apply in the case of an event/phenomenon which has the potential leadoff leading directly to a severe accident, i.e. to pass from DiD level 1 to level 4.

It must be ensured that the practical elimination provisions remain in place and valid throughout the plant lifetime.

WENRA considers that both the DiD approach and the practical elimination of accidents with early or large release are fully applicable for fuel storage pools.

Safety objective O4 “Independence between all levels of DiD” seeks enhancement of the effectiveness of the independence between all levels of DiD, to provide as far as reasonably achievable an overall reinforcement of DiD.

The independence between structures, systems and components (SSCs) is realised by an adequate application of diversity, physical separation (structural or by distance) and functional isolation concepts.

WENRA calls for independence to the extent reasonably practicable between different levels of DiD and considers that independent SSCs for safety functions on different DiD levels shall have both of the following characteristics:

- the ability to perform the required safety functions is unaffected by the operation or failure of other SSCs needed on other DiD levels;
- the ability to perform the required safety functions, is unaffected by the consequences of the postulated initiating event, including internal and external hazards, for which they are required to function.

This is translated in the following requirements for SSCs on different DiD levels:

- SSCs fulfilling safety functions in case of postulated single initiating events (DiD level 3a) or in postulated multiple failure events (DiD level 3b) should be independent to the extent reasonably practicable from SSCs used in normal operation (level 1) and/or in anticipated operational occurrences (level 2). This independence means that no escalation of the failure of SSCs used in normal operation and/or in anticipated operational occurrences will lead to the impairment of a safety function required in the situation of a postulated single initiating event or of a multiple failure event.
- SSCs fulfilling safety functions used in case of postulated single initiating events (DiD level 3a) should be independent to the extent reasonably practicable from additional safety features used in case of postulated multiple failure events (DiD level 3b). For the safety analyses of postulated multiple failure events, credit may be taken from SSCs used in case of postulated single initiating events as far as these SSCs are not postulated as unavailable and are not affected by the multiple failure event in question; SSCs specifically designed for fulfilling safety functions used in postulated multiple failure events should not be credited for level 3a event analyses for the same scenario.

- Complementary safety features specifically designed for fulfilling safety functions required in postulated core melt accidents (DiD level 4) should be independent to the extent reasonably practicable from the SSCs of the other levels of DiD.

As part of DiD, DEC analysis shall be undertaken with the purpose of further improving the safety of the nuclear power plant by:

- enhancing the plant’s capability to withstand more challenging events or conditions than those considered in the design basis,
- minimising radioactive releases harmful to the public and the environment as far as reasonably practicable.

It should be noted that “further improving the safety” as stated in the reference levels does not refer to the concept of “continuous improvement”.

Within the analysis of DEC, cliff-edge effects should be identified and a sufficient margin to avoid them should be demonstrated wherever applicable.

There are two categories of design extension conditions (DECs):

- DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;
- DEC B with postulated severe fuel damage.

In DEC A, radioactive releases shall be minimised as far as reasonably practicable.

In DEC B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to allow sufficient time for protective actions (if any) in the vicinity of the plant, and to avoid contamination of large areas in the long term.

The DEC B analysis should cover a reasonable period of time, until a “controlled state” after severe accident is reached (decay heat removal is ensured, the damaged or molten fuel is stabilised, re-criticality is prevented and long term confinement is ensured to the extent that there is limited release of radioactive nuclides).

Article 8b of Council Directive 2014/87/Euratom requires a reduced impact of extreme external natural and unintended man-made hazards

According to WENRA, the new reactors are expected to be designed, sited, constructed, commissioned and operated in such manner that in case of […] “Accidents without core melt” (Objective O2), the core damage frequency should be reduced as far as reasonably achievable, taking into account all types of credible hazards and failures and their combinations.

WENRA requires that the threats from natural hazards and from combination of natural hazards shall be removed or minimised as far as reasonably practicable for all operational plant states. This means that all safety-relevant SSCs required to cope with an external hazard are designed and adequately qualified to withstand the conditions related to that external hazards. Moreover, the structural integrity of the spent fuel pools needs to be ensured in case of rare and severe external hazards.

External Hazards considered in the general design basis of the plant should not lead to a core melt accident (Objective O2 i.e. level 3 DiD).

Accident sequences with core melt resulting from external hazards which would lead to early or large releases should be practically eliminated (Objective O3, i.e. level 4 DiD).

Evaluation of natural hazard effects included in DEC analysis should, as far as practicable, demonstrate:

- sufficient margins to avoid “cliff-edge effects” that would result in loss of a fundamental safety function;
- sufficient resources available at multi-unit sites considering the use of common equipment, staff or services.

A margin to cliff-edge effects is defined as the difference between a design basis natural event, and a natural event at which the fundamental safety functions can no more be ensured.

Article 8b of Council Directive 2014/87/Euratom specifies the necessity of an operating experience reporting system. WENRA documents specify that the licence holder shall establish and conduct a programme to collect, screen, analyse, and document plant operating experience in a systematic way. The plant staff shall be required to report all abnormal events and be encouraged to report internally near misses relevant with regard to the safety of the plant. The plant operating experience records shall be evaluated to identify any latent safety relevant failures or potential precursors and possible tendencies towards degraded safety performance or reduction in safety margins.

Article 8c of Council Directive 2014/87/Euratom stipulates the attributes for the periodic safety review. WENRA recommends that the safety review shall be made periodically, at least every ten years, as an opportunity to review not only the conformity of the plant, but also to identify possible safety improvements. The WENRA safety objectives for new nuclear power plants should be used as a reference for identifying reasonably practicable safety improvements for “deferred plants” and existing plants during periodic safety reviews.

The regular review of the overall safety of the nuclear power plant should include the safety demonstration, taking into account operating experience, relevant research findings, and advances in science and technology.

The scope of the PSR shall be as comprehensive as reasonably practical as stated in IAEA SSG-25 [11] or in WENRA SRL P2.2 [21]. The areas that shall be covered as a minimum are derived from the 14 safety factors defined in the IAEA Safety Guide SSG-25 [10] and in WENRA SRL P2.2 [21]:

- Plant design;
- Actual condition of SSCs important to safety;
- Equipment qualification;
- Ageing;
- Deterministic safety analysis;
- Probabilistic safety assessment;
- Hazard analysis;
- Safety performance;
- Use of experience from other plants and research findings;
- Organization, the management system and safety culture;
- Procedures;
- Human factors;
- Emergency planning;
- Radiological impact on the environment.

The review shall use an up-to-date, systematic and documented methodology, taking into account deterministic as well as probabilistic assessments, it shall consider the cumulative effect on safety and identify what safety improvements are reasonably practicable.

It is considered important to use PSR method in all phases of operation including the decommissioning phase of NPPs and for other nuclear facilities, such as research reactors and radioactive waste management facilities.

The safety assessment requires special considerations for multi-unit sites and to address long-term measures as well as to cover all areas with significant amounts of radioactive material at the site.

According to IAEA SSG-25 [11] it is considered as a good practice to communicate results of the PSR to the general public, including resulting safety improvements. It is further emphasised that during the PSR process the regulatory body and/or its technical support staff should communicate with the operating organisation to clarify issues and acquire any necessary additional information.

As lessons learnt from the Fukushima Daiichi NPP accident, the following areas are recognised for improvements [42]:

- the timely and effective implementation of improvements derived from the PSR;
- review of site characteristics regarding external hazards;
- more explicit guidance on the need for the comprehensive analysis of all hazards and plant faults;
- taking into account multiple-unit issues.

5 Additional information from European Utility Requirements EUR

The EUR organisation of major European electricity producers was established in 1991. The main goal was to reduce licencing risk by common rules for NPPs with light water reactors by improved acceptance and safety harmonisation of standard designs. The main objective of the EUR is “*to harmonize and stabilize the conditions in which the standardised light water reactor nuclear power plants to be built in Europe in the first decades of the century will be designed and developed*” [23]. Thus, the EUR aim at improving both nuclear energy competitiveness and public acceptance in an electricity market unified at the European level. Furthermore, the EUR will contribute to promote a world-wide harmonisation of the design bases of the next generation of nuclear power plants.

EUR members are: CEZ group (Czech Republic), EdF (France), EdF Energy (UK), ENDESA (Spain), Energoatom (Ukraine), Fortum (Finland), GEN (Slovenia), IBERDROLA (Spain), MVM (Hungary), NRG (the Netherlands), Rosenergoatom (Russia), Tractebel (Belgium), TVO (Finland), Vattenfall (Sweden) and VGB (Germany).

The EUR have been prepared by owners and operators of NPPs but do not have a regulatory character. They cover design objectives and fundamental requirements as well as the assessment of compliance of designs of interest for EUR utilities. They are neutral regarding NPP design and can serve as possible basis for bids.

EUR documents consist of four volumes:

- Volume 1: Main policies and objectives
- Volume 2: Generic nuclear island requirements
- Volume 3: Application of EUR to specific projects
- Volume 4: Generic conventional island requirements

The EUR contain about 4,000 requirements in Volumes 1, 2, and 4. Volume 3 contains the comparison of selected LWR designs with EUR generic requirements. Detailed results are strongly restricted for open distribution, only main results are highlighted. The EUR documents have been revised in line with technology development, from Revision A to Revision D [23], which was issued in 2012. The last EUR Revision E [24] was issued in December 2016. EUR Volume 3 was applied on request to nine designs of NPPs for instance BWR90, EPR 1000, SWR 1000, ABWR, WWER AES 1200 and APR 1400.

Major changes in Revision D:

- recognition of five defence-in-depth levels;
- consideration of intentional aircraft crash;
- limited application of lesson learnt from the Fukushima accident;
- requirement of 60 years design lifetime;
- improvement of plant availability by shorter outages;
- consideration of grid faults and internal overcurrent events;
- better consideration of emergency operating procedures (EOPs);
- utilisation of full training and engineering simulators for V/V process of I/C; and
- incorporation of state-of-the art knowledge in decommissioning.

While drafting EUR Rev. E [24], the most recent documents from IAEA and WENRA were considered in order to harmonise the EUR with the expectations of the European regulators. The major improvements of Revision E are summarised in [36]:

- Changes in basic safety concept
  - Enhancing defence-in-depth concept
  - Introducing the term “practical elimination”
  - Description of acceptable deterministic safety analysis methods
- Fukushima lessons learnt
  - Strengthening plant autonomy
  - Creditability of non-permanent equipment
  - Emphasising ultimate heat sink and electric power supply
  - Extension of the scope of external hazards
  - Addressing spent fuel pool cooling in complex sequences
  - Renewed safety classification

In addition, the chapter on the design of instrumentation and control systems was revised to reflect recent developments in international standards as well as to consider lessons learnt from recent licensing processes in Europe.

EUR documents shall serve as reference technical documents for the development of new NPPs and for bidding processes of GEN 3 plants.

The EUR will continue to have influence on vendors and relations with regulators and to promote open competition within the European market through industrial standardisation and harmonisation.

The EUR will continue its contacts with ENISS (European Nuclear Installation Standard Safety), WENRA group of European regulators and IAEA as international organisation responsible for the preparation of safety standards for nuclear installations and activities.

Within this project, limited access to EUR Rev. E [24] was granted by the EUR. As the EUR are not endorsed by the competent nuclear regulatory authorities, this study uses the EUR only as a complementary source of information.

The EUR provide qualitative radiological criteria for DECs which are referred to as criteria for limited impact (CLI). Vidard and Bassanelli [37] discussed quantitative CLI in the context of ICRP Recommendation No. 63 and the IAEA Basis Safety Standards [39].

Table 5 provides an overview of the protective actions as function of distance and projected doses according to EUR Revision D [23] and Basic Safety Standards [39]. However, IAEA published a revised version of the Basic Safety Standards (GSR Part 3) [40] in 2014. In the new IAEA Safety Standard GSR Part 3, the intervention levels have also been updated.

Nevertheless, the paper of Vidard and Bassanelli describes an approach to derive quantitative radiological acceptance criteria for the safety demonstration of accident conditions more severe than design basis accidents. In this approach, the intervention levels for various emergency response measures are used to define a dose limit. It has to be demonstrated that the releases e.g. in case of severe accidents, will not exceed these dose limits. In addition, the distance from the reactor is taken into account.

Table 5: Overview of criteria for limited impact (CLI) as described in [37][38].

<table>
<tr><th></th><th>Up to 800 m</th><th>Limited action zone < 3 km</th><th colspan="3">Beyond 3 km radius</th></tr>
<tr><td rowspan="2">Permanent relocation</td><td rowspan="2">May be needed</td><td>No</td><td colspan="3">No</td></tr>
<tr><td colspan="4">< 100 mSv in 50 years</td></tr>
<tr><td rowspan="2">Evacuation</td><td rowspan="2">May be needed</td><td>No</td><td colspan="3">No</td></tr>
<tr><td colspan="4">< 50 mSv dose in 7 days[3]</td></tr>
<tr><td rowspan="2">Temporary relocation</td><td rowspan="2">May be needed</td><td rowspan="2">May be needed</td><td colspan="3">No</td></tr>
<tr><td colspan="3">< 30 mSv in 30 days</td></tr>
<tr><td rowspan="2">Restricted trading of foodstuff[4]</td><td colspan="3">Only for 1st year</td><td>Only for 1st month</td><td>No</td></tr>
<tr><td colspan="3">< 5 mSv/a from ingestion after 1 year</td><td>< 5 mSv/a from ingestion after 1 month</td><td>< 5 mSv/a from ingestions</td></tr>
<tr><td></td><td>0 m</td><td>800 m</td><td>3 km</td><td>10 km</td><td>100 km</td></tr>
</table>

In the process of establishing EUR Rev. E [24], the radiological criteria were revised. Qualitative radiological acceptance criteria were substantiated by quantitative radiological acceptance criteria provided either in doses (in mSv) or in terms of radioactive releases (in TBq). The acceptance criteria are provided for difference plant states. Whereas EUR Rev. D [23] discusses criteria for restricted trading of foodstuff and economic impact, Rev. E defines only criteria for restrictions in food consumption. An overview of the revised radiological acceptance criteria as provided in [24] is shown in Figure 2.

3 Projected dose rate for evacuation as proposed in IAEA GSR Part 3.

4 The area limits for restricted trading of foodstuff differs in literature. The shown radii of 10 km and 100 km correspond to information from [37], whereas in [36] smaller radii corresponding to areas of 10 km² and 30 km² are noted.

Plant state	Qualitative radiological criteria		Quantitative radiological criteria		
DBC 3	No, or only minor off-site radiological impact	Very limited restrictions on foodstuff consumption	< 1 mSv	Ground level release 4.4 TBq ^{131}I 0.5 TBq ^{137}Cs	Elevated level release 73 TBq ^{131}I 7.9 TBq ^{137}Cs
DBC 4			< 5 mSv		
Complex sequences	No, or only minor off-site radiological impact		< 10 mSv		
Severe accidents	No long term action	> 800 m	< 100 mSv 50 years after termination of release		
	No evacuation	> 3 km	< 50 mSv in 7 days		
	No sheltering	> 5 km	< 10 mSv in 48 hours		
	No iodine prophylaxis		< 10 mSv		
	Limited food restrictions		Release 2,800 TBq ^{131}I 400 TBq ^{137}Cs		

Figure 2: Overview of radiological acceptance criteria proposed by EUR Rev. E [23].

6 Concepts and definition in selected national regulatory frameworks

In addition to the analysis of the available international and European Guidance documents, the applied defence-in-depth concepts and definitions used in selected European countries (Czechia, France, Germany, Hungary, Lithuania, Romania, and Slovakia) were compared. This analysis provides further insights into a harmonised usage of the applied concepts in the different Member States and could also provide additional input for the preparation of the questionnaire for the technical survey on practical approaches to implement objectives defined in Articles 8a to 8c within Task 2. Information on the seven countries can be found in Annex 1 to Annex 7 of this report.

Defence-in-depth concept

The selection represents countries with existing nuclear power plants in operation, but also includes countries with new build projects (France, Hungary and Lithuania) as well as countries with specific reactor designs (e.g. Romania with CANDU reactors).

It can be summarised that a common understanding of the defence-in-depth concept exists among the countries selected for the analysis. The defence-in-depth concepts applied for existing reactors comply with the defence-in-depth concept described in the WENRA Safety Reference Levels [21]. It was observed that minor differences exist. For example, Germany splits level 4 of defence-in-depth in three sub-levels. Countries with new build projects (France, Hungary and Lithuania) apply the defence-in-depth concept as described in the WENRA report “Safety of new NPP designs” [8]. Some of these countries also distinguish the concepts for existing and new reactors as it is the case in Hungary. Romania, operating Canadian design reactors (CANDU) adapted the defence-in-depth concept of the vendor country. Based on this limited survey it can be concluded that a common understanding of implementing the defence-in-depth concept exists amongst these countries. Further, it can be concluded that for new nuclear power plants, the more advanced defence-in-depth concept as described in the WENRA report “Safety of new NPP designs” is applied. For existing reactors, the defence-in-depth concept as described in the WENRA Safety Reference Levels are applied.

“to prevent”, “to avoid” and “to mitigate”

In all seven countries, a harmonised application of the terms “to prevent” and “to mitigate” can be observed. In the applied safety concepts, “to prevent” is used in the context that a further escalation to more severe plant states will not occur. “to mitigate” is used to describe the measure to deal with consequences of a certain accident condition, whereas “to prevent” is also used to describe the avoidance of a transition from operational states to accident conditions and “to mitigate” is only used in the context of accident conditions.

Early release

All seven countries apply the qualitative description of an early release. This means, a release occurs in a magnitude requiring urgent protective measures in the vicinity of a nuclear installation, but with insufficient time to effectively implement the necessary off-site emergency response measures.

Large release

Most of the seven countries have a qualitative definition of the understanding of a large release. The common understanding is that a large release is a release which would require off-site protective actions which cannot be limited in area and time. A few countries apply quantitative criteria (e.g. Lithuania and Romania) based on activity levels or doses. Lithuania defines a large release as a release of more than 100 TBq ^{137}Cs or other radionuclides which corresponds to an activity level of more than 100 TBq ^{137}Cs after three months. Romania has established a relationship of frequency and dose to the public in the exclusion zone. It can be summarised that differences in the practical implementation of the demonstration that large releases are avoided exists amongst the Member States. The comparison of the selected countries already reveals that qualitative criteria but also quantitative criteria in terms of activity levels or doses are applied.

Reasonably practicable

The short review of the selected seven countries shows that more information from a technical point of view is required. Safety improvements are considered as reasonably practicable if they can contribute to a further reduction of risk by safety measures in line with the progressing state of the art in science and technology. It was emphasised that available technologies, recognised good practices and insights from research and development have to be considered. The need to assess the contribution to risk reduction was also raised.

7 SIMILARITIES AND DIFFERENCES IN INTERNATIONAL GUIDANCE DOCUMENTS

Relevant publications from INSAG, IAEA and WENRA were analysed with the aim to identify if these documents provide sufficient guidance to support European regulators in implementing Articles 8a to 8c of Council Directive 2014/87/Euratom [25].

With respect to the nuclear safety objective expressed in Article 8a it can be summarised that the objective is in line with the radiological objectives expressed in IAEA SSR 2/1 [12], the WENRA safety objectives for new nuclear power plants [7] and the WENRA Safety Reference Levels for existing reactors [21]. Whereas the report INSAG-10 [4] does not explicitly discuss the radiological safety objective, the IAEA and WENRA express a clear position that in case of new reactors, sequences leading to a large or early release have to be practically eliminated by design. It can be further stated that the formulation used in the WENRA safety objectives for new nuclear power plants [7] and in IAEA SSR 2/1 [12] are stricter than the wording in Article 8a of Council Directive 2014/87Euratom [25]. Whereas WENRA and the IAEA require the practical elimination of early or large releases, Council Directive 2014/87Euratom requires only its avoidance. Both organisations adapt this objective also for existing reactors as a reference. In contrast to new reactors, this objective has to be achieved by a systematic approach to analyse design extension conditions and implementation of backfittings supplemented by procedures and guidelines for accident management. In addition, WENRA published clear qualitative expectations to interpret the meaning of a large release (see Table 3) [8].

WENRA as well as the IAEA separate hazards from postulated initiating events. It is discussed in the WENRA report on “Safety of New NPP designs” [8], the WENRA Safety Reference Levels [21], IAEA Safety Standard Series No. SSR 2/1 [12] and in TECDOC 1791 [16] that depending on the impact of a certain hazard, an event can be initiated. A common view is expressed that up to a design basis hazard, the impact shall not lead to a failure of the safety systems to control design basis accidents or postulated single initiating events. Being aware that the impact of a hazard might be higher than the impact assumed for the design basis, the safety expectation is to avoid sequences leading to large or early release. WENRA requires in its Safety Reference Levels Issue T [21] a systematic approach to identify events more severe than the design basis events as part of the DEC analysis. In particular, it is demanded to demonstrate sufficient margins to avoid “cliff-edge effects” that would result in a loss of a fundamental safety function and the identification and assessment of the most resilient means for ensuring the fundamental safety functions. It should be considered that events could simultaneously challenge several redundant or diverse trains of a safety system, multiple SSCs or several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures. Licence holders are expected to demonstrate that sufficient resources remain available at multi-unit sites, considering the use of common equipment or services. In contrast to WENRA, the IAEA only requires that the design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site [12]. Up to now, the implications of the above-mentioned requirement have not yet been formally addressed in any safety standard of the IAEA [16].

Differences exists in the interpretation of acceptable radiological consequences as expressed in the new draft of IAEA DS491 “Deterministic Safety Analysis for Nuclear Power Plants” [14] and in the WENRA report “Safety of new NPP designs” [8]. These differences are mainly due to the different assignment of DEC without fuel melt (IAEA DS491) and postulated multiple failure events (WENRA) to level 4 of defence-in-depth and level 3.b of defence-in-depth, respectively. As a consequence, the radiological objectives defined by WENRA for this category of accident conditions are more stringent than those of the IAEA. In practice, this may result in different technical solutions to demonstrate compliance with the formulated radiological objectives.

7.1 Summary of main findings

After reviewing international and European guidance documents, the following findings can be summarised.

Article 8b of Council Directive 2014/87/Euratom corresponds to the objectives for the first four levels of defence in depth as described in INSAG-10. In addition, Article 8b requires the minimisation of the impact of extreme external natural and unintended man-made hazard.

With respect to new nuclear power plants, the defence-in-depth concept addressed in Article 8b of Council Directive 2014/87/Euratom is fully in line with the expectations expressed in INSAG-10 and IAEA SSR 2/1. For nuclear power plants, WENRA provides guidance [8] in which the defence-in-depth concept is elaborated in more detail.

Article 8b of Council Directive 2014/87/Euratom requires in general a defence-in-depth concept for nuclear installations, which is fully in line with the expectations expressed in INSAG-10 and IAEA SSR 2/1. As it applies to all nuclear installations, the concept needs to be detailed in further guidance documents. For nuclear power plants, WENRA has elaborated a more detailed defence-in-depth concept [8] with six levels of defence-in-depth (1, 2, 3a, 3b, 4 and 5) addressing five different plant states (normal operation, anticipated operational occurrences, postulated single initiating events, postulated multiple failure events and postulated core melt accidents). In particular, the group of multiple failure events on level of defence in depth 3b is introduced. The authors of this study consider the defence-in-depth concept developed and published by WENRA (see [8]) as the most advanced description of the defence-in-depth concept for new nuclear power plants and as currently the most advanced description of the defence-in-depth concept for new nuclear power plants which would serve to interpret Article 8b.

The term “to avoid” is frequently used to express a radiological safety objective and is applied in the context of emergency preparedness and response. The term “to control” is frequently used to express that a certain plant state can be managed with the safety features provided on the respective level of defence in depth. This term is internationally used in a fully harmonised manner.

There is a common understanding that “large releases” would require off-site counter measures, which are not limited in area and time. Most internationally available documents do not further discuss this point. It can be concluded that no quantitative values are provided in either international or European documents to provide further guidance for the national regulators in implementing a harmonised meaning of this term. At least WENRA published an overview of the acceptable off-site countermeasures as function of distance (see Table 3). EUR Rev. E describes an approach to determine quantitative radiological acceptance criteria taking intervention levels for emergency response countermeasures into account. A short analysis of the practices in seven selected countries revealed that besides the definition of qualitative radiological objectives, some countries also apply quantitative targets like maximum activity levels or doses. In addition, the revised IAEA Safety Standard SSG-2 Rev. 1 [15] requires that radiological acceptance targets for design extension conditions should be defined in terms of effective dose, equivalent dose or dose rate or may be converted in acceptable activity levels for different radionuclides in order to decouple nuclear power plant designs from site-specific conditions. With respect to further harmonisation in Europe it is recommended to discuss this topic amongst the Member States with the aim to increase the mutual understanding of applied radiological acceptance criteria. Further discussions on the approach to demonstrate compliance with the defined qualitative or quantitative radiological acceptance criteria will be beneficial.

8 PROPOSAL FOR FUTURE ACTIVITIES

Table 6 summarises the main findings from the analysed international and European guidance documents.

Meaning of “to prevent” and “to mitigate”

Ambiguities have been observed in the meaning and application of the terms “to prevent” and “to mitigate”. From the analysed documents, two positions can be clearly identified:

- **Position 1:** A clear distinction is made between prevention and mitigation at the on-set of the core melt. Prevention describes all measure to prevent the occurrence of a core melt accident. Mitigation describes all measure to minimise the radiological consequences in case of a core melt (see for example IAEA Safety Standard Series No. NS-G-2.15 [41]).

- **Position 2:** The term prevention is used in a manner to describe that the escalation to more severe accident conditions (i.e. escalation to a higher level of defence in depth) will not occur. Mitigation describes the measures to reduce the consequences on the respective level of defence in depth.

With respect to position 2 it can be concluded that usually the term “to control” is commonly applied to describe the actions on a certain level of defence in depth to deal with the conditions for a specific plant state. The term “to avoid” is usually used in a broader context. It is proposed to discuss the meaning of the terms “to avoid”, “to prevent”, “to control” and “to mitigate” in the context of the defence-in-depth concept.

This topic has been discussed with the participants during the 1st workshop. It was concluded that a common understanding of how to apply the defence-in-depth concept and the meaning of the terms under discussions exists. Escalation to more severe plant states is achieved by prevention. Mitigation describes the measures to minimise the radiological consequences during accident conditions (i.e. design basis accidents / postulated single initiating events, design extension conditions without fuel melt / postulated multiple failure events and design extension conditions with fuel melt / postulated core melt accidents).

There was a consensual view that the available guidance documents provide sufficient guidance for the Member States:

- WENRA RHWG Report on Safety of new NPP Designs [8];
- IAEA TECDOC 1791 “Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants” [8];
- INSAG-10 “Defence in Depth in Nuclear Safety” [4] and
- INSAG-12 “Basic Safety Principles for NPP” [5].

It is further noted that in the near future, additional guidance will be available, as IAEA is currently developing a safety guide on “Assessment of the Application of General Requirements for Design of Nuclear Power Plants” [20]. In this new safety guide, the assessment of the implementation of defence-in-depth as well as the assessment of practical elimination of event sequences that would lead to an early radioactive release or large radioactive release will be discussed in more detail.

Acceptance criteria for large releases

In the aftermath of the accident at the Fukushima Daiichi NPP, discussions have been started on recommendations to protect the public in the event of a severe accident in Europe. In 2014, HERCA and WENRA published a common approach to harmonise the off-site protective actions during the early phase of a nuclear accident [35]. It was recognised that each European state has developed arrangements to protect the public in the event of a severe nuclear accident. However, due to the freedom to define intervention values and countermeasure on a national basis, sometimes significant differences exist. Especially in Europe a nuclear accident could lead to cross border issues. Thus, a necessity was seen to align protective actions along adjacent national borders to issue harmonised recommendations to the public on both sides of the border. In contrast to the field of emergency preparedness and response as addressed in the common HERCA / WENRA paper, the situation with respect to the safety demonstration for new reactors is somewhat different. Here, the applicant has to demonstrate that in the case of a postulated core melt accident only accepted protective actions may be necessary. **This means that clear quantitative acceptance criteria are necessary which can be compared with the results of the safety analyses. In addition, it may be necessary to harmonise the methodology of the radiological consequence analysis to ensure comparable results across Europe.**

Table 6: Synopsis of terms applied in selected international documents.

	INSAG	IAEA	WENRA	EUR	Category
'to avoid'	Not used	Used in the context of emergency preparedness and response	Frequently used in various contexts in the WENRA Safety Reference Level.		Ambiguous guidance
'to prevent'	• No further escalation to higher levels of defence in depth • All measures before onset of core melt	• No further escalation to higher levels of defence in depth	• No further escalation to higher levels of defence in depth		Harmonised guidance
'to mitigate'	• Minimisation of radiological consequences	• Minimisation of radiological consequences	• Minimisation of radiological consequences		Harmonised guidance
'to control'	Applying provisions provided on a certain level of defence in depth.	Applying provisions provided on a certain level of defence in depth.	Applying provisions provided on a certain level of defence in depth.		Harmonised guidance
'early release'	Only for new NPP designs the term 'large release due to an early containment failure' is defined.	A release of radioactive material for which off-site protective actions are necessary but are unlikely to be fully effective in due time	A release that would require off-site emergency measures but with insufficient time to implement them		Harmonised guidance (only qualitatively)
'large release'	Not defined.	A release of radioactive	Situations that would	Acceptable off-site	Harmonised guidance

	INSAG	IAEA	WENRA	EUR	Category
		material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment.	require protective measures for the public that could not be considered as limited in area or time. Acceptable off-site countermeasures as function of distance are proposed.	countermeasures as function of distance are proposed. Dose limits are established by links to intervention levels for protective actions.	(only qualitatively)
‘reasonably practicable’	Not defined.	Not defined	see Ref. [9]		No guidance available (except from WENRA)
‘timely implementation’	Not defined.	Not define	see Ref. [9]		No guidance available (except from WENRA)

Acceptance criteria for “early release”

“Early release” is commonly defined as a release which will not allow for a proper implementation of mitigative countermeasures before the release happens. For the safety demonstration, the site-specific boundary conditions to implement countermeasures (such as sheltering, iodine prophylaxis, evacuation, etc.) have to be taken into account as well as the plant-specific accident scenarios. In the safety demonstration it should be demonstrated that the release will not happen before the countermeasures have effectively been implemented.

Reviewing the international and European documents revealed that no further guidance is available on how to demonstrate that early releases are avoided. During the 1st workshop, it was discussed that sharing the different approaches in the Member States would be desirable to foster a mutual understanding of the different approaches applied. This could be considered as a good starting point for developing a guidance on common principles for such a demonstration in the future.

“reasonably practicable” and “timely implementation”

The most advanced guidance on this topic can be found in the WENRA Guidance “Article 8a of the EU Nuclear Safety Directive: ‘’Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants’’”. In this paper, it is clearly expressed that “reasonably practicable” as well as ‘timely implementation’ is judged on a case-by-case basis.

It is proposed to discuss the matter in more detail. In general, “reasonably practicable” is the result of a decision-making process involving the licence holder as well as the regulator. This process should start with the safety objective to be achieved, a thorough investigation of the technical solutions available, the impact on the design of the existing facility, further implications of the implementation of the identified safety improvement and the final judgement of the regulator. A process may be developed which ensures a harmonised decision-making process in the Member States.

The term “timely implementation” first describes a requirement with respect to the attitude of the licence holder and the regulators. It expresses the expectation that all involved parties will not unduly delay the implementation of a reasonably practicable safety improvement. It is therefore first of all a prerequisite for the safety culture of all parties involved. Secondly, it can be discussed from the technical point of view. Here, the licence holder may propose a time schedule for the implementation of a safety improvement, which can be reviewed and assessed by the regulator. Based on the review and assessment of the proposed timeline, the timely implementation can be judged. After licence holder and the competent regulatory authority have agreed on the timeline, it is very important to follow up the activities to ensure the timely implementation in accordance with the proposed time schedule.

As a result of Task 1, the following three suggestions for further activities were proposed:

Proposed suggestion 1.1:

"*Sharing the different approaches in the Member States would be desirable to foster a mutual understanding of the different approaches applied to demonstrate avoidance of "early releases*."

Proposed suggestion 1.2:

"*Foster mutual understanding of demonstrating avoidance of "large releases" by applying different (quantitative/qualitative) approaches.*"

Proposed suggestion 1.3:

"*Sharing approaches to enhance mutual understand of regulatory decision making on 'reasonably practicable'.*"

A final discussion of the proposed suggestions took place during the 2nd workshop considering also the proposed suggestions resulting from Task 2 and Task 3. The final suggestions are described in Chapter 7 of Volume 1 "Summary Report".

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[23] European Utility Requirements for LWR Nuclear Power Plants, Revision D, October 2012

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[27] BMU, “Safety Requirements for Nuclear Reactors”, Federal Gazette BAnz AT 30.03.2015 B2, 30 March 2015

[28] National Report under the Convention on Nuclear Safety, Seventh Revision (August 2016)

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[32] NSN-17 – Nuclear safety requirements on ageing management for nuclear installations (2016)

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ANNEX 1 – CZECHIA

New nuclear legislation was issued during the time period 2016 / 2017. It replaced completely the old legislation which was issued in 1997. The new legislation includes Act on peaceful utilisation of nuclear power and ionising radiation No. 263/2016 Coll. and 17 implementing regulations for all areas of responsibility of the Czech regulatory body-the State Office for Nuclear Safety (SÚJB). During the preparation of the new Czech legislation new or revised documents of the IAEA, WENRA, ICRP and EU were taken into account. The Atomic Act and all SÚJB regulations are available in English language on SÚJB web site www.sujb.cz.

The topics of the project are covered mainly by the following documents:

- Atomic Act No.263/2016 Coll. (general safety requirements);
- SÚJB Decree No. 162/2016 on safety assessment (deterministic, probabilistic, periodic, regular, specific);
- SÚJB Decree no. 21/2017 on assurance of nuclear safety of nuclear facility (safety requirements during operation);
- SÚJB Decree No. 329/2017on design requirements of nuclear installation (application of the defence-in-depth, safety functions, design basis, DEC, etc.).

The accident at the Fukushima Daiichi NPP raised new safety requirements, methods of safety assessment and also new terms; but not all of them were sufficiently defined.

SÚJB management decided in 2017 to initiate the preparation of Safety Guides (SG) which are not legally binding documents, but they could facilitate the technical dialogue between regulator and applicant / licensee. More than 20 safety guides were proposed for preparation.

The terms “reasonably practicable”, “timely implementation” and “practically elimination” are used in the text of Atomic Act and implemented regulations.

The following definitions are part of new Czech legislation:

- practically eliminated means phenomenon, state, conditions or event which occurrence is physically impossible, or which is with high credibility very improbable;
- basic safety function means safety function which assures fulfilment of safety usage of nuclear energy in compliance with Atomic Act;

- safe state of nuclear facility means that fulfilment of basic safety functions is assured for long term;
- normal operation of nuclear facility means that limits and conditions for safe operation are followed;
- abnormal operation of nuclear facility means that deviation from normal operation occurred in nuclear facility which is able without repair to return to normal operation state;
- accident conditions mean that nuclear facility which is not in normal or abnormal operation state;
- design basis accident means accident conditions during which correct function of safety system is assured and corresponding reference levels or exposure limits are not exceeded;
- design extension conditions mean accident conditions with more severe scenarios than design basis accident which are part of design of nuclear facility;
- severe accident means accident conditions during which serious damage of active core and irreversible loss of its structure occurred;

The design of a nuclear facility has to fulfil the following safety goals:

- prevention of accident conditions
- mitigation of accident conditions, if occurred assurance that practically eliminated are radiation accidents during which no sufficient time for introduction of urgent protection measures for population is available (early release) and radiation accidents during which urgent protection measures for population cannot be limited from site and time point of view (large release).

But the term “release” is not used in legislation

In SÚJB regulation on safety assessment the following numerical values are provided: release of more than 1 % of ^{137}Cs inventory during less than 10 hours after accident declaration

New term “reasonably practicable” (ALARP) has been introduced by Council Directive 2014/87/Euratom which is analogous to ALARA for radiation protection. Its application in practice is much more complicated than in ALARA cases since data for application of cost / benefit analysis (CBA) are not available. For practical implementation of optimisation SÚJB recommends detailed analysis and evaluation of

- available technologies;
- good practice;
- application of the best accessible technologies for reasonable price etc.

It was proposed that the project will provide the definitions of following terms: “prevention”, “mitigation”, “avoid” and “defence-in-depth”.

For the first three terms official definition is not provided, since they have been used as terms of standard notion according to the Oxford dictionary.

Term “defence-in-depth” was introduced many years ago by the IAEA Safety glossary. Definition has been widely accepted by nuclear community Member States (including the Czech Republic) and international organisations. The detailed discussion of “defence-in-depth” has been provided in the INSAG publication Defence-in-depth in nuclear safety [4].

Need of definition is confirmed by the WENRA Guidance on Article 8a of EU Directive “Timely implementation of Reasonably Practicable Improvements to Existing NPPs”, which was elaborated by ad-hoc group to WENRA in June 2017. Document confirms that term has been defined in the IAEA document Safety Fundamentals by the following wording “The safety measures that are applied to facilities and activities that give rise of radiation risk are considered optimised if they provide the highest level of safety that can reasonably be achieved throughout the lifetime of the facility or activity without unduly limiting its utilisation”.

The IAEA TECDOC 1791 “Considerations on the applications of the IAEA safety requirements for the design of nuclear power plants” [16] can provide users assistance. It defines plant states, defence in depth strategy and requirements on independence of levels of defence in depth, concept of practical elimination, cliff edge effects and safety margins, design for external hazards as well as reliability of heat transfer to ultimate heat sink. The document contains Appendices with valuable practical examples.

ANNEX 2 – FRANCE

The Order of 7 February 2012 setting the general rules relative to basic nuclear installations – with complementary guidance provided in a recently published guide on the Design of Pressurized Water Reactors – requires that the licensee applies the principle of defence in depth, which consists in deploying sufficiently independent levels of defence aiming at:

- first level: preventing incidents;
- second level: detecting incidents and applying measures that will firstly prevent them from leading to an accident, and secondly restore a situation of normal operation or, failing this, place and maintain the installation in a safe condition;
- third level: controlling accidents that could not be avoided or, failing this, limit their aggravation by regaining control of the installation in order to return it to and maintain it in a safe condition; this level is split between level 3a - prevention of fuel meltdown in the “design reference envelope” - and level 3b - prevention of fuel meltdown in the “design extension envelope”, consisting in more complex sequences of events;
- fourth level: managing accident situations that could not be controlled so as to mitigate the consequences, especially for humans and the environment.

Furthermore, a fifth level of defence in depth targeting emergency management by the public authorities aims at limiting the radiological consequences of radioactive releases that could result from accident conditions. Specific design measures shall be planned for in this respect

These levels of defence shall be sufficiently independent to meet the installation objectives.

Regarding NPP, the recently published guide recommends the adoption of safety objectives consistent with the Council Directive 2014/87/Euratom [16] and WENRA ones [8]. First, the number of incidents and the possibilities of accidents occurring shall be minimised and, in the event of incidents or accidents, the releases of radioactive or hazardous substances or the hazardous effects, and their impacts on human and the environment, shall be limited to levels that are as low as practicable. Moreover:

- for accidents without fuel meltdown, the radiological consequences shall not lead to the need to implement population protection measures;
- accident situations with fuel meltdown which could lead to significant radioactive releases that develop too rapidly to allow deployment of the necessary population

protection measures in due time shall be rendered physically impossible or, failing this, extremely unlikely with a high degree of confidence

- the population protection measures that would be necessary in the event of the other accidents with fuel meltdown shall be very limited in terms of extent and duration.

Considering the safety objectives and the principle of DiD mentioned above, large releases and early releases should be avoided:

- by defining provisions allowing to significantly limit the consequences of severe accident situations when feasible;
- and by "practically eliminating" severe accident situations where it appears to be impossible to define such provisions or to demonstrate their adequacy with the knowledge and techniques available at the time of the design orientations.

The justification of "practical elimination" shall preferably rely on the physical impossibility of the situation. Where this is not possible, the applicant shall demonstrate with a high degree of confidence that the situation is extremely unlikely.

The NPP recently published guide – and its corresponding objectives and recommendations – may be used, for reference, to seek improvements to be made to reactors in operation, for example during their periodic safety reviews, in accordance with article L. 593-18 of the Environment Code and articles 8a and 8c introduced by the European Directive of 8th July 2014. Indeed, article L. 593-18 of the Environment Code requires that the licensee carries out periodic safety review of its installation taking the best international practices into consideration.

ANNEX 3 – GERMANY

In the recently published “Safety requirements for Nuclear Power Plants” the defence in depth concept is implemented in the German regulatory framework. The German defence in depth concept is dedicated to existing nuclear power plants, as new builds are prohibited by the German Atomic Energy Act.

The DiD concept is concretized in the German “Safety Requirements for Nuclear Power Plants” and follows the traditional concept of five levels of defence in depth. As a particularity level 4 of defence in depth is split into three sublevels:

- Level 4a: Anticipated transients without scram (ATWS);
- Level 4b: Events with multiple failures of safety systems;
- Level 4c: Severe accidents with core melt.

[27] gives an overview of the levels of defence in depth, the assigned plants states and the objectives for each level of defence in depth. In the German “Safety Requirements for Nuclear Power Plants” both terms ‘to prevent’ and ‘to avoid’ are used. Although the use of the terms ‘to avoid’ and ‘to prevent’ is due to the translation of two German words without an adequate English translation, there is a specific meaning assigned to both terms. Also the glossary of the German “Safety Requirements” contains definitions for both terms. This definition reads as follows:

> “***Avoidance (to avoid)***
> *The approach of avoiding events or event sequences can apply to the case if higher level designed measures and equipment (on a subsequent level of defence) are available for their management in the reliability and effectiveness required. By this means, it has to be ensured that the occurrence of such events or event sequences on level of defence 3 is not to be expected during the operating life-time of the plant, but which have to be postulated in any case.*”

> “***Prevention (to prevent)***
> *Events or event sequences for whose control there are no higher level designed measures or equipment on a subsequent level of defence shall be prevented. Thus, the progression of events and event sequences on level of defence 3 to level of defence 4 shall be prevented.*”

In addition, a clear understanding of prevention and mitigation is expressed in the German regulatory framework. Prevention comprises all measures and provisions in place before the onset of severe fuel degradation (i.e. core melt accident). Mitigation describes all measures to minimize the radiological consequences in case of a severe accident.

Table 7: Defence in depth concept based on German "Safety Requirements for Nuclear Power Plants"

<table>
<tr><th colspan="2">Level of defence in depth</th><th>Plant State</th><th>Objective</th></tr>
<tr><td colspan="2">1</td><td>Normal operation</td><td>To avoid abnormal operation</td></tr>
<tr><td colspan="2">2</td><td>Anticipated operational occurrences</td><td>To control abnormal operation and to avoid the occurrence of accidents</td></tr>
<tr><td colspan="2">3</td><td>Accidents</td><td>To control accidents and to prevent events involving the multiple failure of safety equipment</td></tr>
<tr><td rowspan="3">4</td><td>4a</td><td>Very rare events (ATWS)</td><td>To control very rare events</td></tr>
<tr><td>4b</td><td>Accidents involving severe fuel assembly damages</td><td>Avoiding severe fuel assembly damages by preventive measures of plant internal accident management.</td></tr>
<tr><td>4c</td><td>Accidents involving severe fuel assembly damages.</td><td>To maintain the integrity of the containment for as long as possible, excluding or limiting releases of radioactive materials into the environment (early or large releases) and achieving a long-term controllable plant state by mitigative measures of the plant internal accident management in case of accidents involving severe fuel assembly damages.</td></tr>
<tr><td colspan="2">5</td><td>Disaster control measure</td><td>Minimizing radiological consequences by off-site countermeasures (e.g. sheltering, evacuation, iodine-prophylaxis, re-settlement)</td></tr>
</table>

In the German “Safety Requirements for Nuclear Power Plants” the meaning of ‘to avoid’, ‘to prevent’ and ‘to exclude’ is used with increasing strength. For legal reasons the German regulations use ‘to exclude’ instead of ‘practical elimination’ but aiming for the same objective.

The meaning of early and large releases is defined in para. 2.5 (1) of the German “Safety Requirements for Nuclear Plants”: An ‘early release’ is a release of radioactive materials into the environment of the plant, caused by the early failure or bypass of the containment and requiring measures of the external accident management for the implementation of which there is not sufficient time available. A large release is a release of radioactive materials into the environment of the plant requiring wide-area and long-lasting measures of the external accident management.

With respect to ‘reasonably practicable’ safety improvements, § 7d of the German atomic energy act requires a further precaution against risks. During the lifetime of the nuclear power plant the licence holder is obliged to realize safety measures according to the ongoing state-of-the-art of science and technology which are developed, suitable and adequate for providing not only an insignificant contribution to further precaution against risks for the public.

ANNEX 4 – HUNGARY

The defence in depth concept is implemented in the Hungarian regulatory framework. Requirements for nuclear facilities are in the Government Decree 118/2011 (VII. 11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities. Latest version with only minor modification in the text regarding defence in depth concept of the decree came into force on 11 of April 2018.

Eleven Annexes are connected to the Govt. Decree:

1. Nuclear safety authority procedures of nuclear facilities
2. Management systems of nuclear facilities
3. Design requirements for operating nuclear power plants

3a. Design requirements for new nuclear power plant units

4. Operation of nuclear power plants
5. Design and operation of research reactors
6. Interim storage of spent nuclear fuel
7. Site survey and assessment of nuclear facilities
8. Decommissioning of nuclear facilities
9. Requirements for the design and construction period of a new nuclear facility
10. Nuclear Safety Code definitions

The terms ‘early release’, ‘large release’, ‘reasonably achievable’ and ‘Defence in Depth’ are defined in Annex 10 “Nuclear Safety Code Definitions”. These definitions read as follows:

> “***Large release****: A radioactive release where off-site protective measures cannot be limited in space and time.*”

> “***Early release****: Radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction.*”

> “***Reasonably achievable****: Such a degree of actions that considers the present standards of science and technology, while graded to the severity of the different risks and undesirable consequences, which is determined by the authority based on the proposal of the licensee.*”

> “***Defence In Depth****: Multi-level defence, which is a hierarchically structured system of engineering solutions, nuclear safety principles and measures which*

guarantee the expected level of nuclear safety. On the physical level an important part of this system is the system of multiple barriers."

Article (7) of the above mentioned Government Decree declare that the release of radioactive materials into the environment shall be prevented by the application of defence in depth, and it shall be ensured that failures or the combination of failures resulting in accidents resulting in significant radioactive material discharges may only occur with adequately low probability.

The five levels of defence in depth defence are the following:

a) prevention of deviations from normal operational conditions and faulty actuations;

b) detection of abnormal operating conditions and prevention that the anticipated operational occurrences become design basis accidents;

c) management of design basis accidents according to pre-determined procedures;

d) termination of accident and severe accident processes and mitigation of their consequences;

e) in the case of a significant release of radioactive materials, mitigation of radiological consequences;

The most important components are the design solutions applying the appropriate safety margins, implementation and operation to a high standard; application of regulatory, limiting and protection systems and testing and monitoring solutions as well as documents regulating operation; safety systems, instructions and trainings; application of supplementary measures and accident management guidelines as well as organisation of drills; and preparation for carrying out nuclear emergency response activities on and off the site.

The independence of the levels of defence in depth shall be ensured to the extent reasonably achievable.

There are different requirements for existing and new nuclear power plant units. Volume 3 of Annex of Govt. Decree 118/2011 (VII. 11.) is dealing with the design requirements for existing nuclear power plants, Volume 3a of the Annex deals with the design requirements for new nuclear power plants.

It is declared for existing nuclear power plants in Annex 3 that multiple physical barriers shall be applied during the design to prevent uncontrolled release of radioactive materials to the environment and these barriers shall be protected. The design of the barriers shall be conservative and shall be implemented according to the highest standards to ensure that the probability of failures and deviations from normal operating conditions shall be as low as reasonably achievable. The design shall ensure that DBC4 and DEC conditions will be prevented to a reasonably achievable level and no cliff edge effect can arise.

It shall be ensured as far as reasonably achievable that events that challenge the integrity of the barriers are prevented, and the endangering circumstances are tolerated. A concurring failure of more than one barrier has to be avoided and a barrier shall not fail due to the failure of another barrier or another system component. The detrimental consequences of human errors during operations or maintenance shall be avoided.

It is stated in Annex 3a for new nuclear facilities that in addition to the requirements under Article 7, supplementary requirements shall be met during the application of the five levels of defence in depth. It shall be ensured with independent protection levels that possible failures and abnormal operation can be detected, compensated and managed. During design, multiple physical barriers shall be applied for preventing an uncontrolled release of radioactive materials into the environment.

In order to apply the defence in depth principle, the following four physical barriers shall be distinguished:

a) fuel matrix;

b) fuel element cladding;

c) boundary of the primary circuit of the reactor;

d) containment system.

The protection of the barriers shall be ensured. The fulfilment of the safety functions and the acceptance criteria shall be ensured by design solutions even in the case of damage to any level of protection.

Five levels of defence in depth are defined, level 3 is split into 2 sub-levels (3a and 3b). Table 8 gives an overview of the levels of defence in depth for new nuclear power plants with the objectives, means to be applied, the radiological consequences and the relevant operating condition.

Prevention of events jeopardising the integrity of barriers and toleration of hazard factors, avoidance of the failure of more than one barrier at the same time and avoidance of the failure of one barrier as a result of the defect of another barrier or other system components shall be ensured to the extent reasonably practicable.

Design of the barriers shall be conservative, and they shall be implemented in accordance with high quality norms in order to reduce as low as reasonably achievable the possibility of failures and deviations from the normal operating condition. By design DBC4 and DEC conditions shall be excluded at a reasonably achievable level and cliff edge effects have to be avoided.

Table 8: The levels of defence-in-depth for new nuclear power plants in Hungary.

A	B	C	D	E
Level of defence in depth	Objective	Means to be applied	Radiological consequences	Relevant operating condition
1.	Deviations from the normal operating condition and prevention of failures	Conservative design, implementation and operation to a high standard; maintaining the main operating parameters between the prescribed limits	No off-site radiological effects exceeding the regulatory limits	Normal operation (DBC1)
2.	Management of deviations from the normal operating condition and failures	Control and safety protection systems; other surveillance methods		Anticipated operational occurrences (DBC2)
3. 3.a	Management of design basis accidents in order to limit radioactive releases and to prevent fuel melting	Safety systems, emergency operating procedures	No or only minimum off-site radiological effects	Design basis accident (DBC3-4)
3. 3.b		Added safety features for the elimination of complex accidents, emergency operating procedures, on-site emergency response measures		Complex accidents (Postulation of multiple failures) (DEC1)
4.	A practical exclusion of large or early releases, management of accidents involving a fuel melting in order to limit off-site releases	Supplementary safety features to limit fuel melting, accident management guidelines, on-site emergency response measures	An off-site radiological effect may warrant the introduction of protective measures limited in space and time for the population	Severe accident (DEC2)
5.	Mitigation of radiological consequences of a significant release of radioactive materials	On- and off-site emergency response measures; intervention levels	An off-site radiological effect warrants protective measures for the population	Very severe accident

ANNEX 5 - LITHUANIA

In 2017, the provisions of the Council of the European Union Directive 2014/87/Euratom were transposed into Lithuanian legislation in the form of amendments to the high-level documents, i.e. the Law on Nuclear Energy and the Law on Nuclear Safety. The meaning of the terms “avoidance” and “prevention” is the same. Article 3 of the Law on Nuclear Safety states that one of the basic principles for ensuring nuclear safety is:

> “*the principle of accident **avoidance (prevention)**, i.e. application of all rationally practicable measures preventing nuclear and radiological accidents and mitigating their consequences, if any.*”

Pursuant to Article 35 of the Law on Nuclear Safety, the defence in depth concept shall be implemented during design, commissioning and operation of nuclear installations to ensure the prevention and management of accidents, as well as mitigation of accidents consequences. It is stated that during the implementation of the defence in depth concept more than one physical barrier shall be incorporated, the technical and/or organizational measures to assure the integrity of these barriers shall be provided, and the technical and/or organizational measures for mitigation of harmful radiation, which are applied if these barriers are damaged or its effectiveness is decreased, shall be implemented.

Although the currently development of new NPP project is suspended, the legal system of the Lithuanian Authority (VATESI) is been continuously improved. In March 2018 new Safety Regulation “Nuclear Power Plant Design” was issued which establishes design requirements for NPP and its SSC and reflects the statements of Directive 2014/87/Euratom regarding new NPPs. The technical safety objective, which has to be achieved during whole life of NPP, is described as the assurance of implementation of all reasonably practicable measures to ensure NPP safety in order to avoid nuclear incidents, nuclear and radiological accidents, and, if they occur, to avoid or mitigate radiological consequences.

In the Safety Regulation “Nuclear Power Plant Design” the concept of defence in depth is understand as:

> “*the principle by following which a system of more than one independent level of protection is created, so that if safety measures at one level become ineffective, the safety objectives are achieved by means of other levels.*”

The defence in depth system dedicated to new NPPs is defined by 5 layers. Table 9 gives the description of each layer of defence in depth.

Table 9: Defence in depth concept based on Lithuanian Safety Regulation “Nuclear Power Plant Design”

Layer of defence in depth	Name	Objective
1	Assurance of normal NPP operation, prevention of deviations from normal NPP operation and prevention of failures	To ensure that the frequency of failures is as low as possible
2	Management of deviations from normal NPP operation and detection of failures	To detect deviations from normal NPP operation and to manage them in such a way as to prevent the such deviations to be developed into nuclear incidents or nuclear or radiological accidents, and to ensure that NPP is returned to normal NPP operational state as soon as possible
3	Management of accidents considered in NPP design that have not caused the damage of reactor core	To ensure the management of the accidents considered in NPP design, that have not caused the reactor core damage, and to prevent the development of such accidents into accidents causing the damage of the reactor core
4	Management of accidents considered in NPP design that have caused the damage of reactor core	Management of the accidents considered in NPP design that have caused the damage of reactor core, and the mitigation of the radiological consequences of such accidents
5	Mitigation of radiological consequences in the events of large release into the environment	To mitigate the consequences of accidents with significant radiological consequences

There is clear definition of the terms of “large release” and “large early release” in the Safety Regulation “Nuclear Power Plant Design”.

A **large release** is a release of radionuclides when more than 100 TBq of ^{137}Cs isotope are released into the environment or when the effect of other radionuclides (others than Caesium) emissions into the environment after three months after the release is higher than that caused by the emission of a 100 TBq ^{137}Cs isotope.

Large early release: the release of radionuclides into the environment, which can lead to harmful effects on people outside the NPP site, before the emergency preparedness measures, defined in regulating legislation and designed to protect them from this effect, can be carried out.

ANNEX 6 – ROMANIA

The regulation "Fundamental Nuclear Safety Requirements for Nuclear Installations" [34] includes requirements transposing the provisions of the Council Directive 2014/87/Euratom (July 2014), establishing a Community framework for the safety of nuclear installations. The regulation includes the provision of Article 8a of the new directive:

To fulfil the general nuclear safety objective,

> "*[…] 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."*

This objective will be 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.

Safety goals currently in use in Romania include:

- *Dose-frequency criteria* (maximum doses allowed for accidents of specified frequencies and/or maximum frequency allowed for accidents leading to doses in a certain range); these are established in the regulation on design and construction of NPPs [29];
- *CDF (Core Damage Frequency) and LERF (Large Early Release Frequency)* values based on INSAG-12; these are not formalised in regulations, but are used in review and assessments, in accordance with the principles outlined in paragraph 27 of INSAG-12. The results of Level 1 PSA (frequencies of occurrence for different accident sequences leading to core damage) and Level 2 PSA (type and amount of radioactive substances that could be released in the environment as the containment safety function is lost, together with the frequency of these releases) are necessary. [30]

It is recommended that the cumulative frequency of all events that could result in the exposure of a person outside the exclusion area, beyond the effective dose limit for the population, to be less than 10^{-3} per year. [29]

It is required by regulations that the concept of Defence-in-Depth should be applied to nuclear installations in all activities with impact on nuclear safety, with the following objectives: [34]

(a) minimizing the impact of extreme external hazards of natural origin as well as those caused unintentionally by humans;
(b) prevention of failures and abnormal operating conditions;
(c) detection and repair of defects and control of abnormal operating conditions, in order to prevent accidents;
(d) control of design basis accidents in such a way as to prevent the exceeding of the nuclear safety margins;
(e) (e) the control of severe conditions that could be caused by multiple failures such as the complete loss of all functions of a safety system or an extremely unlikely event, including the prevention of large-scale accidents and the mitigation of the consequences of severe accidents;
(f) on-site emergency management.

The safety philosophy of CANDU reactors, based upon the principle of Defence-in-Depth, employs redundancy (using at least two components or systems for a given function), diversity (using two physically or functionally different means for a given function), separation (using barriers and/or distance to separate components or systems for a given function), and protection (seismically and environmentally qualifying all safety systems, equipment, and structures). [28]

An important aspect of implementing Defence-in-Depth in the NPP design is the provision of a series of physical barriers to confine radioactive material at specified locations. In CANDU design these barriers are the fuel matrix, the fuel clad, the Heat Transport System, and the Containment. An additional administrative barrier is the exclusion area boundary.

The integrated and coordinated management system should ensure that the nuclear safety requirements have priority over any other requirements, considerations, and interests.

CNCAN has not yet issued any regulatory guides in relation to hazards, but it has endorsed the general guide issued by WENRA / RHWG to support the implementation of the reference levels in Issue T.

The licensee shall establish and implement a systematic process for reporting, collecting, sorting, analysing and documenting the operating experience and events from the nuclear installation for which it is responsible, as well as the relevant events and operating experience reported by other installations nuclear, nuclear organizations and organizations from other industrial sectors at national and international level. [33] The licensee shall ensure that CNCAN is promptly informed of any event with potential impact on nuclear safety.

Periodic safety reviews are performed in accordance with a national regulation which is based on the IAEA safety standards and WENRA Safety Reference Levels. Opportunities for improvement, including plant upgrades, are identified based on the review against the latest standards and implemented. In addition, safety reassessments, including new or revised safety analyses, are performed every time new information, significant in relation to the prevention and/or mitigation of nuclear power plant accidents, including severe accidents, becomes available, from operational experience or from research activities.

The Romanian regulation is based on the Safety Guide NS-G-2.10, having the 14 “safety factors” defined as “areas of review”, for each of these having specified most of the “generic review elements”. [28]

Periodic review of nuclear safety must be carried out systematically and periodically, at least once every 10 years. and should be done to ensure compliance with the current / updated design bases and to identify new improvements in nuclear safety, taking into account the cumulative effects of ageing of the nuclear installation, modifications, operational experience as well as the results of the updated nuclear safety analyses, the latest research results, scientific and technological developments and the latest internationally recognized standards and best practices.

Deterministic and probabilistic analyses and assessments, as well as engineering judgment, should be used in the review. [34]

The authorization holder shall review and revalidate nuclear safety analyses periodically at least once every 10 years from the start of operation of the nuclear installation to demonstrate that their assumptions remain valid and that the effects of ageing are effectively controlled that the nuclear safety margins are maintained throughout its lifetime. [32]

The PSR must determine: [31]

- to what extent the NPP meets current international nuclear safety standards and good international practice;
- (b) the completeness and validity of the nuclear safety documentation;
- if adequate measures are in place to continue to safely operate the plant until the next PSR or the end of the plant lifetime;

the necessary corrective actions to be implemented and the improvements that can be made to increase the nuclear safety of the plant.

ANNEX 7 – SLOVAKIA

Nuclear Regulatory Authority of the Slovak Republic (NRA SR) exercises state supervision over nuclear safety as whole. NRA SR work is driven by complex legislative framework arising from international treaties on nuclear safety of nuclear installations and nuclear materials management.

The topmost law is formed by Act no. 541/2004, so called Atomic Act. Top level obligations stated in Atomic Act are detailed in set of specific decrees. These decrees are periodically updated in order to reflect actual conditions especially European legislative framework as well as WENRA issues.

Decrees directly related to obligations in [25] are:

- Decree No.430/2011 Coll. on details on nuclear safety requirements for nuclear facilities covering obligations in Article 8a and according paragraphs 1(a) to 1(e) in Article 8b as well as paragraph 2(a) of [25].
- Decree No. 55/2006 Coll. on details concerning emergency planning in case of nuclear incident or accident covering obligation according paragraph 1(f) in Article 8b of [25].
- Decree No. 33/2012 Coll., on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear equipment and Decree No. 48/2006 Coll. on details of notification of operational events and events during shipment, as well as details of investigation of their reasons covering obligations according paragraph 2(b) to 2(c) in Article 8b of [25].
- Decree No. 34/2012 Coll., amending and supplementing Nuclear Regulatory Authority of the Slovak Republic Decree No 52/2006 Coll. on professional qualification and Decree No 52/2006 Coll. on professional qualification covering obligations according paragraph 2(d) in Article 8b of [25].
- Decree No. 33/2012 Coll., on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear equipment covering obligations sated in Article 8c of [25].

In addition, above mentioned decrees are added by NRA SR safety guides devoted to specific aspects as are safety analyses, single failure criterion, periodic safety review etc. Each act, decree and safety guide contains definitions part that introduces basic terms. The most significant from [16] point of view is Decree No.430/2011. Even if Decree No.430/2011 contains only high-level definition of basic terms meaning of particular terms follows from decree text. For example, defence in depth concept is briefly introduced as a system of multiple barriers and administrative measures. But Appendix 3 Part A Section C of Decree No.430/2011 defines 5 level of defence that are in compliance with categorization used in [12] and defines rules to implement defence in depth.

In order to unify communication NRA SR summarised nuclear safety terminology used in the up-to date (2016) legislation, safety guidelines and internal directives of the NRA SR or other relevant documents. The glossary, primarily oriented on nuclear safety, provides explanation of particular entries and serves as for the NRA SR internal use to create internal or external documents as for the communication between the regulatory authority and licensee during preparation, processing and assessment of the required documentation.

So, all obligatory legislative documents and guidance use consistent terminology that is summarized in NRA SR Glossary.

Even if some terms such ‘to prevent’, to mitigate’, ‘to avoid’, do not have separate entries or definitions their meaning follows implicitly from context of particular NRA SR documents and is consistent with meaning in [16].

The Slovakian regulatory framework as such does not use formulation or “the timely implementation of reasonably practicable safety improvements”. Equivalent of mentioned formulation is requirement stated in §3 article 2) of Act no. 541/2004: Upon new significant information being obtained about the risk and consequences of use of nuclear energy, the above mentioned level must be reassessed and measures shall be taken as necessary to meet the conditions of this Act. So, NRA SR regularly reassesses new information and if necessary particular NRA SR issue to reflect new conditions is released (e.g. specific issues following after Fukushima stress tests as well as after periodic safety reviews).

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

FINAL STUDY

VOLUME 3

ASSESSMENT OF APPROACHES AND METHODOLOGIES SET IN PLACE AT NATIONAL LEVELS FOR THE IMPLEMENTATION OF THE EU NUCLEAR SAFETY DIRECTIVE - REPORT BASED ON THE QUESTIONNAIRE ON NATIONAL APPROACHES

August 2020

EC Contract No. ENER/17/NUCL/SI2.769200

Table of contents

1 BACKGROUND

As stated in the inception report [3], the objective of Task 2 is to perform a technical evaluation of how the requirements of Articles 8a-8c of Directive 2014/87/Euratom are interpreted and applied in practice in the EU Member States based on the respective national frameworks. In order to perform this evaluation, a questionnaire was drafted by the consortium, and discussed during the first Workshop that was held in Luxemburg in July 2018. The first workshop [9] was an opportunity to foster a better understanding of the goal of this questionnaire: gathering answers in order to highlight similarities and differences between the Member States' approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom [2]. The questionnaire was amended taking into account the insights of the discussions held during the first workshop.

The questions aim at identifying practical approaches that contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. As such, it is worth noting that the assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study.

The consortium would like to highlight the very high response rate as well as the quality of the answers received. Out of the 16 Member States surveyed, the consortium received 11 answers, from Belgium, the Czech Republic, Finland, Germany, Hungary, the Netherlands, Slovakia, Slovenia, Spain, Sweden and the United Kingdom. No information was received from Bulgaria, France, Italy, Poland and Romania. However, the consortium used its experience as a collection of ETSON members to ensure the consistency of the analysis with the practices of the countries which did not provide an answer. Appendix 1 shows the questionnaire, whilst the answers received can be found in Appendices 2 through to 12.

This report presents the analysis of the answers performed by the consortium. Consistently with the tender specifications and the technical discussions of the first workshop, on the basis of the similarities and differences between national practices, this report:

- provides insights into how the articles appear to be practically implemented, then
- identifies potential technical areas on which further discussions and sharing the experience between Member States may be worthwhile.

The outcomes of this report have been presented and discussed during the second workshop of this project, held at the European Commission in November 2019. This led to several improvements, including an update of the suggestions, the final of which versions are presented in Chapter 4 of this report.

2 SCOPE OF THE REPORT

The current report presents the results of the technical evaluation of how the requirements of Articles 8a-8c are interpreted and applied in practice in the EU Member States based on the respective national frameworks. As such, the technical overall scope of the report covers the nuclear safety objective, its implementation as well as the initial assessment and the periodic safety reviews. According to the inception report [3], this report focuses on a set of nuclear installations, i.e.:

- nuclear power plants (NPPs), and
- research reactors with a thermal power above 1 MWt_h.

It is based on the evaluation of the answers provided by the Member States' Safety Authorities to a questionnaire prepared by the consortium and on the consortium's own knowledge and experience with regard to the actual situation in the countries represented.

Regarding the similarities between the national practices as well as topics where further international cooperation may be beneficial, the consortium provides some suggestions for further discussions.

Specific case of research reactors

It should be noted that very few countries provided elements on research reactors in their answers. However, those who did insisted on the similarities in the safety approaches they deploy compared to NPPs, while recommending the use of a graded approach with regards to the potential consequences in case of an accident. A clear and straightforward example could be the Netherlands, whose guidance document "Dutch Safety Requirements for Nuclear Reactors" is to "*be applied notably to new nuclear reactors during licensing in the Netherlands and used as a reference for safety evaluations of existing reactors. This guidance is also to be applied with a graded approach to new research reactors and again as a reference for existing research reactors*". The United Kingdom recognises that research reactors have differences from power reactors but need to be considered on a case by case basis. Moreover, regarding the most recent international standards, the consortium has enhanced in the Task 1 report [10] that the recently published IAEA Safety Standard SSR-3 Safety of Research Reactors (2016) [12] follows a similar philosophy as equivalent standards for NPPs.

The consortium is also aware of WENRA's on-going work that intends to develop safety reference levels (SRL) for research reactors on the basis of safety reference levels for NPPs.

Regarding the limited elements transmitted, the consortium is not able to provide non-disputable technical suggestions for research reactors. Nevertheless, the nature of approaches developed in the most recent international standards leads it to consider that the situation for research reactors is applicable to nuclear power plants. Furthermore, the consortium notices that performing combined studies both for NPPs and for nuclear installations other than NPPs tends to focus the results of the studies on NPPs.

3 ANALYSIS OF NATIONAL PRACTICAL APPROACHES

As indicated above, the report is based on the evaluation of the answers provided by the Member States' Safety Authorities to a questionnaire on national approaches established by the consortium. This questionnaire has 41 questions categorised into 9 topics. This structure and the questions themselves were devised to collect detailed information on the implementation of Articles 8a through 8c as efficiently as possible. Considering that the final goal of Task 2 is the performance of a technical evaluation related to the interpretation and application of Articles 8a-8c of Directive 2014/87/Euratom and not a straightforward collection of answers, an in-depth analysis and reconstruction of the answers was necessary in order to get back to Articles 8a-8c. Indeed, there is not a strict allocation of the topics and individual articles of Directive 2014/87/Euratom. The report structure follows Council Directive 2014/87/Euratom to allow a better understanding of the technical evaluation. For clarity purposes and for a better understanding the analyses, the consortium highlights the following links between the topics of the questionnaire and Articles 8a-8c of Council Directive 2014/87/Euratom:

- evaluation for Article 8a mainly relies on the answers related to the topic "safety objectives for new reactors and for existing ones";
- evaluation for Article 8b (1) mainly relies on the answers related to the topics "design and operational aspects of defence-in-depth", "probabilistic safety assessments", "design extension conditions without fuel melt", "design extension conditions with fuel melt", "design of the containment", "practical elimination";
- evaluation for Article 8b (2) mainly relies on the answers related to the topic "safety culture and operational experience";
- evaluation for Article 8c mainly relies on the answers related to the topic "periodic safety review".

These simplified relationships should be used as a complementary interpretative framework for a better understanding of the following chapters.

3.1 General principles of the objective and its application (Article 8a of the Directive)

The general principles that Article 8a (1) of Directive 2014/87/Euratom outlines are de facto addressed in national regulations as well as in technical practices. From the replies to the questionnaire that were received it can be seen that those countries who have answered the questionnaire systematically have taken into account the relevant safety objectives for new nuclear power plant designs established by WENRA in their national regulatory frameworks along with other international standards. The consistency of these publications with the objective of Article 8a, more particularly the link between the principles of WENRA's safety objectives and Article 8a, are developed hereafter.

Prior to this development, it is worthwhile to provide complementary explanations of what the above-mentioned "systematic reference" implies in practice. For instance, a typical example is Belgium which states that "*the Belgian Safety "philosophy" and regulatory framework for safety are fully in line with the WENRA work and publications that are based on the most recent IAEA Safety Guides. The Belgian regulation in safety of nuclear installations is regularly adapted to be fully in line with the WENRA Safety Reference Levels*". Similarly, Slovenia explains that the "*Slovenian legislation is in line with the IAEA safety standards, WENRA SRLs (2014) and WENRA Safety Objectives (which basically envelope the Articles 8a – 8b of Directive 2014/87/Euratom)*." The United Kingdom has a very similar philosophy, underlining that "*the IAEA Safety Standards and the WENRA Safety Reference Levels / Safety Objectives should be considered to be UK relevant good practice. Therefore, although international terminology is not always adopted, the requirements of the IAEA and WENRA are always encompassed within the UK regulatory framework.*" Germany states in its answer that while developing the German "Safety Requirements for Nuclear Power Plants", the most recent state of the art in science and technology has been considered. Amongst other documents, the WENRA Safety Objectives, WENRA Reference Levels, IAEA Safety Standards (in particular SSR 2/1 and SSR 2/2) have been consulted and implemented in the German regulations.

Concerning the consistency between Article 8a and the standards that European countries refer to, the principles of preventing accidents and, should an accident occur, mitigating its consequences and avoiding early and large radioactive releases are reflected in WENRA's safety objectives for new nuclear power plant designs [5] in a more detailed way. WENRA sets an objective of reducing the occurrence and the consequences of any kind of abnormal event and accident situation.

As a complement, qualitative objectives are also associated with these situations:

- no off-site radiological impact or only minor impact is tolerable for accidents without core melt;
- accidents with core melt[1] which would lead to early or large releases have to be practically eliminated;
- for accidents with core melt that have not been practically eliminated, only limited protective measures in area and time are needed.

As a result, it could be considered that early and large releases are avoided by the achievement of these WENRA qualitative objectives.

Achievement of WENRA safety objectives could be considered as a prerequisite to fulfil the general principle of the objective stated in Article 8a (1) of Directive 2014/87/Euratom.

This could be determined based on an overall analysis of these objectives, the achievement of which should obviously be included in the safety demonstration under Article 8c.

Article 8a (2) clarifies the framework of application of the objective from Article 8a (1). In particular, the objective is required for installations with a construction licence granted after August 14th, 2014. This means that the fulfilment of the safety objective in Article 8a (1) is mandatory for nuclear installations for which a construction licence is granted for the first time and thus is a design requirement for those installations. It should be noted that the word "new" is not mentioned: the Nuclear Safety Directive neither refers to nor specifies a technical concept of new nuclear installations.

In the case of existing nuclear installations with a licence granted before 14 August 2014, the nuclear safety objective serves as a reference to identify reasonably practicable safety improvements and foster the continuous improvement of nuclear safety. That same idea is reflected within the context of WENRA's safety objectives, which expect a reduction of frequencies of accidents and of their consequences for "new reactors" compared to currently operating nuclear power plants: "reduction" is not an absolute expectation, it reflects an improvement expectation.

[1] The complementary report "safety of new NPP design" [5] explains that "core melt" includes any severe degradation of fuel (not only in the core and not only due to melting).

The answers provided indicate that all the European countries are periodically updating their regulations to consider safety objectives in line with international standards. Based on the answers received, even if these updated safety objectives might not be strictly required for existing reactors, they are considered to seek improvements for existing reactors in line with the continuous improvement of safety.

For example, in Germany, "*the state of the art in science and technology in nuclear safety has to be applied when making regulatory decisions to grant a licence irrespectively if it is a new or existing nuclear installation*."

Spain provides a very similar answer: even though "*the new reactors are out of the scope of the future Spanish energy supply strategy, […] the RD 1400/2018 by which the Directive 2014/87/Euratom is transposed to the Spanish regulatory framework […] clearly states [that] whatever the energy planning be, […] the safety objective must be required to new facilities and considered as a referring for the existing ones. It is the aim of the Spanish regulatory body to require the existing NPPs to fulfil the safety objectives established by the Directive, as if they were new reactors*."

France has also developed a guidance for the "design of new pressurized water reactors" (guide n° 22, July, 2017), the objectives of which are "adopted" "consistently with the provisions mentioned in Article 8a introduced by the European Directive of 8th July 2014". Even if "*this guide applies primarily to the design of new-generation PWRs, its recommendations may also be used, for reference, to seek improvements to be made to reactors in operation, for example during their periodic safety reviews, in accordance with article L. 593-18 of the Environment Code and articles 8a and 8c introduced by the European Directive of 8th July 2014*".

However, some countries underline that the approaches for applying some safety requirements may differ considering whether they are reasonably practicable or not. Sweden illustrates it in its answer to the questionnaire: in Sweden, "[*The Swedish Radiation Safety Authority] review of the Code of Statutes includes to develop requirements that apply to both existing and new nuclear power plants. Most of the requirements will be the same for existing and new reactors. Some differences are considered. One important difference concerns radiological acceptance criteria, which most likely will be more stringent for new nuclear power reactors than for existing ones*."

Indeed, applying new objectives may not be reasonably practicable for installations that already exist to date. This is highlighted in Article 8a (2) b), which states that the objective should be *"used as a reference for the timely implementation of reasonably practicable safety improvements"*. WENRA also highlights [5] that its safety objectives *"should also be used as a* *reference to help identify reasonably practicable safety improvements for “deferred plants” and existing plants during Periodic Safety Reviews”*. WENRA has directly applied this principle when developing the 2014 set of reference levels.

The consortium highlights a valuable similarity that contributes to the fulfilment of Directive 2014/87/Euratom safety objective:

- **considering the most up-to-date safety objectives for any reactor, whether it be in operation (notably for periodic safety review), under construction, or in the design phase;**
- **for operating reactors, applying those objectives on the basis of an analysis taking into account both the specificities of the installation involved and the promotion of a continuous improvement of safety.**

The first suggestion of the consortium is that the similarities and differences of these case-by-case analyses for application of the most up-to-date safety objectives could be a relevant scope to be implemented in future work for ENSREG and WENRA. Exchanges on this scope could notably foster a mutual understanding of regulatory decision-making. Being aware that pursuing the reinforcement of the DEC concept application is an important issue, the scope of such an exchange amongst Member States should be prioritised to (selected) DECs as a first step. Within this context, on-going WENRA work related to the implementation of RLs of issue F at the plants may contribute to such an activity.

To complement this suggestion, it is worth pointing out that some expected objectives may be too demanding to be fulfilled for reactors in operation. As such, the consortium points out that a statement on whether these goals have been sufficiently fulfilled can only be made on the basis of a case-by-case analysis and by judgement of the national regulatory authority. On-going WENRA work related to the implementation of RLs at the plant could be an adequate tool to provide first insights.

The approaches to implement Article 8a are the main challenge to assess and are dealt with in the following chapters, regarding in particular the proper consideration of relevant issues (Article 8b) through an adequate process (Article 8c, in particular regarding periodic safety review).

3.2 Approaches to implement the objective (Article 8b of the Directive)

This chapter aims at providing insights on the implementation of expectations expressed in Article 8b of Council Directive 2014/87/Euratom, based on the answers gathered from the questionnaire. Article 8b outlines how the objective expressed in Article 8a should be implemented, mostly through applying defence-in-depth (Article 8b (1)) and promoting an "effective nuclear safety culture" (Article 8b (2)).

As mentioned above, the details of Article 8b (1) present a link with the topics of the questionnaire "design and operational aspects of the defence-in-depth", "probabilistic safety assessments", "design extension conditions without fuel melt", "design extension conditions with fuel melt", "design of the containment", "practical elimination", considering that this article mainly refers to adequate consideration of events or conditions, be they hazards, failures, abnormal conditions or accidents. The consortium highlights again that the link is not bijective and should only be used to help illustration of the analysis.

Regarding safety culture, all the answering countries agree that promoting and encouraging a strong safety culture is crucial to ensuring the safety of nuclear installations. However, the answers were very high-level, since this topic is hard to apprehend through documents only, especially through a questionnaire. Therefore, this report will not expand on safety culture. To further appreciate how safety culture is promoted and encouraged in practice, some dedicated work would be necessary, involving case studies and onsite analyses.

Reinforcement of levels of defence-in depth – General application of principles

Defence-in-depth remains a fundamental concept systematically underlined as a basis for safety in international literature (see task 1 report [10]) and considered as a "key concept" within the context of harmonized improved safety in Europe: Two positions of the WENRA report on new NPP designs [5] are dedicated to defence-in depth; WENRA reference levels [4] particularly highlight defence-in-depth when dealing with the prevention of events as well as the mitigation of their consequences (issues E and F). Answers to the questionnaire confirm a well-established concept both in the regulations and in actual practices, applied equally to new reactors and existing reactors.

Typically, the Czech Republic regulations clearly state that:

"(*1) Nuclear installation design shall, in the context of ensuring compliance with requirements for the application of defence-in-depth, ensure that a failure of a system, structure or component or loss of a safety function at one level does not reduce the effectiveness of the* *safety functions at the subsequent levels of defence-in-depth necessary to remedy or mitigate the consequence of an initiating event.*

(2) In order to create systems of subsequent defence-in-depth levels, the nuclear installation design may only use those systems, structures and components of the systems of the preceding defence-in-depth level that has been broken which:

a) have not been compromised in the course of the development of the nuclear installation's response to an off-site or on-site initiating event or scenario, and

b) are separable from the compromised or unusable parts of the systems of the preceding defence-in-depth level that has been broken."

In Spain as well, "*DiD is specially enhanced and reinforced by [a new Spanish regulation (RSN)] transposing EU Directive 2014/87. Specifically, the RSN distinguishes among [Design Basis Accidents], [Beyond Design Basis Accidents], external and internal events, […] in order to provide for a clear and overall view of the different plant safety analysis.*"

More specifically, "*at the level 4 of DiD, the requirements imposed by the RSN are fulfilled by both the safety evaluations that have been carried out since the initial and the subsequent licensing processes, and the implementation of the Fukushima National Action Plan, upcoming from the incorporation to the plants licensing basis the mandatory Technical Instructions to undertake the stress tests and to accomplish the requirements for both DEC and severe accidents.*"

Sweden expands on the reinforcement of defence-in-depth by explaining that "*measures for enhanced defence-in-depth and a reinforced independence between levels in the defence-in-depth have been done over a long period of time, including the actions taking after the TMI accident. Measures for further enhance the defence-in-depth by improving the degree of separation and diversification was also a result of the regulations that was decided in 2004.*"

Even though issuing licences for new nuclear power plants is not permitted in Germany, the German safety concept nevertheless demands that "*the state of the art in science and technology in nuclear safety […] irrespectively if it is a new or existing nuclear installation"* has to be considered. In the German defence-in-depth concept, levels 1 to 3 comprise operational states and design basis accidents. Level 4 is dedicated to those accident conditions that are more severe than design basis accidents and are not originally considered in the design.

Thus, consistent with other countries, it "*comprises measures and equipment that are allocated to different levels of defence. These comprise normal operation, anticipated operational occurrences (like transients), design basis accidents and very rare events (beyond design* *basis accidents). The prevention of events and the control of events is provided by measures and equipment for all these levels of defence as well as against internal and external hazards including very rare human induced external hazards*". Within this approach, level 4a is a sub-level dedicated to the objective of 'the control of events with postulated failure of the reactor scram system (ATWS)". "*Furthermore, additional measures and equipment to identify and limit the consequences of plant conditions that are not allocated to the above-mentioned levels of defence (up to level of defence 4a) due to their low probability of occurrence are provided as a precaution. Therefore, additional measures and equipment of the internal accident management are installed or planned, respectively, on levels of defence 4b and 4c of the defence-in-depth concept. These levels of defence are characterised by the following plant conditions:*

- *Level of defence 4b: events involving the multiple failure of safety equipment*
- *Level of defence 4c: accidents involving severe fuel assembly damages*."

Notwithstanding the specific way to formulate the levels of defence-in-depth, "*to achieve […] radiological objective several technical safety concepts have to be fulfilled. This are the defence-in-depth concept as the overarching concept which is supported by the concept of the multi-level confinement of the radioactive inventory (barrier concept), the achievement of the three fundamental safety functions and the protection concept against internal and external hazards as well as against very rare human-induced external hazards. In addition, for all plant states and levels of defence-in-depth radiological objectives have been formulated as well as technical acceptance criteria and targets*."

Some examples of safety improvements carried out to reinforce defence-in-depth are also provided. For example, Sweden points out that "*the introduction of an independent core-cooling function becomes a significant reinforcement of the nuclear power reactors defence-in-depth and strengthens the reactor's ability to prevent core damage for extreme events previously not included in the design basis. The independent core cooling function protects the plant against the events leading to the extended loss of normal auxiliary core cooling function*."

Belgium also answered with valuable insights into the reinforcement of levels 3 and 4 of defence-in-depth through 3 actions:

"*1° An action plan of the licensee for defining other DEC internal events and DEC hazards is currently in progress. This ensures independent means at levels 3a and 3b (diverse heat sink & diverse diesel generators). For some DEC events, loss of safety features is postulated, for example complete SBO and complete LUHS, leading to the need of (fixed or mobile) ultimate equipment.*

2° Filtered Containment Venting Systems are installed at all NPPs, used only for levels 3b and 4 events (ensuring an independence for the containment pressure control between level 3a and levels 3b/4)

3° Reactor Cavity Injection and Alternative Spray will be installed in Tihange (needed only for core melt events), but this is not yet "implemented" at this moment (independence between levels 3a-3b and level 4)".

Finally, it is worth noting that many safety improvements had already been performed prior to the formal reinforcement of defence-in-depth that came along with the Directive 2014/87/Euratom and the post-Fukushima updates. Those improvements, coherent with the continuous improvement of safety, allowed fulfilling the new requirements before they even came out. A good illustration of that can be found in the German answer to the questionnaire: "*many safety improvements have been already implemented in 1990ies years. Important safety improvements mainly based on research results and operating experience feedback have been implemented in cooperation with the licence holders.*" Several examples are then given, such as containment isolation, filtered venting, diversification of emergency power supply, etc.

Regarding the examples given above, the consortium would like to highlight that the potential strengthening of defence-in-depth and the associated added value should not be sought through its basic application nor through a too straightforward and never-ending undifferentiated expectation of reinforcement of each level. A more refined and relevant approach is suggested below.

First, the progressive development of the concept of "design extension conditions" (DEC) contributes to the deepening of the thoughts related to reinforcement of defence-in-depth. This concept has been developed to reflect that the consideration of more challenging events than those considered in the design basis is an integral part of the safety approach. The former concept of "beyond design basis" aimed at improving confidence in the ability of the reactor to withstand more challenging events. The DEC concept is a major step as its goal is a comprehensive consideration of these events associated with safety objectives whose achievement is expected to be demonstrated.

The aforementioned evolution towards a more harmonized safety approach for any phase of a reactor's life showcases an enhanced defence-in-depth regarding the concept of the DEC. WENRA reference levels constitute an important milestone: WENRA expanded it to any life phase of a reactor whilst IAEA standards (SSR-2/1 [11] or SSR-3 [12]) limit this concept to the early phases of reactor life (mainly design phase). WENRA regularly publishes a report related to status implementation of these reference levels (see http://www.wenra.org/media/filer_public/2019/04/17/status_of_the_implementation_of_the_2014_srls_-_1_january_2019.pdf, http://www.wenra.org/media/filer_public/2018/09/10/rhwg_status_of_the_implementation_of_the_2014_rl.pdf).

Moreover, WENRA also applies the concept to natural hazards and on-going WENRA work tends to apply it to any kind of hazard. This could be considered as a major contribution to Article 8b (1) a) which expects that the "*impact of extreme external natural and unintended man-made hazards is minimised*".

These evolutions should not be misinterpreted as limited to theoretical expectations: The management of DECs has been at the centre of much international practical technical work these past few years, particularly reinforced after the accident at Fukushima Daiichi NPP. The actual and technical improvements of practices or implementation of safety upgrades enhances that the concept of DECs allows for a comprehensive balance: making requirements that are stringent enough to improve the safety-related weak points of a nuclear installation while not deterring the implementation of safety upgrades.

Answers to the questionnaire illustrated that Member States have progressively sought more than just confidence regarding the resilience of reactors to challenging events but rather an actual demonstration that ambitious objectives are achieved.

The search of each country to reinforce adequately the relevant level of defence-in-depth that has already been discussed above might also be illustrated through the process related to safety classification.

Generally, the safety classification of systems is less stringent for systems used in the management of DECs with fuel melt than for systems used in the management of DECs without fuel melt. This is illustrated in some of the answers the consortium has received.

For example, Finland states that "*systems accomplishing safety functions shall be assigned to Safety Class 2 if they are designed to provide against postulated accidents to bring the facility to a controlled state and to maintain this state for as long as the prerequisites for transfer to a safe state can be ensured. Safety Class 3 shall include systems accomplishing safety functions that accomplish the diversity principle and are designed to ensure the bringing of the facility into a controlled state in case of the failure of systems primarily taking care of a corresponding safety function*." Additionally, "*systems needed to reach a safe state after a severe reactor accident are allowed to be non-safety-classified*."

Slovakia uses a very similar approach for classifying their systems. “*There are four safety classes defined: Safety classes BT1 (principally for primary circuit boundary), BT2 (principally for Safety Systems) and BT3 (principally for other SSCs important to safety), and BT4 (principally for SSCs assigned for prevention or limitation the consequences of malfunction of other SSCs, which are classified in safety class I to III (BT1 – BT3))*.” The Slovakian answer illustrates clearly the lower level of classification required for systems used in the management of DECs with fuel melt: “*in general, the SSCs for limiting the consequences of DECs without fuel melt are classified in safety class BT2 and safety class BT3 (of Classified Equipment). They must be seismically qualified. In general, SSCs for limiting of consequences for DECs with fuel melt are classified in safety class BT3 (of Classified Equipment).*”

Hungary could also be quoted with the intent of illustrating that decrease in the safety classification: “*in the operating NPPs mostly the same SSCs are used for limiting the consequences of DEC1 scenarios as for DB accidents, therefore their safety classification remains the same (Safety Class 2-3).The requirements for these SSCs changes for DEC1, namely that the single failure criterion no longer applies to them, assumptions and considerations can be made on a best estimate approach and the 95/95 rule (95% probability at a 95% confidence level of success) does not have to be followed. Systems that play a role in limiting and mitigating the consequences of DEC1 scenarios are required to endure the conditions that may arise during these scenarios. Under the current regulation safety systems dedicated to limiting and mitigating the consequences of DEC2 scenarios fall in SC3 category. There are no redundancy requirements for DEC2 systems, but they shall be designed to endure the conditions occurring under DEC2 scenarios.*”

The consortium highlights that the answers received and illustrated above clearly show that significant results have already been achieved with regards to the application of the DEC concept, leading to many safety improvements, some of which are common to several Member States. As a consequence, **the consortium supports pursuing the reinforcement of the DEC concept application**. This led the consortium to propose to first narrow the focus of its previous suggestion to DEC.

The consortium also highlights that the answers received and illustrated above should not be misinterpreted on the basis of a too straightforward analysis. Indeed, the consortium reckons that there are some slight differences in terminology between Member States regarding DECs or the definition/categorization of levels of defence-in-depth. These slight differences should be analyzed with a sufficiently broad view that is not limited to a single topic but includes the overall view of safety approaches targeting the achievement of safety objectives. Moreover, the history of implementation of safety approaches in different countries should also be considered, especially regarding existing reactors. Notwithstanding the way to associate some conditions with some levels of defence-in-depth or to denominate severe accidents, most countries associate DECs without fuel melt with level 3b of DiD, and all countries associate accidents with fuel melt with level 4 of defence-in-depth. Moreover, all European countries have consistently developed efforts to cope with these conditions and achieve similar objectives. Nevertheless, it may be difficult to illustrate in detail a common position on the safety provisions associated with the corresponding sequences of conditions. Within this context, considering that there are on-going WENRA activities related to the implementation of Safety Reference Levels of issue F (related to DEC) at the plants and considering the conclusions of the RHWG pilot study [15], **this on-going work could be an adequate tool for the first suggestion of the consortium**.

Reinforcement of levels of defence-in-depth – Probabilistic approaches

As illustrated above, the DEC concept leads to major upgrades or improvements of practices. Thus, the scope of DECs should be adequately set and prioritized towards the “weak points”.

PSAs are commonly used as a one of the tools to identify these weak points. The answers to the questionnaire even highlight that some countries use quantitative safety objectives.

Some countries have developed a direct quantitative use of probabilistic approaches.

For example, Finland proposes a threshold value for the Core Damage Frequency (CDF) written in their regulation and a corresponding threshold value for releases of radioactive materials: *“the design of a nuclear power plant unit shall be such that the mean value of the frequency of reactor core damage is less than* 10^{-5}*/year. The mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq shall be less than* $5 \cdot 10^{-7}$*/year.”*

The Netherlands proposes a more complex approach: The Decree on Nuclear Installations, Fissile materials and Ores defines *“radiological acceptance criteria for design basis and for DECs [… with] a maximum allowed effective dose outside the installation depending on the probability of the event:*

	Maximum allowed effective dose	
Probability of event (per year)	*Age ≥16*	*Age under 16*
≥ 10^{-1}	*0.1 mSv*	*0.04 mSv*
≥ 10^{-2} & < 10^{-1}	*1 mSv*	*0.4 mSv*
≥ 10^{-4} & < 10^{-2}	*10 mSv*	*4 mSv*
< 10^{-4}	*100 mSv*	*40 mSv*

The Dutch answer to the questionnaire also highlights that *"for DEC, requirements exist for beyond design-basis events (including DECs). First a maximum individual risk for a fictive person residing permanently and unprotected outside the facility of <10^{-6} per year, that person would die from a beyond design-basis event, including stochastic effects over a long period of time (50 years). Second a maximum group risk, that the probability that at least 10*n persons die of deterministic effects due to a beyond design-basis event of no more than $10^{-5}*n^2$ per year (for n≥1)."*

On the topic of a quantitative use of probabilistic approaches, the consortium underlines that some unknowns, hypotheses and uncertainties exist in PSAs, especially when used for DECs that encompass multiple phenomena and combine multiple failures with corresponding uncertainties related to reliability data. As such, it is suggested to use PSAs with caution with regards to the associated stakes. They could be seen as a useful decision-making tool to complement a deterministic approach that would offset the probabilistic approach weaknesses.

Several answers illustrate that very well. For example, the Czech Republic explains that the *"list of scenarios for DECs must be determined on the basis of engineering judgment with the use of deterministic and probabilistic methods. The PSA must be used, but it does not mean that it is possible to not introduce any mitigation measures for coping with a severe accident simply because of the very low probability of its occurrence."*

Slovakia proposed a comprehensive and illustrative list of the uses of PSA:

"*a) To support safety management and decision-making;*

b) To identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant;

c) *To assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects";*

d) *To assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational events;*

e) *To develop and to validate of the safety significant training programs of the licence holder, including training on a full scope representative simulator;*

f) *To verify that the main contributors to risks are included in the facility's maintenance, inspection and test programs.*"

The Hungarian answer to the questionnaire could also be quoted to further illustrate the uses of PSAs: *"there is a general safety requirement, that the PSA shall prove that the design is "balanced", i.e. there are no outstanding contributors among the initial events or the minimal cut-sets predominantly influencing the PSA results while importance measures shall also be taken into consideration to justify that the design is balanced."*

The safety upgrades implemented in Member States tend to corroborate the importance of an elaborated use of PSAs in the continuous improvement of safety as illustrated above. It permits a focus on reinforcing situations that were reassessed as more probable or more severe than previously thought. PSAs have proven to be a very useful tool in that regard, helping with the identification of some weaknesses in the safety case.

Germany answers that the PSA is applied to supplement deterministic safety demonstrations and to verify the balance of the safety-related design. Furthermore, probabilistic safety analyses shall also be carried out to assess the safety significance of modifications of measures, equipment or the operating mode of the plant as well as of findings that have become known from safety-relevant events or phenomena that have occurred and which may be applied to the nuclear power plants in Germany that are referred to in the scope of application of the "Safety Requirements for Nuclear Power Plants" for which a significant influence on the results of the PSA can be expected.

The French example of safety improvements implemented for molten core retention to avoid containment basemat melt-through could also be given. More details on these improvements can be found in the Task 3 report.

However, as stated above, PSAs are subject to some unknowns and uncertainties; they can underestimate the relative importance of some accident sequences. As such, the consortium advises ensuring the relevance of each use of PSA.

Reinforcement of levels of defence-in-depth – Conclusion

All the considerations detailed above, backed up by several examples taken from the answers gathered, lead the consortium to formulate the following suggestion.

The consortium highlights the benefit of using a systematic approach for ensuring the continuous improvement of safety, based on:

- **identifying the sequences entailing the most critical safety issues, and**
- **adequately reinforcing the requirements on the corresponding safety provisions - either to prevent these sequences or to limit their consequences.**

The consortium suggests that the European nuclear industry, TSOs, ENSREG and WENRA share their practices on this topic to help with the identification and design of new improvements.

It is worth noting that these critical sequences are not necessarily those entailing the most severe consequences.

In that regard, PSAs remain an effective informative tool to help defining those critical sequences, the safety provisions to prevent and/or mitigate them, and their associated requirements. For existing reactors, this approach may be deployed during the Periodic Safety Reviews (PSRs), as detailed in the next chapter.

Furthermore, the answers to the questionnaire suggest that the safety upgrades implemented in different Member States are quite similar and are mostly focused on handling the critical sequences described above. This was illustrated earlier in this chapter through examples of safety upgrades from Sweden and Belgium. However, it is difficult to go further than that solely based on the answers to the questionnaire. This was the motivation for the third task of this project, which aims at realising detailed country-specific studies in order to further investigate these similarities.

Independence of levels of defence-in-depth

In addition to reinforcing the individual levels of defence-in-depth, an adequate independence between the levels should be fostered.

As a basis of adequate independence, a clear and relevant definition of each level is crucial. Regarding that, the WENRA approach concerning DECs constitutes a fundamental step as it affects DECs without core melt up to level 3, considering that the objectives and phenomena involved are similar to DBAs. As a consequence, level 4 of defence-in-depth is dedicated to

DECs with core melt. This reflects that DECs with core melt position have a different place compared with other events in the defence-in-depth concept: While some events (failure of a SSC or multiple failures of SSCs for example) are postulated on the previous level of defence-in-depth, the consequence (core melt) is postulated on level 4 irrespective of the events that have initiated this consequence. Thus, it is fundamental to ensure independence between levels 3 and 4 of defence-in-depth. Current practices tend to enhance this independence even though it is more practicable to implement in the case of a reactor in the design phase than for existing reactors.

Nevertheless, a straightforward analysis showed that dedicated, thus independent, provisions for level 4 of defence-in-depth are not easily implemented. A very practical example is the on-going French programmes for modifying the reactor vessel pit in existing reactors to allow corium stabilisation and avoid containment basemat melt-through in case of reactor vessel failure after core melt. This includes notably the addition of an appropriate concrete layer on the corium spreading area to help corium stabilisation and a passive system to activate corium submersion by water after spreading. This general description does not reflect in detail the modifications that will be implemented in each reactor but is sufficient to show that such modifications may have been considered as not being practicable before adequate research and development has been carried out.

Other examples come from a previous illustration. Belgium has provided answers that highlight different types of independencies, thus different conclusions of analyses are drawn when describing the 3 following actions:

- "*An action plan of the licensee for defining other DEC internal events and DEC hazards is currently in progress. This ensures independent means at levels 3a and 3b (diverse heat sink & diverse diesel generators)"*;
- *Filtered Containment Venting Systems are installed at all NPPs, used only for levels 3b and 4 events (ensuring an independence for the containment pressure control between level 3a and levels 3b/4)"*;
- *Reactor Cavity Injection and Alternative Spray will be installed in Tihange (needed only for core melt events) … (independence between levels 3a-3b and level 4)"*.

Practical elimination

Finally, a couple of questions from the questionnaire focused on “practical elimination”, a topic that is extensively discussed among several international institutions, for example WENRA RHWG [16]. Council Directive 2014/87/Euratom does not explicitly talk about “practical elimination”. However, this concept introduced in the IAEA INSAG-12 [13] in 1999 is now widely acknowledged and used in Member States.

The answers to the questionnaire reflect the complexity of practical elimination - when to use it and how to demonstrate that a situation has been practically eliminated. For example, despite the fact that the existing standards apply practical elimination to some event sequences (or accidents or events or conditions… the wording may differ) that could lead to large or early releases, some countries claim that their objective is to apply this notion directly to early and large releases. However, the common view of the achievement of practical elimination is that something could be considered to have been “practically eliminated” if it is physically impossible for it to occur or if it could be considered to be extremely unlikely to arise with a high level of confidence. Regarding the current state of the art in science and technology concerning nuclear power plants, the physical impossibility of early large releases is not reasonably achievable. Only the physical impossibility of some event sequences that would lead to early large releases makes sense.

As such, the adequate identification of event sequences to be practically eliminated is critical. The answers to the survey reveal a large diversity of theoretical views. Some regulations may even expect the practical elimination of all “severe accident states”, which may be interpreted as inconsistent with defence-in-depth approach.

It may be surprising that such an old concept (one of its first appearances is in the common French-German document GPR/93-27 of 1993, June 14th) is still in its development phase. The WENRA publication [16] is quite recent and the IAEA continues to try and draft further guidance and clarification (see on-going work on DS 508 for example).

Nevertheless, there are similarities in different European approaches.

Typically, as mentioned above, all countries consider that the occurrence of an event or event sequence or a state can be considered as practically eliminated:

- if it is physically impossible to occur, or
- if it can be considered with a high degree of confidence to be extremely unlikely to arise.

There seems to be a common understanding in Member States that practical elimination, as part of the safety case, is used as a complement or reinforcement to DiD and contributes to the achievement of Article 8a of the Directive. In particular:

- it appears to be systematically applied at least to severe accident sequences that cannot be mitigated;
- it should be considered at an early stage of the design phase. Practical elimination achievement requires an accurate case-by-case analysis fully dependent on the reactor design for the identification of accident sequences to be practically eliminated and a definition of design provisions with adequate requirements.

Consequently, the discrepancies in the theoretical views of practical elimination have not hindered the implementation of harmonized safety improvements. It should be stressed that all the fleet in Europe has identified a set of scenarios to define – when relevant – provisions to guarantee the prevention of notably:

- high pressure core melt accidents and Direct Containment Heating (DCH);
- steam explosions leading to failure of the containment;
- hydrogen combustion processes endangering containment integrity;
- rapid reactivity insertion;
- containment bypass (concomitant with a severe accident);
- fuel damage in the spent fuel pool.

WENRA has recently published a RHWG report (reference [16]) showing that there may be different theoretical views related to the notion of practical elimination without challenging the application of fundamental harmonized principles and the achievement of harmonized objectives.

The consortium suggests regulators that TSOs and the European nuclear industry should consider the recent WENRA paper on practical elimination as a reference document. Should they become involved in international cooperation on practical elimination, it might be better to focus the discussion on similarities in the technical application of the concept in order to identify further insights related to its justification. Although discrepancies in the theoretical notion of practical elimination exist, this is a topic that is of less importance than the aforementioned technical one.

3.3 Periodic Safety Reviews (Article 8c of the Directive)

Article 8c-(a) stresses that the safety case should be based on siting considerations, meaning an installation-specific assessment should be performed before granting a construction licence to construct. The consortium believes this consideration is widely acknowledged by the Member States and did not necessitate any specific question in the questionnaire.

Article 8c-(b) describes expectations regarding Periodic Safety Reviews (PSRs) in connection with the safety of nuclear installations. Part of the questionnaire focused on this topic since PSRs are an essential element for ensuring satisfactory safety of these installations for their entire lifetime by both compliance with design expectations and implementation of a continuous improvement approach.

The corresponding approach is a well-established approach within Member States: both aforementioned aspects are integrated in issue P of WENRA SRLs which also expects – to a lesser extent for former version of SRLs - a quite comprehensive scope of PSR and due consideration of “ageing issues, operational experience, most recent research results and developments in international standards”, which is consistent with Article 8c-(b). WENRA has also underlined the importance of PSRs through its position paper related to “Periodic Safety Reviews (PSRs) taking into account the lessons learnt from the TEPCO Fukushima Daiichi NPP accident” (March 2013). As a consequence, answers to questionnaire highlight how sensitive the Member States are to the importance of PSRs. The corresponding processes appear to be quite comprehensive and rigorous.

It should be noted that the range of topics covered by PSRs may vary slightly from one country to another. For instance, Sweden mentions in its answers to the questionnaire that PSRs cover physical protection/security aspects as well as nuclear non-proliferation, exports control and transport safety; Slovenian PSRs include supplementary considerations on radioactive waste and spent nuclear fuel, safeguards and radiation protection. These topics are in fact not dealt with in the WENRA SRLs, but discussed within other international instances. Moreover, the consideration of these topics may also be part of other countries PSRs even though the answers to the questionnaire did not refer to them. The Member States then decide at an individual level on how to integrate the results of all these different aspects in their national regulation, be it through the PSR or by other means.

The answers provided by the Member States’ Safety Authorities, along with the detailed analysis of safety upgrades performed in Task 3, tend to prove that the SRLs requirements are adequately considered in PSRs.

According to the answers to the questionnaire received, several countries consider that the most recent developments in international standards should be taken into account while performing a PSR. Even if the wording may differ, most of the national regulations expect that PSRs consider insights arising from the current state of the art in science and technology and good practices widely used at the time of its performing.

For instance, in the UK, the principle of continuous improvement is implemented to achieve sustained high standards of nuclear safety, and its application ensures that no matter how high the standards of nuclear design and subsequent operations are, improvements should always be sought.

Similarly, in Finland, the PSR process includes the licensee’s assessment of how the modern safety standards can be fulfilled as far as reasonably practicable. Furthermore, any non-compliance of the NPP with the safety standards is reported, and the safety significance of these non-compliance findings and prioritisation of PSR actions is determined afterwards. Following this process, decisions concerning some further safety improvements are made.

In Germany, the licensee is obliged to respect the state of the art in science and technology. This includes recent developments in international standards – which represent the current state of the art in science and technology - in his analyses and, consequently, the identification and determination of safety improvements. This is comprehensively assessed during the decennial PSRs. In addition, for nuclear installations in the post-operational phase, the federal authority and the regulatory authorities have decided that the licence holder has to perform a safety analysis for this phase. Corresponding details are specified in a “Checklist for the performance of an assessment of the safety status of the installation for the post-operational phase”.

Each of the review areas, to the extent applicable, are systematically analysed and assessed to underline how systems, structures, components and activities meet relevant new safety standards and practices, and actions are taken when necessary, to the extent possible and reasonable (e.g. Sweden, Czech Republic). In many countries, safety requirements are regularly updated, considering operating experience and safety research and advances in science and technology (e.g. Finland). In Slovenia, international standards are included as references into the PSR program, being used as comparison for all of the safety factors.

More generally, all countries seem to be using PSRs as an opportunity to reassess the safety objectives of the nuclear installation in order to bring them closer to the most recent safety objectives (typically WENRA safety objectives for new reactors). Such an approach allows the identification of reasonably practicable safety upgrades that the operators should then implement in a timely manner, thus contributing to the continuous improvement of safety.

However, it may be necessary to focus on the aforementioned most critical safety concerns. This can be achieved by using “operational experience, most recent research results and developments in international standards” to perform a first selection of critical sequences. This could be achieved by using notably – but not only - PSAs as a tool to help identifying the “weak points” of a nuclear installation and prioritizing them by assessing the safety benefit of the proposed changes.

The topics and situations which are not the main focus of the PSR should also be considered, but with a different approach: from a compliance perspective, i.e. ensuring that the installation as built and as operated complies with its safety requirements.

The previous observations demonstrate that PSRs play a critical role in the continuous improvement of safety. Within this context and consistently with the answers provided, the most relevant topics requiring particular attention and situations that need to be reinforced are identified, as are potential cases of non-compliance of the reactors as built and as operated with their design expectations.

In the PSR documentation devoted to ageing, the licensee provides a summary of the ageing management programme concerning – depending on the practices of different countries - the operating licence period or design lifetime or any other adequate parameter applied for and/or remaining for the facility. Based on the PSRs results, a decision regarding the continuation of the operation of the plant until the next periodic safety review is made.

The methodology and scope of PSR and LTO are identical but some topics (e.g. ageing) would benefit of a greater attention to LTO and additional time for the review might be necessary. A comparison between different approaches to LTO and the PSR framework is presented in IAEA document No. NP-T-3.18 [8].

Regarding safety enhancements, one important element in the evaluation of what is “reasonable” will be the remaining time for which the considered plant will be operated before final shutdown.

A generic lesson learned in Finland is that the closer NPPs get to the end of their design lifetime, the more difficult is it for the licensees to make decisions to modernise or modify the NPPs. This situation sustains the idea of improving safety as a continuous process from the start of plant operation and not related only for example to the plant’s long-term operation.

In Romania, the planned refurbishment of Cernavoda Unit 1 will provide an adequate frame for the implementation of additional practical modifications to enhance the safety of the facility to a higher level.

4 SUMMARY

The consortium has analysed the answers to the questionnaire provided by 11 Member States and used its collective experience as ETSON members when no answer was provided. On this basis, the consortium has performed a technical evaluation of how the requirements of Articles 8a-8c of Directive 2014/87/Euratom are interpreted and applied in practice in the EU Member States.

The consortium has devised its questionnaire structure and content so as to collect detailed information on the implementation of relevant Articles 8a through 8c as efficiently as possible. There is not a strict bijection between the topics reflected by the structure of the questionnaire and individual articles of Directive 2014/87/Euratom. Thus the technical evaluation could not provide a straightforward collection of answers analysed independently of each other, and an in-depth analysis and reconstruction of the answers was necessary in order to get back to Articles 8a-8c. Nevertheless, it could be highlighted that Article 8a is mainly in relation with the topic “safety objectives for new reactors and for existing ones”; Article 8b1 mainly with topics “design and operational aspects of the defence-in-depth”, “probabilistic safety assessments, “design extension conditions without fuel melt”, “design extension conditions with fuel melt”, “design of the containment” or ”practical elimination”; Article 8b2 mainly with the topic “safety culture and operational experience”; Article 8c mainly with to the topic “periodic safety review”.

Regarding Article 8a of Directive 2014/87/Euratom, the consortium has described how achievement of WENRA safety objectives could be considered as an adequate way to fulfil the general principle of the objective stated in article 8a (1) of the Directive.

The consortium highlights that there is a valuable similarity that contributes to the fulfilment of the safety objective of Directive 2014/87/Euratom

- **considering the most up-to-date safety objectives for any reactor, whether it be in operation (notably for periodic safety review), under construction or in the design phase;**

- **for operating reactors, applying those objectives on the basis of an analysis taking into account both the specificities of the installation involved and the promotion of a continuous improvement of safety.**

As a consequence, the first suggestion of the consortium is that the similarities and differences of these case-by-case analyses for application of the most up-to-date safety objectives could be a relevant scope to be implemented in future work for ENSREG and WENRA. Exchanges on this scope could notably foster a mutual understanding of regulatory decision-making. Being aware that pursuing the reinforcement of the DEC concept application is an important issue, the scope of such an exchange amongst Member States should be prioritised to (selected) DECs as a first step. Within this context, on-going WENRA work related to the implementation of RLs of issue F at the plants may contribute to such an activity.

Indeed, when performing its evaluation related to Article 8b1 of Directive 2014/87/Euratom, the consortium concluded that there is a need to support pursuing the reinforcement of the DEC concept.

Regarding this evaluation, it should be first remembered that defence-in-depth remains a fundamental concept which in international literature is systematically underlined as a basis for safety and considered as a “key concept” within the context of harmonised improved safety in Europe. Answers to the questionnaire confirm that this is a well-established concept both in the regulations and in actual practices, applied equally to new reactors and existing reactors. Moreover, many safety improvements had already been performed prior to the formal reinforcement of DiD that came along with Council Directive 2014/87/Euratom and the post Fukushima updates. Those improvements, coherent with the continuous improvement of safety, allowed fulfilling the new requirements before they even came out.

Henceforth, the consortium would like to highlight that the potential strengthening of defence-in-depth and the associated added value should not be sought through its basic application nor through a too straightforward and never-ending undifferentiated expectation of reinforcement of each level. Within this context, the concept of “design extension conditions” (DECs) contributes to the deepening of the thoughts related to the reinforcement of defence-in-depth. This concept has been developed to reflect that the consideration of more challenging events than those considered in the design basis is an integral part of the safety approach. The DEC concept is a major step as its goal is a comprehensive consideration of events associated with safety objectives whose achievement is expected to be demonstrated: Answers to the questionnaire have shown that Member States have progressively been seeking more than just confidence regarding the resilience of reactors to challenging events but rather actual proof that ambitious objectives are achieved.

The consortium highlights that the answers received and illustrated above clearly show that significant results have already been achieved with regards to the application of the DEC concept, leading to many safety improvements, some of which are common to several Member States. As a consequence and as mentioned above, **the consortium supports pursuing the reinforcement of the DEC concept application and thus proposes to focus on this topic for its first suggestion**.

Whilst the first suggestion of the consortium is related to regulatory decision-making and proposes to use the feedback of past improvements, all the considerations detailed above, backed up by several examples taken from the answers gathered, have lead the consortium to formulate the **second suggestion,** which is related Article 8b1 of Directive 2014/87/Euratom and which is applicable to the whole nuclear community when identifying and designing new improvements.

The consortium highlights the benefit of using a systematic approach for ensuring the continuous improvement of safety, based on:

- **identifying the sequences entailing the most critical safety issues, and**
- **adequately reinforcing the requirements on the corresponding safety provisions - either to prevent these sequences or to limit their consequences.**

The consortium suggests that the European nuclear industry, TSOs, ENSREG and WENRA share their practices on this topic to help with the identification and design of new improvements.

The third suggestion of the consortium is specific to a particular approach: practical elimination.

The consortium suggests regulators that TSOs and the European nuclear industry should consider the recent WENRA paper on practical elimination as a reference document. Should they become involved in international cooperation on practical elimination, it might be better to focus the discussion on similarities in the technical application of the concept in order to identify further insights related to its justification. Although discrepancies in the theoretical notion of practical elimination exist, this is a topic that is of less importance than the aforementioned technical one.

Regarding Article 8b2 of Directive 2014/87/Euratom and safety culture, all the countries that have given answers agree that promoting and encouraging a strong safety culture is crucial to ensuring the safety of nuclear installations. However, the answers were very high level since this topic is hard to apprehend through documents only, especially through a questionnaire. Therefore, this report has not expanded on safety culture. To further appreciate how safety culture is promoted and encouraged in practice, some dedicated work would be necessary, involving case studies and onsite analyses.

Finally, regarding Article 8c of Directive 2014/87/Euratom, common expectations regarding PSRs are identified and are an essential part for ensuring the satisfactory safety of these installations for their entire plant lifetime through:

- compliance with design expectations;
- implementation of an approach of continuous improvement.

Even if the wording may differ, the answers reflect the consideration of ageing issues, operational experience, most recent research results and developments in international standards.

The range of topics covered by PSRs may vary slightly from one country to another (physical protection/security, export control, transport safety, radioactive waste, spent nuclear fuel, safeguards, radiation protection…). Nevertheless, the answers provided by the Member States' Safety Authorities tend to show a harmonisation; especially the WENRA SRLs requirements are adequately considered in PSRs.

5 References

[1] Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations.

[2] Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations.

[3] Inception Report “Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom”, ENER/D3/2017/-209/-2, Rev 2, 21.03.2018.

[4] WENRA Safety Reference Levels for Existing Reactors, September 2014.

[5] WENRA Safety of new NPP designs, March 2013.

[6] IAEA Safety Guide No. NS-G-2.10 Periodic Safety Review of Nuclear Power Plants, STI/PUB/1157, ISBN 92-0-108503-6, Vienna, 2003.

[7] WENRA Pilot Study on Long term operation (LTO) of nuclear power plants, March 2011.

[8] IAEA Nuclear Energy Series No. NP-T-3.18, PLANT LIFE MANAGEMENT MODELS FOR LONG TERM OPERATION OF NUCLEAR POWER PLANTS, Vienna, 2015.

[9] PR-1804-ENER/D3/2017/-209/-2-Proceedings of 1st Workshop, 2018.

[10] FR-2006-ENER/D3/2017/-209/-2 Final Study Vol.2 Study on International and European Guidance Documents, June 2020.

[11] IAEA Safety Standards No. SSR-2/1 Rev. 1 Safety of Nuclear Power Plants: Design, Vienna, 2016.

[12] IAEA Safety Standards No. SSR-3 Safety of Research Reactors, Vienna, 2016.

[13] IAEA INSAG-12 Basic Safety Principles for Nuclear Power Plants 75-INSAG-3 Rev. 1, Vienna, 1999.

[14] PR-1804-ENER/D3/2017/-209/-2-Discussion on updated findings and draft recommendations of the project addressing the main good practices/areas of potential improvements/gaps or harmonization including potential increase of consistency in implementation practices, 2019.

[15] RHWG report - WENRA Safety Reference Levels 2014 - Implementation at the nuclear power plants, reasonably practicable safety improvements and benchmarking - Pilot Study, November 2019, 6th.

[16] Practical Elimination Applied to New NPP designs - Key Elements and Expectations, November 2019, 6th.

6 List of Appendices

Appendix 1: Questionnaire

Appendix 2: Answers – Belgium

Appendix 3: Answers – Czech Republic

Appendix 4: Answers – Finland

Appendix 5: Answers – Germany

Appendix 6: Answers – Hungary

Appendix 7: Answers – Netherlands

Appendix 8: Answers – Slovakia

Appendix 9: Answers – Slovenia

Appendix 10: Answers – Spain

Appendix 11: Answers – Sweden

Appendix 12: Answers – United Kingdom

Appendix 1: Questionnaire

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

Revision R2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name:

Organization:

Position:

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety “philosophy”)

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety “philosophy”) adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.)you require from this level of PSA:

- PSA level 1:
- PSA level 2:
- PSA level 3:

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed?
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis” as mentioned in Article 8b? If so, please explain.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?
- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?
- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Question 9.2: Do you have requirements related to practical elimination?

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment

- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

Appendix 2: Answers to the questionnaire - Belgium

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome
Nuclear Power reactors as well as research reactors are categorized as "Class I" facilities in the Belgian legal and regulatory Framework. This category is broader than that of "Nuclear installations" of the Directive and comprises also waste management and storage facilities, waste disposal facilities and any installation where fissile nuclides (U235, Pu239) are present in a quantity higher than the half of the minimal critical mass. The same regulatory framework is applicable for all of these facilities. Originally, the design and safety analysis of the last four reactor units have been done following the US NRC rules and all the associated documentation (regulatory guides, standard review plans, ASME Code, IEEE standards, ANSI, ANS, etc.) in order to ensure a consistent approach. 10 CFR 20 on radioprotection was not followed, as the corresponding topics were covered by the Euratom Directive on the Basic Safety Standards, which is mandatory for all member States of the European Union. For safety-related pressure vessels, a specific derogation to the Belgian pressure vessel regulations ("Règlement général pour la protection du travail") was elaborated, in order to allow the use of the US rule based ASME Code sections III and XI. The initial license of the Belgian NPP already requested the conduct of ten-yearly periodic safety reviews, to take into account development of regulations and standards and technical and scientific developments for continuous improvement of the safety. In 2011, a Royal Decree on the "safety requirements" for these nuclear installations has been issued, based on the WENRA 2008 reference levels for existing reactors. The content of this decree is described in the Belgian report for the Convention on Nuclear Safety, section II.C.4, pg 27, available on the IAEA web site: http://www-ns.iaea.org/downloads/ni/safety_convention/7th-review-meeting/belgium-nr-7th-rm.pdf. This Royal Decree is currently being updated to include the WENRA 2014 safety reference levels. Belgium is an active member of WENRA and of its working groups (RHWG and WGWD). In this sense, the Belgian Safety "philosophy" and regulatory framework for safety are fully in line with the WENRA work and publications, that are based on the most recent IAEA Safety Guides. The Belgian regulation in safety of nuclear installations is regularly adapted to be fully in line with the WENRA Safety Reference Levels The basic Belgian legal texts relevant for the safety of nuclear installations are: • The Law of 15 April 1994 on the protection of the population and the environment against the hazards of ionizing radiation and on the Federal Agency for Nuclear Control (amended for the last time in 2018), • The Royal Decree of 20 July 2001 laying down the "General Regulations regarding the protection of the public, the workers and the environment against the hazards of ionising radiation" (GRR-2001, amended for the last time in 2018). • The Royal Decree of 30 November 2011 on the Safety Requirements for Nuclear Installations (SRNI-2011 amended for the last time in 2018 to implement the EC Directive 2014/87/Euratom of 8 July 2014 in Belgian regulations). More information can be found in the Belgian national reports (2016 edition) for the Convention on Nuclear Safety (CNS), Article 7 - Legislative and

Regulatory Framework -, published on the IAEA web site) : http://www-ns.iaea.org/conventions/nuclear-safety.asp

In addition, the Belgian regulatory body has developed guidelines on safety demonstration and specific external hazards for new nuclear installations (except waste disposal facilities). The guideline on safety demonstration outlines the expectations of the nuclear regulator with respect to defence in depth, quantified safety objectives and external hazards in general. The specific guidelines on external hazards provide expectations on how one or more hazard levels can be derived with the purpose to include these levels in the safety demonstration; these guidelines address respectively seismic hazards, unintentional aircraft crashes and external flooding. The guidelines have been developed using several sources including IAEA standards, WENRA positions and the recently issued European Council Directive 2014/87/Euratom of 8 July 2014 which amends Council Directive 2009/71/Euratom. The following guidelines were published:

- Safety Demonstration (last revision in April 2017)
- Seismic hazard (last revision in February 2015)
- (Accidental) Airplane crash (last revision in February 2015)
- External flooding (last revision in February 2015)
- Determination of Radiological consequences following a release (last revision in June 2017)

These guidelines are available on the FANC web site : https://afcn.fgov.be/fr/dossiers-dinformation/autres-etablissements-nucleaires/directives-pour-unenouvelle-installation

In particular, the guideline “Safety Demonstration” :

- Is based on WENRA publications (e.g. WENRA report “Safety of New NPP Designs” (2013)) and EC Council Directive 2014/87/EURATOM of July 8, 2014
- Applies to all new nuclear class I installations (except waste disposal facilities)
- Describes Defence in Depth
- Quantifies safety objectives
- translates WENRA safety objectives O2 and O3 in SO1, SO2, SO3
- Uses dose criteria in in line with national radiological emergency planning measures (i.e. for iodine prophylaxis, sheltering, evacuation and food consumption)
- Recent addition: radiological objective for on-site tasks necessary to manage accidents taken into account in the design Provides general approach for external hazards
- Allows for a graded approach to limit hazard evaluation for installations with lower risk profile

As the law of 14 April 1994, modified by the law of 7 may 2017 now allows the FANC to publish binding ”Technical regulations”, some part of this guidance

will be made binding in the near future. Work has started to do this.

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so: • please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view; • please provide concrete examples of more demanding objectives and/or demonstration.
The construction of new NPPs is forbidden by law in Belgium (Law of 31 January 2003 on Nuclear Energy Phase-out). New guidance and regulations, including those for new nuclear reactors and facilities (including the FANC guidance for new nuclear installations) have to be considered in the Periodic Safety Reviews of existing nuclear installations. Application of requirements for new reactors are to be considered in the action plans resulting from the PSR, as far as reasonably achievable. Currently, a new research reactor (MYRRHA) is in a prelicensing stage. The above-referred FANC guideline on “Safety Demonstration” is fully applicable. The Nuclear Safety Objective (art. 8a of the Directive) is explicitly included in the Royal Decree of 30 November 2011 (article 3/1). For the practical implementation of the Safety Objective to new nuclear installations, please see question 3.2 hereafter.
Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?
At the time of the design of the Belgian NPPs, no specific safety objectives associated with different levels of defence-in-depth were defined. Although the FANC guidance on “Safety Demonstration” is applicable to new installations, this guidance is also used as a reference for existing NPPs. The tables below, extracted from the FANC guidance “Safety Demonstration” gives the structure of the DiD (taken from WENRA report “Safety of New NPP Designs” (2013)) and associated design basis categories (see DiD levels 3 and 4 for protective actions).

			Associated Design Basis Categories	
Levels of DiD	**Objective of the DiD Level**	**Qualitative safety objective of the DiD Level (Off-site radiological consequences)**	**Definition**	**Radiological Safety Objective**
Level 1	Prevention of *abnormal operation* and failures	No off-site radiological impact (bounded by regulatory operating limits for discharge)	C1 "*Normal operation*"	GRR-2001
Level 2	Control of abnormal operation and detection of failures		C2 "*Anticipated operational Occurrences*"	SO1
Level 3.a	Control of accident to limit radiological releases and prevent escalation to severe accidents	No off-site radiological impact or only minor radiological impact (part of WENRA Objective O2)	C3a "Postulated single initiating events"	SO2
Level 3.b			C3b "Postulated multiple failure events"	SO2
Level 4	Control of severe accidents to limit off-site releases	Off-site radiological impact may imply limited protective measures in area and time (part of WENRA Objective O3)	C4a[3] "Severe Accidents not practically eliminated"	SO3
		Reduce the risk further	C4b[4] "Severe Accidents practically eliminated"	Not Applicable
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site radiological impact necessitating protective measures	-	

Table 1: Structure of the levels of DiD adopted by the Belgian regulatory authority and associated design basis categories for the deterministic approach

The radiological objectives for each design basis category are coherent with the National Radiological Emergency plan and are given in the table below (from the same FANC guideline):

Design Basis Categories	Radiological Safety Objective:	
C1 "Normal operation"	GRR-2001:	See article 20.1.4 of the GRR-2001
C2 "Anticipated operational Occurrences"	SO1:	For events at least as frequent as once in a year: • Effective dose/event < 0,1 mSv/event; • Equivalent thyroid dose/event for the infant, child or adolescent < 0,3 mSv/event; For events less frequent than once in a year: • Effective dose/event < 0,5 mSv/event; • Equivalent thyroid dose/event for the infant, child or adolescent < 1,5 mSv/event;
C3a "Postulated single initiating events" C3b "Postulated multiple failure events"	SO2:	• Effective dose/event < 5 mSv/event; • Equivalent thyroid dose/event for the infant, child or adolescent < 10 mSv/event. Furthermore, additional criteria of safety objective SO3, notably lifetime dose and restrictions for food consumption should be met with sufficient margin. If relevant this should be demonstrated explicitly.
C4a "Severe Accidents not practically eliminated"	SO3:	• Effective dose/event < 50 mSv/event beyond the evacuation zone. The dose should be integrated over 7 days; • Effective dose/event < 5 mSv/event, beyond the sheltering zone. The dose should be integrated over 24 hours; • Equivalent thyroid dose/event for the infant, child or adolescent < 10 mSv/event during cloud passage, beyond the sheltering zone; • Lifetime effective dose/event < 1 Sv beyond the site boundary and integrated over a period of 50 years after the passage of the cloud. Alternatively it is acceptable if it is shown that the annual effective dose is less than 20 mSv/year for any year after the passage of the cloud • Agricultural products are consumable beyond the sheltering zone within one year after the accident
C4b "Severe Accidents practically eliminated"	Not Applicable	

Table A1: Summary table of off-site radiological safety objectives

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?
At the time of licensing the Belgian NPPs (1975-1985), the concept of DiD was different compared to the one being applied nowadays. Indeed the DiDapproach is now complemented by considering also Design Extension Conditions (DEC) events , including severe accidents, which was not the case in the late 70's or early 80's. For new installations, the safety objectives are described in the "Safety Demonstration" guideline published by FANC in 2017, which is also used as a reference for existing NPPs.
Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.
A particular characteristic of the Belgian nuclear power plants, that merits to be described in some more detail, is their high level of protection against accidents of external origin. Indeed, for the four most recent units, it was requested at the licensing stage that accidents of external origin had to be taken into account, such as an aircraft (civil and military) crash, a gas explosion, a major fire and the effects of toxic gases. These requirements resulted in a duplication and diversification of a significant number of safety systems, installed in bunkered structures to withstand an aircraft crash, which is the most demanding loading case. Moreover, explosive or toxic gases detection systems isolate the ventilation systems in a redundant way in order to prevent the introduction of such gases in the control rooms and of explosive gases in the bunkered part of the installations. This high protection against accidents of external origin resulted in a greater redundancy or diversity in some cases, of the protection and engineered safety systems. For example, the Doel 3 and 4 units, as well as Tihange 2 and 3, are three loop plants equipped with 3 independent and redundant safety trains (each train having its own safety diesel group in a non-bunkered building) and with 3 emergency trains to mitigate accidents of external origin (each train with a diesel located in a bunkered area and built by a manufacturer different from the one of the normal safety diesels, ensuring diversity). The safety trains and the emergency trains are not designed to cope with the same accidents (of internal origin or of external origin respectively) but the emergency trains provide an equipment diversity which can be very useful even for some accidents of internal origin, according to the probabilistic safety studies results. Afterwards, the protection against external accidents for the older units (Doel 1 & 2 and Tihange 1) was also considerably improved but to a lesser extent than that of the most recent units, amongst others by adding dedicated and bunkered systems to these plants. Following the Fukushima Daiichi accident, the licensee was asked to conduct Stress Tests. Safety assessment reports for the Doel and Tihange sites have been established by the licensee and reviewed by the FANC and Bel V and external experts. The basic idea was to look at the robustness of the installation in case of more severe natural hazards than those considered for the design basis. The ability to survive a postulated SBO and a postulated LUHS was also evaluated. Action plans have been developed. Various modifications were made to the facilities or are in the process of implementation. Specific inspections are carried out at Doel and Tihange to monitor the implementation of these modifications. Belgian general requirements on natural hazards are those from issue "T" from the WENRA 2014 safety reference levels, that will be shortly incorporated into the Belgian regulations. The FANC also issued (see page 1 of this questionnaire) specific guidance related to Seismic hazard, (Accidental) Airplane crash and External flooding
Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences

and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms early and large release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

The "Safety Demonstration" guideline via safety objective SO3 quantifies what is considered as a large release. The values adopted are under revision in light of harmonization with practices in other European countries and the IAEA.

For "Early" releases, the time available to implement off-site emergency measures has not been generically defined and depends on the surroundings. This issue is evaluated on a case-by case basis in license applications.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is timely implemented and reasonably practicable. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

For implementing "actions plan" for safety improvements, the licensee has to establish and propose the time frame to the regulatory body, taking into account the importance for safety of the improvements, availability of resources and prioritization of actions. The Regulatory Body will discuss the action plan with the licensee. Delays have to be justified. In addition :

- Highly significant actions plan may require completion of the actions before restart of the reactor unit (several recent examples exist for Belgian NPPs : Flaw indications in Doel 3 and Tihange 2 reactor pressure vessels, reparation of concrete of bunkered buildings to withstand airplane crash, ...)
- A FANC guideline (2010-095) on the conduct of Periodic Safety Reviews specifies the timeframe for implementing the actions results from PSR : from 3 to 5 years with extension in exceptional cases, in order to avoid overlap between PSRs. This guideline will be made binding in the near future.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

The Belgian approach for safety improvements result from a combination of all those elements.
Systematic safety improvements results from ten-yearly periodic safety reviews (PSRs) and from experience feedback (continuous).
In addition, additional safety improvements (action plans) result from non-recurrent situations :
- Safety improvements resulting from the European Stress test
- "Regulatory driven" improvements :
  o Safety improvements for complying with WENRA (2014) requirements
  o Safety improvements for Long term operation (design upgrades imposed by license)
- Improvements from peer reviews :
  o Safety improvements from European peer reviews (TPR)
  o Safety improvements from peer reviews (SALTO, WANO)

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-indepth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.
The "Safety Demonstration" guideline provides an enhanced defense-in-depth approach that is based on the WENRA report "Safety of New NPP Designs" (2013). In the Royal Decree of 30 November 2011, article 20.1 requires that, while considering defense-in-depth, the failure of one barrier is not allowed to cause the failure of another barrier. The NSD does not explicitly identify levels 3 and 4 of DiD, the WENRA report "Safety of New NPP Designs" (2013) defines levels 3a and 3b of DiD (accidents without core melt) and level 4 (accident with core melt). For DiD level 4, the core melt is systematically postulated in order to ensure the definition of mitigation strategies and means. Some elements are : 1° An action plan of the licensee for defining other DEC internal events and DEC hazards is currently in progress. This ensures independent means at levels 3a and 3b (diverse heat sink & diverse diesel generators). For some DEC events, loss of safety features are postulated, for example complete SBO and complete LUHS, leading to the need of (fixed or mobile) ultimate equipment. 2° Filtered Containment Venting Systems are installed at all NPPs, used only for levels 3b and 4 events (ensuring an independence for the containment pressure control between level 3a and levels 3b/4) 3° Reactor Cavity Injection and Alternative Spray will be installed in Tihange (needed only for core melt events), but this is not yet "implemented" at this moment (independence between levels 3a-3b and level– 4)
Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).
Noticeable safety improvements to existing plants have been realized in the frame of European Stress tests. For example : - All Belgian NPPs are now equipped with filtered containment vents - Several civil works have been performed to protect NPP sites (mainly Tihange) against revised design flooding. - On-site emergency centres have been reinforced or renewed A summary of the stress test achievements(status end 2018) is published on the FANC web site : https://afcn.fgov.be/fr/system/files/2019-03-11-best-

2018-final.pdf

Several improvement to the oldest NPPs (Tihange 1 and Doel 1&2) have been implemented in the frame of their Long Term Operation (LTO):

https://afcn.fgov.be/fr/system/files/2012-06-30-electrabel-lto-rapport-technique-tihange-1.pdf https://afcn.fgov.be/fr/system/files/2012-06-30-electrabel.pdf

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.)you require from this level of PSA: • PSA level 1: • PSA level 2: □ PSA level 3: Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies...) required / provided at the different stages (licensing, operation...)?
For existing NPPs, PSA level 1 and level 2 is required by the Royal Decree of 30/11/2011 (which implements the WENRA Reference Levels of 2008 in Belgian regulations). Article 29.3 of the Royal Decree stipulates that the PSA should "be used throughout the design and operation of the NPP in order to facilitate the decision-making process regarding nuclear safety" and that PSA should "be used to show that the design is balanced" (i.e., "that no device or postulated initiating event contributes, in a disproportionate manner, to the global risk and that small deviations of the power plant parameters which may generate a very abnormal behavior, have been avoided"). PSA level 1 and PSA level 2 have been performed since the first Periodic Safety Reviews (i.e. after the initial licensing stage). These PSAs are plant-specific (i.e. a PSA Level 1 for each individual unit and a PSA Level 2 for each representative unit). In the framework of the subsequent Periodic Safety Reviews, these PSAs have been, or are being, kept up-to-date (taking into account plant modifications, operational experience feedback, generic or plant-specific data), further improved (following international Standards or state-of-the-art practices) or extended (e.g. for internal fire and internal flooding). No safety goals are imposed for either PSA Level 1 or PSA Level 2. The end points of PSA Level 2 are frequencies of Containment Failure Modes as well as frequencies of Release Categories (characterized by timing and size, i.e. Early/Late Releases and Very Large/Large/Medium/Small Releases). Level 3 PSA is not performed (and not required in Belgian regulations either). For existing NPPs, the level of detail which is required/provided, during each Periodic Safety Review, follows international Standards and state-of-the-art practices as much as possible. As Belgium has currently a nuclear phase out policy, no licensing of new NPP is foreseen.
Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown...)?
For existing NPPs, Level 1 and Level 2 PSA is expected (Royal Decree of 30/11/2011) in "all the operating modes of the power plant". In the current Royal Decree, it is mentioned that internal initiating events including fire hazards and floods should be taken into account. In the near future (after implementation of the Reference Levels of WENRA of 2014 in the Belgian regulations), relevant external events (e.g. seismic events, external flooding events, and external events inducing Loss of Offsite Power or Loss of Ultimate Heat Sink) will have to be integrated in the PSA.
Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

PSA for spent fuel storage is up to now not required and not performed. In the near future (after implementation of the WENRA Reference Levels of 2014 in the Belgian regulations), PSA for the spent fuel pools (of the NPP) will be required and performed. The level of detail of the PSA for the spent fuel pools is expected to be similar to the level of detail of the PSA for the reactors.
Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?
Up to now it is not expected that PSA covers multiple installations. Some issues could be raised within the context of the application of the new regulations (WENRA Reference Levels of 2014 that will be implemented in the Belgian regulations). All relevant internal and external initiating events will have to be integrated in the PSA. Some external events (e.g. seismic events) could have an impact on several units simultaneously and it will have to be determined how this can be integrated in the PSA. For twin units (i.e. Doel 1 and Doel 2), a full PSA which covers both units will be required to highlight interactions between the units (long term requirement – applicable after 2025)).
Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.
The Royal Decree of 30/11/2011 asks to use the PSA to check the adequacy of the modifications made to the technical specifications. PSA doesn't thus induce any changes to the technical specifications but, in case of adaptations of the technical specifications (i.e. due to other reasons than PSA), a PSA evaluation has to be performed in order to verify that there is no additional risk (e.g. no CDF increase if Level 1 PSA is suitable for the application) due to the proposed technical specification modifications. PSA are used to select and define safety improvements namely in the frame of PSRs and LTO.
Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.
No frequency for PSA update is mentioned explicitly in the Belgian regulations. Article 29.2 of the Royal Decree of 30/11/2011 mentions nevertheless that the PSA should be maintained up-to-date. According to this same article, proven methodologies and relevant international experience should be used (and can thus motivate PSA updates). In practice, frequencies for update are proposed by the licensee. Generally, major updates (including methodological improvements) come together with the Periodic Safety Reviews and intermediate updates (with limited scope and focused on plant modifications or updates of the data) are also organized.
Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so: • What levels of PSA are performed? • What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)? • How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?

• Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?
Currently, in Belgian regulations, PSA is not required for research reactors. Nevertheless, a PSA is developed for one existing research reactor (BR2 ; > 1 MWth). Further: • What levels of PSA are performed? For one existing research reactor (BR2 ; > 1 MWth), a PSA Level 1 has been performed and updated in the framework of Periodical Safety Reviews. A PSA Level 2 is not performed. For new (large) research reactors, consideration is given to PSA in the frame of pre-licensing process. • What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)? Up to now, for the existing research reactor (BR2 ; > 1 MWth), the scope of the PSA Level 1 is limited to internal events (including fires leading to loss of on-site power supplies). Only "operating cycles" are considered. The level of detail is lower than for NPPs. • How are PSAs used in the safety case (whether it be for the application of a new reactor, a license modification/renewal of an existing reactor, or the implementation of a safety improvement)? For the existing research reactor (BR2 ; > 1 MWth), PSA was used to assess the core damage risk and to identify opportunities for improvement (e.g. in the maintenance programs of SSCs). For new research reactors, PSA should be performed within the context of the licensing. • Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)? Currently, in Belgian regulations, PSA is not required for research reactors. Hence, there is no regulatory requirement associated to a threshold either.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.
DEC without fuel melt are associated with the DiD level 3b (see above the FANC guidance on "Safety Demonstration"). The safety objectives for DiD level 3b and DiD level 3a (Design Basis) are the same (see table form question 3.2) in line with the WENRA DiD levels. However, less conservative methodology and assumptions than for design basis can be used for the analysis of DiD level 3b events.
Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?
In accordance with the WENRA Safety reference levels of 2014, DECs without fuel melt are considered as "DEC-A" events. The full set of the WENRA 2014 SRLs is currently being included in the Belgian regulations. The proposed current list of DECs without fuel melt is derived from several sources: a) the plant-specific PSA Level 1 models, which are available for internal events and for some internal hazards (fire, flooding) b) credible combinations of two hazards or events (including internal and external events or hazards) c) finally, the DECs already identified in previous PSRs or during the stress test are included if not yet present in the list of DEC obtained from sources a and b. This list will be then checked by the regulatory body and discussed with the licensee. This method for establishing a list of DEC without fuel melt is applied for the first time and is still under development. A revision is possible, namely within the scope of a future PSR (as part of Safety Factor 5 of the IAEA SSG-25) and/or as a result of experience feedback. This list of DEC is established for existing power reactors. For existing research reactors, a list of DEC has not been established up to now. For new research reactors, the design is still under development. Pre-licensing activities will determine the design basis conditions and the design extension conditions.
Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences. Yes, DEC-A sequences are considered for the spent fuel pool. These DECs are selected deterministically from international sources. Examples are "loss of SFP coolant" and "loss of SFP cooling". Examples of improvements realized in the frame of the action plans resulting from "Stress Tests" are summarized in the document available on : https://afcn.fgov.be/fr/dossiers-dinformation/centrales-nucleaires-en-belgique/stress-tests-nucleaires/rapports In a later stage, when the PSA Level 1 models for the Spent Fuel Pools are established, the adequacy of the list of DECs will be verified.
Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular: • What are the rules to analyse DEC without fuel melt? • Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt? • Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

• The rules that will be followed for the analysis of DEC correspond with those of the WENRA RL2014, Issue F, RL F3.1. • For DEC without fuel melt, radiological safety objectives are defined for new installations (see "Safety Demonstration" guideline) and this might implicitly require to implement specific provisions for respecting these safety objectives. • It is not required explicitly in our regulations but this aspect is taken into account in discussions about safety improvements and safety demonstration with the licensee. See also answer to question 4.1
Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?
In Belgium, for DECs without fuel melt which cannot be managed by means of the safeguard systems foreseen in the plant design, some additional SSC have been installed. Those SSC belong to the safety class "UM" ("Ultimate Means"), implying requirements on: - Withstanding the beyond-design hazards associated to DEC without fuel melt conditions as well as hazard-induced effects and correlated hazards - Independent specific electrical supply (independent from the DBA electrical supplies) - Design according to adequate construction codes and quality-assurance programs - Environmental qualification (including irradiation) - Surveillance, testing and maintenance programs, documented in the technical specifications - Operation described in the FSAR
Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?
No new reactor is currently being built in Belgium.

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.
"Severe conditions" as mentioned in Article 8b of the Directive 2014/87 is defined as "conditions that are more severe than conditions related to design basis accidents"; In our understanding of the Directive, "Severe conditions" corresponds to DEC conditions, including DEC-A (without core melt) and DEC-B (with core melt). Our regulation considers separately DEC-A and DEC-B conditions in accordance with WENRA 2014 SRLs and IAEA SSG-2/1. The objective of DEC-A conditions is to ensure the fundamental safety functions. The objective of DEC-B conditions is to maintain the confinement and avoid releases.
Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?
The list of DECs with fuel melt conditions is derived from the plant-specific PSA Level 2 models, which are available for internal events and for some internal hazards (fire, flooding). The general approach is thus entirely probabilistic. This method for establishing a list of DEC with fuel melt conditions is applied for the first time and is still under development. A revision is possible, namely within the scope of a future PSR (as part of Safety Factor 5 of the IAEA SSG-25) and/or as a result of future (international) experience feedback. This list of DEC is established for existing power reactors. For existing research reactors, a list of DEC has not been established up to now. For new research reactors, the design is still under development. It is not yet decided what will be the design basis conditions and what will be the design extension conditions.
Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.
The list of DECs with fuel melt conditions in the spent fuel pool should be derived from the PSA for Spent Fuel Pools. However, such a SFP PSA is not yet available for the existing power reactors. Hence, a list of DECs with fuel melt conditions in the spent fuel pool might need to be established at a later stage, in the light of the SFP PSA results unless fuel melting in the spent fuel pool can be considered as highly unlikely with a high degree of confidence.
Question 7.4: What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular: • What are the rules to analyse DEC with fuel melt? • Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt? • Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?
• The rules that will be followed for the analysis of DEC correspond with those of the WENRA RL2014, Issue F, RL F3.

• Yes: for example : filtered containment venting systems (FCVS), passive autocatalytic hydrogen recombiners (PARs), etc. • It is not required explicitly in our regulations but this aspect is taken into account in discussions about safety improvement and safety demonstration with the licensee. See also answer to question 4.1
Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?
Similar classification as SSC for limiting the consequences of DECs without fuel melt (see question 6.5), with similar requirements. A “survivability approach” for certain SSC (instrumentation for example) is applied.
Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?
No new reactor is currently being built in Belgium

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?
The Royal Decree of 30/11/2011 (which implements the WENRA Reference Levels of 2008 in Belgian regulations), stipulates the following requirements for the containment function: "20.7.4. Containment functions A containment shall be provided such as if radioactive materials are released into the environment during a design basis accident, the release remains below the prescribed limits. This system can, according to the design requirements, include the following: a) Leaktight structures containing the reactor coolant. b) Related systems for controlling pressures and temperatures. c) Equipment for isolating, managing, retaining or removing fission products, hydrogen, oxygen and other substances which could be released into the atmosphere of the containment. It shall be possible to automatically, and in a reliable manner, isolate any pipe connected to the primary circuit, which is routed through the containment or which communicates directly with the atmosphere of the containment, in the event of a design basis accident during which the leaktightness of the containment plays a key role for preventing the release of radioactive materials into the environment at rates exceeding the prescribed limits. This pipework shall be provided with at least two appropriate isolation devices, placed in series, and it shall be possible to operate each device in a reliable and independent manner. The isolation devices shall be located as near as possible to the containment. Any pipe routed through the containment, which is not connected to the primary circuit and which does not communicate directly with the atmosphere of the vessel, shall be provided with at least one appropriate isolation device. This equipment shall be located outside the containment and as near as possible thereto." "21.4. Protection of the containment against certain beyond design accidents The operation of the containment in the event of an accident sequence not involving beyond design accidents shall be preserved as much as possible, the following in particular: o Isolation of the containment shall remain possible. If an event causes a bypass, the consequences shall be minimised. o The leaktightness of the containment cannot be significantly degraded during a reasonable period of time following a severe accident. o The temperature and the pressure inside the containment shall be controlled during a severe accident. o The combustible gases shall be managed when severe accident occurs. o The containment shall be protected from overpressure during a severe accident. o High-pressure core fusion scenarios in the primary circuit shall be avoided. o The degradation of the containment by melted core shall be prevented or minimised as far as possible." For the Belgian NPPs, the containment functional design for design basis accidents complies with requirements and acceptance criteria stipulated in the USNRC Standard Review Plan 6.2.1 (Containment Functional Design) and 6.2.1.1.A (PWR Dry Containments). The basis functional design requirements are given in the General Design Criteria (GDC) 4, 16 and 50 in Appendix A to 10 CFR Part 50.
Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case

scenario...)?
For the Belgian NPPs, the containment functional design for design basis accidents complies with the requirements and acceptance criteria stipulated in the USNRC Standard Review Plan 6.2.1.1.A (PWR Dry Containments), in particular: • During design and construction, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break. • For operating plants, the peak calculated containment pressure and temperature following a loss-of-coolant accident, or a steam or feedwater line break, should be less than the containment design pressure and temperature.
Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?
For the Belgian NPPs, the containment leaktightness and leakage testing complies with the requirements and acceptance criteria stipulated in the USNRC Standard Review Plan 6.2.6 (Containment Leakage Testing). The basis of the periodic leak rate testing of the containment for the Belgian nuclear power plants is the Appendix J (Primary reactor containment leakage testing for water-cooled power reactors) of 10 CFR Part 50. This document describes the test requirements and the acceptance criteria for 3 types of tests : • Type A test : this test measures the integrated leakage rate of the primary containment structure. • Type B tests: these tests are used to determine the leakage rate of containment penetrations like air locks, electrical penetrations, penetrations with expansion bellows, etc. • Type C tests: these tests are used to determine the leakage rate of the containment isolation valves for each piping penetrating the containment structure. Although Appendix J of 10 CFR Part 50 served as a basis, specific Belgian test requirements and acceptance criteria have been adopted. The Type A test is performed every ten years. For Type B and Type C tests, given the potential impact on radiological consequences and the fact that the Belgian NPPs have a double containment (with an annular space between both containments), leakage rate tests for leak paths by-passing the annular space should be performed after every outage for refuelling, whereas leakage rate testing for leak paths ending in the annular space (which is maintained at subatmospheric pressure and equipped with filtering capacity) should be performed every 40 months (i.e. approximately every 3 years).

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Not really. We consider that practical elimination is achieved when the event is physically impossible or extremely unlikely with a high degree of confidence.

Question 9.2: Do you have requirements related to practical elimination?

See §5.3 of the above mentioned FANC guidance on "Safety Demonstration"

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any): **FANC+BelV** For existing power reactors?

- For new power reactors?
- For existing research reactors?
- For new research reactors?

The concept of practical elimination has been used, at least implicitly, for decades (e.g. RPV failure). Recently WENRA introduced this term and the EU directive introduced a similar (but different?) term of "avoidance". Belgium is active in a WENRA working group that via a pilot study attempts to clarify the approach to practical elimination. When finalized this pilot study can be used to assess and harmonize the concept of practical elimination.

Question 9.4: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

All NPPs in Belgium are equipped with filtered vents.

Question 9.5: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

Pending further guidance by WENRA (i.e. the pilot study mentioned in 9.3), for the moment a case-by-case approach is followed. The WENRA booklet for new NPPs provides some inspiration.

Question 9.6: Is the practical elimination concept applicable also for fuel storage pools?

As severe accidents in fuel storage pools can lead to large and early releases, this concept should be used for the pools.

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?
Yes. See question 10.2 hereafter for more details.
Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account: • Plant design • Actual condition of structures, systems and components (SSCs) important to safety • Equipment qualification • Ageing • Deterministic safety analysis • Probabilistic safety assessment • Hazard analysis • Safety performance • Use of experience from other plants and research findings • Organization, management system and safety culture • Procedures • Human factors • Emergency planning • Radiological impact on the environment
The initial license of all Belgian NPPs included the obligation to perform Periodic Safety Reviews. From 2007 on, the FANC has required (FANC guideline ref. 2010-095) that plant operators perform this periodic safety review following a methodology based on the 14 safety factors described in the IAEA Safety Guides (NS-G-2.10 superseded by SSG-25).

Article 14 of the SRNI-2011 (The Royal Decree of 30 November 2011 on the safety requirements of nuclear installations) requires a ten-yearly periodic safety review for each nuclear installation. The general objectives of these periodic safety reviews are as follows:

- to demonstrate that the unit has at least the same level of safety as it had when the license was granted to operate it at full power, or since its latest periodic safety review;
- to investigate the condition of the unit, devoting more particular attention to ageing and wear and to other factors which may affect its safe operation during the next ten years;
- to justify the unit's current level of safety, taking into account the most recent safety regulations and practices and, if necessary, to propose appropriate improvements.

Article 14.2 of this decree requires that the methodology of the PSR shall include the 14 safety factors as defined by the WENRA reference level P2.2. The Nuclear Safety Objective (article 8a of the Directive) has to be considered in the PSR.

The regulatory body oversees this process by reviewing the analysis and approving the resulting action plan.

For more information, we also refer to the Belgian national reports (2013 & 2016) for the Convention on Nuclear Safety, Article 14. Assessment and Verification of Safety, published on the IAEA web site : http://www-ns.iaea.org/conventions/nuclear-safety.asp

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

The licensee is required by the Belgian regulations (Royal Decree of 30 November 2011, article 14.1), to take into account the developments in international standards and technology. The licensee reports his findings and a proposal for safety improvements to the regulatory body. The regulatory body assesses the licensee's proposal.

To ensure that the most recent developments in international standards have been taken into account, the representatives of the Belgian regulatory body participate in several international groups in relation to nuclear safety, making them knowable of the most recent developments:

- At the European level, FANC and Bel V experts participate in WENRA, HERCA, ENSREG and in their working groups.
- At the IAEA level, the FANC with Bel V participate in the Nuclear Safety Standards Committee (NUSSC), the Waste Safety Standards Committee (WASSC), the Transport Safety Standards Committee (TRANSSC) the Radiation protection Safety Standards Committee (RASSC),– the Emergency Preparedness and Response Standards Committee (EPReSC), the Nuclear Security Guidance Committee (NSGC) and the INES advisory committee.

- At the OECD level, the FANC also participates in the steering committee of the NEA and in the activities of the following NEA committees: the radioactive waste management committee (RWMC), the Committee on Radiation Protection and Public Health (CRPPH), the Committee on Nuclear Regulatory Activities (CNRA) and the Committee on the Safety of Nuclear Installations (CSNI).

  Bel V participates also in the activities of the following NEA committees and working groups: CNRA, CSNI, the Nuclear Science Committee (NSC), the Working group on inspection practices (WGIP), the Working group on operating experience (WGOE), the Working group on fuel cycle safety (WGFCS), the Working group on risk assessment (WGRISK), the Working group on analysis and management of accidents (WGAMA), the Working Group on Electrical Power Systems (WGELEC), the Working Group on External Events (WGEV), the Working group on human and organizational factors (WGHOF), the Working group on integrity of components and structures (WGIAGE), the Working group on fuel safety margins (WGFSM), the RWMC Integration Group for the Safety Case (IGSC), the RWMC Working Group on Decommissioning & Dismantling (WPDD), and in various NEA projects.

The regulatory watch process of the regulatory body also includes developments in the US regulations (NRC and 10CFR).

We also refer to the section II.D.8 (pg.45) "International relations" of the 2016 Belgian report of the Convention on Nuclear Safety.

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?
Safety culture requirements (WENRA reference levels C.7) have been incorporated into the Belgian regulations (the Royal Decree of 30 November 2011) in October 2018. FANC and Bel V have developed a common safety culture policy document. It introduces the following five principles: leadership for safety, promoting individual responsibility, establishing cooperation and open communication, implementing a holistic approach, and ensuring continuous improvement. The references used in the development work were IAEA GSR Part 2 and GS-G-3.5, and OECD/NEA Green Booklet "The safety culture of an effective RB". Both parts of the Regulatory Body developed a process to assess internal safety culture. Regarding safety culture oversight of the Licensees, Bel V has implemented a process since 2012. In a nutshell, this process is based on field observations provided by inspectors or safety analysts during any contact with a licensee (inspections, meetings, phone calls…). These observations are recorded within an observation (excel) sheet – aiming at describing factual and contextual elements – and are linked to IAEA Safety Culture attributes. On an annual basis, a detailed report is released and a synthesis is inserted within the yearly safety evaluation report transmitted to the concerned licensee. The content of this yearly safety evaluation report is discussed with the licensee in order to be sure that the regulatory concerns are understood. Pluriannual safety culture assessments are also performed in order to obtain a deeper cultural picture of a nuclear installation. On the basis of the annual and pluriannual safety culture assessments different type of safety culture inspections are scheduled. See also Section II. F Article 10 - Priority to Safety - of the Belgian report for the Convention on Nuclear Safety (2016 edition).
Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.
According to the Royal Decree of November 30, 2011, licensees have to take into consideration the lessons learned from national and international operating experience (art. 4.2 / art. 11). In particular, licensees should implement an OPEX process (art. 11.1), develop methods for event analysis including human factors issues (art. 11.3), and organize the documentation, i.e. recording and retrievability of events (art. 11.4). OPEX should be specifically considered in the framework of the training (art. 11.5 / art.19), ageing (art. 10.3) and maintenance (art. 12.2) programmes, the design of the installation (art. 20.9) and the revision of procedures (art. 27.4).

Concerning the notification of events by the licensees, the FANC Directive (2010-054) describes the way to declare events (type of event, communication channels, sending date...) to the Regulatory Body.

Appendix 3: Answers to the questionnaire - Czech Republic

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

Revision R2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: State Office for Nuclear Safety, Czech Republic

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

The "safety philosophy" of Czech national nuclear legal framework is expressed in introductory part of the Atomic Act No.: 263/2015 Coll. §5, "Principles of the peaceful use of nuclear energy and ionising radiation". In article 1) it is required that:

(1) Anyone who uses nuclear energy or performs activities in exposure situations shall

a) precede radiation extraordinary events and, if they occur, ensure that radiation extraordinary event management procedures are followed and minimise their consequences,

b) ensure the safe performance of these activities and protection of natural persons and the environment from the effects of ionising radiation and

c) act in a way ensuring that the risk to natural persons and the environment is kept as low as can reasonably be achieved taking into account the current state of technical knowledge and economic and societal aspects.

The article (2) requires that:

(2) Anyone who uses nuclear energy, manages a nuclear item or performs activities in exposure situations shall:

a) as a matter of priority, ensure nuclear safety, safety of nuclear items and radiation protection, while respecting the present level of science and technology and good practice,

b) perform assessment of intention to perform activity and of its expected results from perspective of their benefits for the society and individuals (hereinafter „justification"),

c) within the justification take into consideration techniques which do not use nuclear energy and ionising radiation, but which can provide comparable results,

d) perform only activity with benefits for the society and individuals prevailing over a risk emerging from the activity or its consequences; such activity is considered as justified,

e) perform justification repeatedly whenever new and important evidence about the efficacy or potential consequences of the activities performed or new and important information about other techniques or technologies is available.

For limiting of maximal consequences of the operation of nuclear installations, the safety objectives are established in the Decree no. 329/2016 Coll. in compliance with Articles 8a-8c of Directive 2014/87/Euratom, valid for existing and planned nuclear installations (see answer in following Article 3).

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

The new Czech nuclear legislation (the Atomic Act No. 263/2016 Coll., and other related implementing Decrees) doesn't differentiate between existing and new reactors. However, it is supposed that the new requirements and safety objectives will be applied to new reactors in larger extent, compared to existing ones, simply due to fact that it is more "reasonably practicable" .

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

"The implementation of the Safety Directive 2014/87/ Euratom is realised by new Czech nuclear legislation. All these documents implement fully recommendation of the WENRA Reference Levels 2014. Considering Article 8b its requirements are fully incorporated in the SÚJB Decree No: 329/2017 Coll., on requirements for the design of nuclear facility.

The new legislation includes technical acceptance criteria and radiological acceptance criteria defined for the different plant states. The limiting values for radiological risks are established for normal operation and AOOS, for DBAs and DEC A and separately for DEC B.

The new legislation requires off-site emergency plans for protection of the population, as well.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

See the answer to the question Q 3.1 - the new Czech nuclear legislation doesn’t differ between existing and new reactors.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Requirements for coping with external hazards are specified in SÚJB Decree No. 378/2016 Coll. on siting of nuclear installation and SÚJB Decree No. 329/2017 Coll. on requirements on nuclear installation design. These requirements correspond to recommendations of WENRA Reference Levels 2014. The methodology used for application of the legislation requirements requires evaluation of all hazards relevant to the site, assessment of frequencies of occurrence and corresponding magnitudes and design solution of protection concept against them.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires “preventing accidents and, should an accident occur, mitigating its consequences and avoiding” “early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

The SÚJB Decree No. 329/2017 Coll. requires to ensure that the following are practically excluded events:

1. a radiological accident where there is not sufficient time to implement urgent action to protect the population (hereinafter referred to as 'early radiological accident') and

2. a radiological accident requiring urgent action to protect the population that cannot be limited in terms of location or time (hereinafter referred to as 'large radiological accident')

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

According the Atomic Act requirement, the licensees were required to adapt their nuclear facilities to the new legal framework, unless otherwise provided in this Act, within two years of the entry into force of this Act (the end of 2018). There is no specific definition of what is meant by "timely implemented".

The term "reasonable practicable" is specified in the SÚJB Decree No. 329/2017 Coll., §6, art. (6). The term *"reasonably practicable"* means *"reaching of compliance with a requirement set out in this implementing Decree when the risk of a radiological accident due to insufficient capability of the nuclear installation to meet the set safety objectives is reduced, while the reasons and conditions for using of the nuclear installation do not change significantly.*

Adequate application of this approach is evaluated for licensing purposes "case by case".

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

The continuous improvement of nuclear safety is assured, above all, by periodic safety reviews performed in 10 years intervals.

The SÚJB Decree No. 162/2017 Coll. on the safety assessment by the Atomic Act specifies in the part concerned to PSR following factors to be addressed:

- compliance of NPP reactor with actual state of its design requirements and technical specification,
- actual state of reactor systems, structures and components qualification documentation
- results of its ageing management.
- implementation of results of this review to actual probabilistic safety analysis
- evaluation of compliance of the reactor actual state with requirements based on actual state of technology, science and international operational feedback."

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Updating of Czech regulatory framework was realised by new Czech legislation, namely by the Atomic Act No.: 263/2016 Coll., and by SÚJB Decree No: 329/2017 Coll., Decree No. 21/2016 Coll. and other relevant Decrees.

All these documents implement fully recommendation of the IAEA SSR-2/1 (Rev. 1) and WENRA Reference Levels 2014. In these legal documents the levels of defence in depth are not strictly named but all principles concerned to function of different level of defence in depth for different categories of design extension conditions and the independence of the levels of DiD are strictly required.

The Atomic Act strictly requires also the Management System application to all activities influencing Nuclear safety through Decrees No. 408/2016 and No. 358/2016 Coll. and requires development of the Safety culture (§ 30 of Atomic Act and §5, art. D of the Decree V 21/2017). The processes of Periodic Safety Review , fixed on basis of long term experience (among others processes of ongoing Safety Assessment) by Decree No. 162/2017) led to preparation of continuous safety improvement of existing NPP in all levels of defence in depth. The modifications and improvements prepared and still not realised, were also basis for the Post – Fukushima National Action Plan. The Stress – tests review led in principle only to limited additional findings.

The Post-Fukushima National Action Plan included particularly and among others, provisions strengthening the level 4 of DiD.

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

The most noticeable safety improvements at existing Czech plants were realized under not only after the Post-Fukushima National Action Plan (NAP) which summarized findings and conclusion of the stress tests as well as outcomes of peer reviews of stress tests organised by ENSREG in April 2013 but also relatively long term before. These modifications were for example the alternate safety cooling and heat sink, strengthening of AC/DC power supplies by mobile devices and by bunkered/hardened diesel generators systems, molten core in-vessel retention system and strategy for VVER 440 units, etc. The NAP is almost fully implemented. Actually only measures for protection of the Temelin NPP containment against molten core debris influence (No.50) are delayed due to the technical problems and high uncertainty of analyses.

The NAP of the Czech Republic is a live document, which has been regularly revised taking into account the latest technical knowledge and operational experience.

Revision in the English language is available http://www.sujb.cz/fileadmin/sujb/docs/jaderna-bezpecnost/Czech_National _Action_Plan_rev2.pdf.

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- PSA level 1:
- PSA level 2:
- PSA level 3:

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

For activities related to the use of nuclear energy the SÚJB requires PSA as a part of the documentation to be provided during the licensing process.

PSA study is a licensing document for:

- the construction of a nuclear facility
- the operation of nuclear facility
- the individual phases of decommissioning of a nuclear facility
- the carrying out of modifications affecting nuclear safety

PSAs that are performed for individual periods of the life cycle may be based on generic reliability data for equipment, but the specific data for relevant external hazards must be used. After accumulating of experience from the nuclear facility operation the use of specific data is generally required as much as possible.

Pursuant to SÚJB Decree No. 162/2017 Coll., on the requirements for a safety assessment by the Atomic Act the PSA has to be used:

(1) during the life cycle of a nuclear facility in the evaluation of important information on the risk and the consequences of the use of nuclear energy,

(2) to reduce the risk which represents a nuclear facility to detect the need of changes in design of nuclear facility resulting from deficiencies of nuclear facility or its operating procedures,

(3) for determination of priorities in planning of measures to increase the level of nuclear safety, radiation protection, security and emergency planning,

(4) for the evaluation of the overall risk which represents a nuclear facility,

(5) to verify

a) the equilibrium of risk contributors in the design of the nuclear facility,

b) the absence of minor deviations of the nuclear facility characteristics from their usual values defined in the legislation, which can cause a significant reduction in the level of nuclear safety

c) the absence of important elements of the nuclear facility design or group of initiating events representing a disproportionately large contribution to the overall risk (balanced risk distribution)

d) the proportional influence of factors which are set with a significant uncertainty to the PSA on achieving an overall low level of risk, which represents a nuclear facility.

(6) for the evaluation of

a) the necessity and safety acceptability of the nuclear facility changes,

b) the necessity and safety acceptability of changes of the TechSpecs,

c) the necessity and acceptability of changes of the operational procedures, and

d) the severity of different events occurring at the nuclear facility,

(7) for verification of emergency operating procedures accuracy,

(8) to check whether In-service Inspections Programme includes all SSCs which have influence on nuclear safety,

(9) to check whether all SSCs with an impact on nuclear safety are subject of the process of aging management,

(10) to identify the SSCs with an impact on nuclear safety, whose operability must be always ensured

(11) as input information for preparation and validation of significant training programmes of the licensee, including simulator training of control room operators

(12) for monitoring of the risk level during the operation of the nuclear facility, which must be carried out (i.e. risk monitor tool must be used).

Pursuant to SÚJB Decree No. 329/2017 Coll. on requirements on nuclear installation design (§ 23(3)) nuclear installation design shall set requirements for systems, structures and components of the nuclear installation relevant to nuclear safety so that the likelihood of fire at the site where they are located is as low as reasonably practicable and selected equipment (SSCs important to safety) shall resist to the effects of fire and maintain its ability to perform safety functions.

§ 25, art. 3 and § 28 require that in the framework of the evaluation of the safety of the design, in case of the design extension conditions, analyses of the design extension conditions management must be performed. These analyses must take into account the results of the PSA level 1 and 2.

The application of PSA level 3 is not required by Czech nuclear legislation due to actual situation in its application by international nuclear community. The methodology for realisation of the PSA level 3 is not stabilised, the IAEA is actually preparing a new guidance (the previous document is from 1996 year and it is considered as obsolete). The important technical problem is also the influence of uncertainties to the results of the analysis.

Nevertheless the acceptance criteria for results of the PSA level 2 analyses are required to be justified by analyses of spreading of radionuclides of corresponding source terms and consequent exposures (radiological consequences).

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

Pursuant to SÚJB Decree No. 162/2017 Coll. on the requirements for a safety assessment, by the Atomic Act, a full scope PSA is required (i.e. all relevant internal initiating events, all relevant internal and external hazards, evaluation in PSA level 1 and 2). Pursuant to the same Decree, the uncertainty analysis and sensitivity studies must always be carried out as an integral part of the PSA.

The main application of the PSA level 1 and 2 is the verification of the safety goals (as mentioned in the document Basic Safety Principles for Nuclear Power Plants. 75-INSAG-3 Rev. 1, INSAG-12. IAEA, Vienna, October 1999),"

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Yes. Pursuant to SÚJB Decree No. 162/2017 Coll. on the requirements for a safety assessment by the Atomic Act, art.5, the PSA shall cover both the reactor and the spent fuel storage with the same level of details.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Pursuant to SÚJB Decree No. 162/2017 Coll. on the requirements for a safety assessment by the Atomic Act, Art.. 6, interdependencies among various nuclear facilities, located at the same site, must be taken into account in the PSA model.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

As mentioned above, pursuant to SÚJB Decree No. 162/2017 Coll. on the requirements for safety assessment by the Atomic Act, the PSA shall be used for the safety evaluation of the necessity and acceptability of changes of the Technical Specifications (TSs).

The licensee (CEZ Ltd.) realised a large-scale project for the Dukovany NPP on the re-assessment of existing TSs from the point of view of the risk in nineties of last century. This activity led to proposal for softening of TSs, not finally accepted by Regulatory body. Actually similar project is planned for year 2019 for both NPPs with goal to take into account actual state of units and its PSA studies in review of TSs adequacy.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Pursuant to SÚJB Decree No. 162/2017 Coll. on the requirements for a safety assessment by the Atomic Act, Art. 9

(1) PSA must be updated within 12 months after implementation of changes in the use of nuclear energy, which could be affected by this change, if this change has the influence on

a) the site of the nuclear facility ,

b) the current status and the operation of a nuclear facility when change of the design of a nuclear facility or change of the methods of testing and maintenance of nuclear facility was performed,

c) the current status of the operational procedures,

d) the reliability data of the SSCs and the probability of human errors, mainly based on specific data from nuclear facilities of a similar type,

e) the current technical information about the status of the nuclear facility,

f) the current information on the properties and behaviour of the nuclear facility when safety important operating event occurred.

(2) PSA must be checked, updated and upgraded at least once every 5 years of nuclear facility operation (update of reliability data, the use of the currently available analytical methods and tools corresponding to the best international practice).

(3) PSA study for Dukovany NPP, which is included in Chapter 19 of FSAR, is updated, as well, as the whole FSAR each year. The Chapter 19 includes essentially the Summary Report from Living PSA project. The PSA study for Temelín NPP will be updated in the same way in the context of the process of unification of the content of the FSAR for both Czech NPPs in the near future.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed?
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

Yes, research reactors with thermal power higher than 2 MW must have a PSA study level 1 and 2.

In the Czech Republic this duty concerns the research reactor LVR-15 at Research Centre Rez. Its PSA study is an obligatory part of the licensing safety documentation.

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis” as mentioned in Article 8b? If so, please explain.

In the new Czech nuclear legislation the number of the DiD levels isn´t explicitly prescribed. However the SÚJB Decree No. 329/2017 Coll. on requirements for the design nuclear facility defines three categories of accident conditions for which it specifies the basic design requirements:

(a) design basis accident conditions, where the proper functioning of the safety systems ensures that the relevant reference or exposure limits are not exceeded,

(b) design extension conditions for accident conditions caused by scenarios that are taken into account in the design of a nuclear installation,

(c) severe accidents involving serious damage to nuclear fuel by serious degradation of and irreversible loss of the structure of the core of the nuclear reactor or the nuclear fuel storage system by melting of nuclear fuel assemblies.

Taking into account the marking of DiD levels according WENRA, the design extension conditions without core melt pertain to third level of DiD (3b) and the safety analyses results of these shall fulfil the same acceptance criteria for radiological consequences, as analyses of Design Basis Accidents (3b). The difference between these is in application of conservative approach to the analyses. For the analyses of events and scenarios resolved in the level 3b of DiD also results of realistic approach to safety analyses could be accepted according Czech Nuclear legislation.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

List of scenarios for DECs must be determined on the basis of engineering judgment with the use of deterministic and probabilistic methods.

The applicant shall use a recommendations of safety standards (national and international) containing the recommended list of DECs and also use the plant specific PSA study to verify selected and identify others DEC scenarios which are plant specific and which are not included in safety standards.

Regulatory body checks the list of scenarios used for identification enveloping scenarios of DECs without fuel melt for licensed plant.

The review of the list adequacy shall be made within the scope of PSR.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Yes, these scenarios are analysed to verify fulfilment of adequate acceptance criteria, e.g. for total loss of SFP cooling safety systems.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?
- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

What are the rules to analyse DEC without fuel melt?

The rules to study DEC without fuel melt are:

– the realistic approach should be used,

- a single failure in systems, structures and components may not be considered,
- use of non-safety systems may be considered,
- systems, structures and components considered by design to be used during DECs must have sufficient capacity and must be proven that they can to fulfil safety function even under the DECs conditions for a necessarily long time.

Do you require provisions dedicated to the mitigation of DECs without fuel melt?

Yes, the general requirements are set out in the SÚJB Decree No. 329/2017 Coll., on requirements for design of nuclear facility (e.g. general DiD requirements, basic operability requirements, etc.)

Do you expect provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

Not directly, but in general the SÚJB Decree 329/2017 Coll. requires:

(1) Nuclear installation design shall, in the context of ensuring compliance with requirements for the application of defence-in-depth, ensure that a failure of a system, structure or component or loss of a safety function at one level does not reduce the effectiveness of the safety functions at the subsequent levels of defence-in-depth necessary to remedy or mitigate the consequence of an initiating event.

(2) In order to create systems of subsequent defence-in-depth levels, the nuclear installation design may only use those systems, structures and components of the systems of the preceding defence-in-depth level that has been broken which

a) have not been compromised in the course of the development of the nuclear installation's response to an off-site or on-site initiating event or scenario and

b) are separable from the compromised or unusable parts of the systems of the preceding defence-in-depth level that has been broken.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

In accordance with the SÚJB Decree No. 329/2017 Coll., SSC are divided in order to ensure the fulfilment of the safety functions into these categories:

a) SSC without impact on nuclear safety,

b) SSC with impact on nuclear safety which are not the “selected equipment”, and

c) the SSCs which are “selected (classified) equipment” are subdivided into

1. SSCs classified as “selected equipment” which are not safety systems;
2. safety systems.

The design of a nuclear installation must specify requirements for both

- SSC with impact on nuclear safety, which are not the “selected equipment” but still can be used for prevention and mitigation of DEC (alternative provisions)

and for

- “Selected Equipment” (including also diverse provisions to safety systems).

The rules for the Safety Classification of SSC according its safety function are specified in the Appendix 1 of the Decree 329/2017 Coll. giving Safety Classes BT1 (principally for primary circuit boundary), BT2 (principally for Safety Systems) and BT3 (Principally for other SSCs important to safety). This classification serves only for graded approach to the Quality Assurance of the “Selected Equipment”, it means to methodology of quality management and review of compliance of SSCs with Technical Specification done by design during the whole lifetime of the SSC (including redundancy, electrical supply, qualification….etc.)

SSCs for limiting the consequences of DECs without fuel melt are classified in Safety Class BT3 (of Selected Equipment), permitting also the industrial quality without any additional specific associated design requirements. Nevertheless, such additional specific associated design requirements may result from PSA studies depending on their safety importance in limiting accident consequences.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

The list must be plant specific (using PSA), so the list for new NPP should differ. On the other hand the current legislation of Czech Republic doesn´t make difference between currently operated and newly build NPP in its requirements.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

DEC B means in the Czech Republic "severe accident conditions". The terms “severe conditions” and “severe plant conditions” used randomly and colloquially in the IAEA INSAG 10 are not defined in the IAEA Glossary and there are used only in citation of the INSAG 10 concerning DiD. From legal point of view these terms were evaluated as indefinite and unusable in our nuclear legislation. But in this context it is very important, how the terms “severe accident” and “severe accident conditions” are defined (see our Decree No.:329/2017 Coll., Art. 2, k) :

‘severe accident’ means accident conditions involving serious damage of nuclear fuel either due to serious damage to and irreversible loss of the structure of the core of the nuclear reactor (hereinafter referred to as the ‘core’) or the system for nuclear fuel storing due to damage to fuel assemblies as a result of nuclear fuel melt,

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

During DECs with fuel melt the safety objectives for the NPP must be met.

List of scenarios for DECs must be determined on the basis of engineering judgment with the use of deterministic and probabilistic methods. The PSA must be used, but it does not mean that it is possible to not introduce any mitigation measures for coping with a severe accident simply because of the very low probability of its occurrence.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Yes, it does, e.g.:

- Events in the cooling system spent fuel storage pool
    - Opening the drainage line from the spent fuel pool
    - Coolant leak from the spent fuel storage pool
    - Loss of Cooling of the Spent Fuel Storage (for different refuelling states)
- Damage to the spent fuel in the pool storage during refuelling
    - Damage to the fuel assembly by the refuelling machine
    - Falling of a spent fuel assembly into a spent fuel pool
- Fall of a heavy object into a pool
- Events related to the change of reactivity in the spent fuel pool
    - Reduction of boron concentration in spent fuel pool as a result of operational personnel error
    - Leakage of cooling water through a leaky heat exchanger

- Long-term replenishment of the pool with non-borated water using mobile equipment

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?
- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

What are the rules to analyse DEC with fuel melt?

The deterministic analysis of the accident scenario must be based on clearly defined initial and boundary conditions determining:

- operating mode at initiation event,
- initiation event (if the initiation event can by caused by station blackout (SBO), it is also necessary to analyse the SBO option);
- expected system failures and operator errors.

The PSA-1 and PSA-2 probability safety analysis must also be included. PSA-3 is not currently required by SÚJB, however, if it exists, its results may be used in the safety report. PSA models must match the current state of the nuclear facility under consideration. PSA must include internal and external events for all modes of operation. The PSA must be specific to the site of the nuclear facility. PSA must include the mutual influence of all nuclear installations in the site.

When simulating fission product release, natural deposition and capturing phenomena can be considered due to the operation of safety systems up to the pressure boundary of the containment, or the output of a filter system designed for accident conditions

Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?

The design of a nuclear facility must include requirements for diverse and alternative means and procedures to protect the integrity of the containment system, which allow in the case of the fuel melting to a reasonably practicable extent

a) retention of the corium within the hermetic zone of the containment,

b) suppression of reactivity in the melted corium,

c) long-term cooling of the corium with heat transfer to heat removal systems from the containment and

(d) maintaining the capability of the containment system to contain the radioactive substances within its hermetic zone.

Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

No, there are no specific requirements for independency of provisions for DECs with fuel melt from provisions for other plant states as these, applied for independence of DiD levels – see answer to Question 6.4..

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

SSCs for limiting of consequences for DECs with fuel melt are classified in Safety Class BT3 (of Selected Equipment) or as SSC with impact on nuclear safety which are not the “selected equipment” as SSCs, supporting the safety function providing for limited time. This classification permits the industrial quality without any additional specific associated design requirements. Nevertheless, such additional specific associated design requirements may result from PSA studies depending on their safety importance in limiting accident consequences.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

The list must be plant specific (using PSA), so the list for new NPP should differ. In the other hand the current legislation of Czech Republic doesn´t make in design requirements difference between currently operated and newly build NPP.

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Regulatory requirements are explicitly stated in Art. 43 paragraph 4 letter b and Art. 44 para. 2 of the SÚJB Decree No. 329/2017 Coll., on requirements for nuclear installation design to set acceptance criteria for securing and protecting the functions of containment, including design limits of temperatures and pressures inside the containment, leak tightness and permissible deformations of its construction.

Since the containment is the last physical barrier against radioactive substances leak into the environment, further technical and organizational measures against exceeding the acceptance criteria that are carried out during design basis accidents and in design extension conditions without severe damage of the nuclear fuel at minimum during the time until the implementation of measures necessary for reaching the safe state of the nuclear facility are made.

After the occurrence of a severe accident, the containment system functions shall be provided at least for the period needed for taking measures to manage the severe accident and radiological emergency.

Art. 45 paragraph 4 of the SÚJB Decree No. 329/2017 Coll. furthermore says that the nuclear facility design has to set the requirements for diverse and alternative means and procedures for the integrity protection of containment system, which in case of the core meltdown allow the catching of the core melt inside the hermetical space within reasonably practicable measures for suppression of reactivity within the melt of the core, long-term cooling of the core melt with heat dispersion into the containment heat sink systems and for sustaining of the ability to catch the radioactive substance inside the hermetical space of containment.

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

Design pressure and temperature are based on thermomechanical analysis of design basis accident, i.e. (large LOCA) guillotine break of the primary circuit. The resulting design pressure and temperature for the containment were set by the designer to higher value of supposed effects of maximum design accident. For technical design of the containment itself the margin was increased taking into account the uncertainties of procedures for the design, construction, control and testing of the engineering structures by using of partial safety coefficients, for individual loads and their combinations and physical-mechanical characteristics of used materials. The real safety margin for design extension conditions will increase also due to permitted realistic approach to the analysis in frame of accident studies.

Objective validation of design bases for certain type of NPP on higher than planned pressure (approximately by 15%) was carried out after the construction completion and before the start up of the nuclear facility at both types of nuclear facilities containments (Dukovany NPP – containment with bubbler condenser pressure suppression system and Temelín NPP - full pressure containment), in accordance with regulatory requirements.

The requirements for the containment system testing are stipulated in SÚJB Decree No. 329/2017 Coll, (Art. 9 paragraph 6 letters (a) to (c) and in Art. 45 paragraph (1).

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

Requirements for penetrations are stipulated in Art. 43 para.3a and extended in detail in Art.44 of the SÚJB Decree No. 329/2017 Coll. The containment system must, according to the nuclear facility design, consist of systems securing the separation of hermetical space from outer tubing and cabling systems in the area of their penetration and hermetical closing of passages to hermetical space. This requirement also applies to design extension conditions and it is used for both the pipeline and cable penetrations through the containment wall and entrance and emergency passages for personnel.

Containment penetrations are regularly tested in all nuclear facilities with containment systems (Dukovany NPP and Temelín NPP) in Czech Republic. This requirement is stated in Article 45 para.1 of the SÚJB Decree No. 329/2017 Coll.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

"The definition is in SÚJB Decree No. 329/2017 Coll., §.2(a):

"practically eliminated matter means a condition, state or event, the occurrence of which is considered physically impossible or which is, with a high degree of confidence, very unlikely".

Question 9.2: Do you have requirements related to practical elimination?

There are several requirements related to the "practical elimination"" concept specified in the Decree No. 329/2017 Coll., on requirements for the design of nuclear facility, in particular Article 7, paragraphs 5 to 7. Its application is mainly connected with analyses of design extension conditions as required in the Issue F of WENRA Safety Reference Levels (SRL).

While SRL F1.1 requires to perform analyses verifying possibilities of enhancing NPPs capability to withstand more challenging events or conditions than those considered in their design basis, SRL F2.2 allows to exclude consideration of events with a high degree of confidence to be extremely unlikely to occur"". Although the term ""practical elimination"" is not used in the SRLs, the concept is effectively the same. We have allowed use of this condition as justification of acceptability of the current design in cases where it is not fully capable to cope with certain specific low-frequent events, e.g., when verifying capability of the Dukovany NPP to withstand software common cause failure in the digital reactor protection system occurring concurrently with individual postulated initiating events.

Question 9.3a: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

For the existing NPPs it has been already used several times in particular in connection with analyses of design extension conditions (see answer to Question 9.2).

This concept is also applied (based on the PSA analysis) for existing research reactor LVR-15 at Research Centre Rez in excluding a severe accidents with early or large release from safety considerations.

Question 9.3b: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

In our understanding this concept is not directly intended to ""improve operating reactors"" but primarily to verify that the existing design (or proposed design for new NPPs) is sufficiently safe and adequately reducing risk of early or large off-site radioactivity release. If this verification reveals that the existing/proposed design is not adequate in certain aspect to the legislation requirements the Regulatory body requires corresponding improvement of the design, i.e. requires implementation of safety improvements. The safety improvements, till now gradually prepared and realised were described before in answer to Question 4.2.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

The demonstration of “practical elimination” can be bases only on case by case approach.

For the specific cases, where this concept has been used so far, it served to demonstrate that the residual risk not excluded by implemented NPP safety features is sufficiently low, i.e. that the early or large release due to considered un-mitigated complex events can be considered extremely unlikely with a high degree of confidence. As those demonstrations have been acceptable to us no safety improvements have been required so far in the reviewed cases.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

The Czech nuclear legislation allows to use this concept in design of any nuclear facility including the facilities for storage of the spent fuel.

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

The PSR has been a part of legal requirement since mid of 90-ties . It has been based on IAEA recommendation.

Currently the PSR is explicitly required by the new Atomic Act and its scope is defined in SÚJB Decree No. 162/2017 Coll. on the safety assessment.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

According § 13 of the Decree No. 162/2017 the periodic safety review must compare the state of nuclear safety, radiation protection, technical safety, radiation monitoring, accident management and physical protection at the nuclear installation with the requirements of the legislation and the requirements arising from the current state of science and technology and good practice in force at the time of its performing.

Periodic safety review must systematically and comprehensively investigate at predetermined intervals:

(a) the nuclear facility design,

(b) the actual state of the systems, structures and components,

(c) the capability of systems, structures and components to fulfill the functions required by the nuclear installation design (facility qualification),

(d) aging of systems, structures and components,

(e) deterministic safety analyses,

(f) probabilistic safety assessment,

(g) risks analysis,

(h) operational safety,

(i) exploitation of operational feedback from other nuclear facilities and science and research,

(j) organization and management,

(k) operating procedures and regulations

(l) human factor,

(m) accident management and

(n) the impact of the operation of the nuclear installation on its surroundings in terms of radiation protection.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

See answer to Q 10.2 above - the PSR must consider requirements arising from the current state of science and technology and good practice widely used at the time of its performing.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Safety culture should be promoted across licensee's management system.

Requirement in new Atomic Act (Act No. 263/2016 Coll.) Art.30 paras 7 and 9 stipulates: "(licensee) shall introduce the management system in a manner ensuring that through this system are permanently developed and regularly evaluated characteristics and attitudes of persons performing activities related to the use of nuclear energy and activities in exposure situations and of their personnel, which ensure that nuclear safety, radiation protection, technical safety, radiation monitoring, management of radiation extraordinary event and security are approached with a seriousness corresponding to their safety importance (hereinafter „safety culture).Also dialog with licensee should be established to achieve common In compliance understanding in this matter with the Atomic Act applicant/licensee have issued documents which are promoting safety culture and evaluate it on the basis of questionnaires regularly distributed to the operating personnel. The Decree no. 21/2017 Coll, Art. 5, (1) d) requires the investigation of operational events together with evaluation of the impact of the safety culture.

The principles defined in INSAG-4 have been followed in implementation of Safety Culture concept on Czech NPPs.

, ..

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

"The new nuclear legislation - the Atomic Act and in particular the Decree No. 21/2017 on assuring nuclear safety define requirements and conditions for implementation of the operating feedback in the Czech nuclear facilities. Any event that has a negative impact on nuclear safety or radiation protection shall be classified among operational events and shall be evaluated by the feedback system.

The feedback system must also use the relevant experience from other power plants in the Czech Republic and abroad. The licensee must ensure that the feedback system draws conclusions from the analyses and that appropriate corrective action is taken to prevent recurrence and adverse safety developments."

Appendix 4: Answers to the questionnaire - Finland

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

Revision R2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: Radiation and Nuclear Safety Authority (STUK)

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

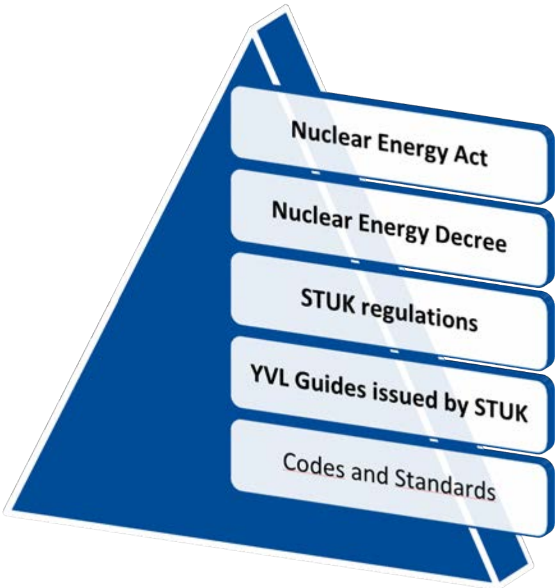
The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

The structure of Finnish nuclear safety regulations:



Finnish regulations and safety requirements are regularly updated considering operating experience and safety research and advances in science and technology. STUK regulations are binding. The objectives of safety requirements in YVL Guides are binding on the licensee, while preserving the licensee's right to propose an alternative procedure or solution to that provided for in the regulations. If the licensee can convincingly demonstrate that the proposed procedure or solution will implement safety level in accordance with the Nuclear Energy Act, STUK may approve this procedure or solution.

The general rules for comprehensive and systematic periodic safety assessments at existing nuclear facilities are presented in the Finnish Nuclear Energy Act. Section 7 a also requires that "The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology."

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

Finnish regulations and safety requirements are regularly updated considering operating experience and safety research and advances in science and technology. STUK regulations are binding and possible exemptions for certain nuclear facilities are written in the end of regulation (transitional provisions). Current exemptions are included in STUK regulation Y/1/2018 (in English https://www.finlex.fi/fi/viranomaiset/normi/555001/42574) stating that certain Sections in the regulation (Section 10(3) subparagraph (c) containment building integrity, Section 11 safety functions and provisions for ensuring them, Section 14 protection against external hazards affecting safety, and Section 16(4) supplementary control room) shall be applicable to a nuclear power plant unit for which an operating licence was issued prior to 1 January 2016 (meaning Loviisa units 1 and 2 and Olkiluoto units 1 and 2) to the extent required with respect to the technical solutions of the facility in question, under the principle laid down in Section 7 a of the Nuclear Energy Act (continuous safety improvements principle, see the answer to previous question). The details are dealt at the YVL Guide level.

The revised regulatory guides (YVL Guides) are applied as such for new nuclear facilities. For the existing facilities and facilities under construction, separate facility specific implementation decisions are made. Before an implementation decision is made by radiation and nuclear safety authority (STUK), the licensees are requested to evaluate the compliance with the new guide. In case of non-compliances the licensee has to propose plans for improvement and schedules for achieving compliance. After having heard those concerned, STUK makes a separate decision on how a new or revised YVL Guide applies to operating nuclear facilities, or to those under construction. STUK can approve exemptions from new requirements if it is not technically or economically reasonable to implement respective modifications and if safety is justified and considered adequate. This is case by case decision.

There are many safety improvements carried out at the operating Finnish NPPs that have been considered reasonably practicable (examples given in relation to next questions). On the other hand, typically plant modifications requiring major plant layout changes have been considered not reasonably practicable. For example currently operating NPPs don't fulfil all the modern requirements for separation of safety systems. Also the seismic hazard was not included in the design basis at the Finnish NPPs during 1970's. In these cases, reasonably practicable plant safety improvements have been identified for example by using the PRA model. Another example, where safety improvements have not been considered reasonably practicable, is the protection against large civil airplane crashes (APC). The requirement was introduced for new NPPs in Finland after the 2001 terrorist attack in the USA. For operating NPPs, it was considered not reasonably practicable to backfit any major structural modifications. However, whenever major modifications are done, APC must be considered (for example, when an enlargement of the spent fuel interim storage at the Olkiluoto site was carried out in 2009-2015, the protection against APC was required). In general, the topic has been handled mainly in security related cooperation activities.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

The Nuclear Energy Decree Section 22 b gives radiological acceptance criteria for different levels of DiD:

- normal operation and anticipated operational occurance (expected to occur once or several times during any period of a hundred operating years); effective annual dose limit 0,1 mSv for the public

- Class 1 postulated accidents (DBC 3, assumed to occur less frequently than once over a span of one hundred operating years, but at least once over a span of one thousand operating years) public dose limit 1 mSv, Class 2 postulated accidents (DBC 4, assumed to occur less frequently than once during any one thousand operating years) public dose limit 5 mSv, design extension conditions (an accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function; or an accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment; or an accident caused by a rare external event and which the facility is required to withstand without severe fuel failure) public dose limit 20 mSv.

- Severe accidents; The release of radioactive materials arising from a severe accident shall not necessitate large scale protective measures for the population nor any long-term restrictions on the use of extensive areas of land and water. In order to restrict long-term effects the limit for the atmospheric release of cesium-137 is 100 TBq. The possibility of exceeding the set limit shall be extremely small. The possibility of a release in the early stages of the accident requiring measures to protect the public shall be extremely small.

Finnish regulatory guide YVL B.1 gives detailed requirements concerning deterministic safety analyses (in English https://www.stuklex.fi/en/ohje/YVLB-3) .

Finnish regulatory guide YVL C.3 (in English: https://www.stuklex.fi/en/ohje/YVLC-3) explains in more detail what is meant by “large scale protective measures”: Analyses must be provided to demonstrate that any release of radioactive substances in a severe accident shall not warrant the evacuation of the population beyond the protective zone (appr. 5 km) or the need for people beyond the emergency planning zone (appr. 20 km) to seek shelter indoors. Guide YVL A.7 (in English: https://www.stuklex.fi/en/ohje/YVLA-7) states that a nuclear power plant unit shall be designed in compliance with the Government Decree principles in a way that:

- the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq is less than 5E 7/year;

- the accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency.

Operating NPPs (Loviisa NPP and Olkiluoto units 1 and 2) in Finland fulfill the radiological criteria for normal operation, anticipated operational occurrences, postulated accidents and design extension conditions. There are no exemptions granted.

There is an exemption in the Nuclear Energy Decree that Section 22 b shall be applicable to a nuclear power plant unit for which an operating licence was issued prior to year 1990 (meaning Loviisa units 1 and 2 and Olkiluoto units 1 and 2) to the extent required with respect to the technical solutions of the facility in question, under the principle laid down in Section 7 a of the Nuclear Energy Act (continuous safety improvements principle, see the answer to question 1). Probabilistic safety goals related to severe accidents apply as such to new NPP units. For operating units, the SAHARA principle and the principle of continuous improvement are applied. The large release frequency has been decreasing over the years at the Finnish NPPs also after the severe accident management modifications (see question 7.6) mainly due to the decrease of the core damage frequency. Olkiluoto units 1 and 2 don't fulfil the early release criteria either (in about 30% of the reactor core damage frequency, accident sequences lead also to an early containment bypass sequence). There is not much opportunities at the Olkiluoto units 1 and 2 to improve the situation anymore at the plant, because the bypass sequences are mainly related to outages when the containment is open. However, the licensee is still assessing the possibilities to inert the containment earlier after the outage. At Loviisa units 1 and 2, in about 2% of the reactor core damage frequency, accident sequences lead also to an early containment bypass sequence. This fulfils the goal of a small contribution to the reactor core damage frequency but the licensee still needs to continue assessing possibilities to decrease the risk of early release according to SAHARA principle.

Olkiluoto unit 3 under commissioning fulfills all the radiological acceptance criteria.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

In Finland the safety objectives are written to new reactors. New reactor in Finland means an NPP without a construction license. There are no differences of the safety objectives associated with different levels of DiD for new or existing reactors but the applicability of the safety objectives related to severe accidents is assessed taking into account the principle of continuous improvement. The application of the severe accident criteria for operating NPPs is explained in the previous question.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

STUK Regulation on the Safety of a Nuclear Power Plant Section 14: The design of a nuclear power plant shall take account of external hazards that may endanger safety functions. Systems, structures, components and access shall be designed, located and protected so that the impacts of external hazards deemed possible on plant safety remain minor. The operability of systems, structures and components shall be demonstrated in their design basis external environmental conditions. External hazards shall include exceptional weather conditions, seismic events, the effects of accidents that take place in the environment of the facility, and other factors resulting from the environment or human activity. The design shall also consider unlawful actions with the aim of damaging the plant and a large commercial aircraft crash.

Design extension condition includes an accident caused by a rare external event and which the facility is required to withstand without severe fuel failure.

YVL B.7 “Provisions for internal and external hazards at a nuclear facility” gives more detailed requirements for external hazards (in English: https://www.stuklex.fi/en/ohje/YVLB-7)

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires “preventing accidents and, should an accident occur, mitigating its consequences and avoiding” “early radioactive releases that would require off-site emergency measures but with insufficient time to implement them” and “large radioactive releases that would require protective measures that could not be limited in area or time”. Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

Nuclear Energy Decree Section 22 b: “The release of radioactive materials arising from a severe accident shall not necessitate large scale protective measures for the population nor any long-term restrictions on the use of extensive areas of land and water. In order to restrict long-term effects the limit for the atmospheric release of cesium-137 is 100 TBq. The possibility of exceeding the set limit shall be extremely small. The possibility of a release in the early stages of the accident requiring measures to protect the public shall be extremely small.”

Finnish regulatory guide YVL C.3 (in English: https://www.stuklex.fi/en/ohje/YVLC-3) explains in more detail what is meant by “large scale protective measures”: Analyses must be provided to demonstrate that any release of radioactive substances in a severe accident shall not warrant the evacuation of the population beyond the protective zone (appr. 5 km) or the need for people beyond the emergency planning zone (appr. 20 km) to seek shelter indoors.

Guide YVL A.7 (in English: https://www.stuklex.fi/en/ohje/YVLA-7) states that a nuclear power plant unit shall be designed in compliance with the Government Decree principles in a way that:

- the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq is less than 5E 7/year;
- the accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires “timely implementation of reasonably practicable safety improvements to existing nuclear installations”. Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Finnish regulatory framework does not include any systematic methods for assessing what are considered reasonably practicable safety improvements. They are considered case by case mainly using "engineering judgement". Since the responsibility of safety relies with the licensees, it's the licensees' responsibility to justify whether some safety improvements are needed. Most common approach is that STUK regularly updates the regulatory requirements for new NPPs based on operating experiences, safety research and advances in science and technology taking into account also international safety standards. Separate implementation decisions are made for operating NPPs and NPPs under discussion based on the licensees' assessments (see previous answers). Also periodic safety assessments and use of probabilistic risk assessment (PRA) can bring new sights for safety improvement needs when looking the overall picture of the plant safety.

Licensees have limited resources for safety improvements, so focusing safety improvements for the most significant ones is important. PRA is a good tool to prioritise plant modification needs and to compare the safety significance of alternative solutions. Other aspects to be taken into account when assessing the justifications for safety improvements include radiation doses to workers (doses received during the plant modification or decreased doses after the modification) or to the public (normal operation or accident conditions). There can also be some risks related to the plant modification itself which needs to be considered. Systematic quantitative cost-benefit analysis is not used in Finland because of its uncertainties. Licensees can compare the costs of the plant modification to the gained safety improvement and for example propose alternative solutions based on the PRA results and overall safety of the plant. Level 3 PRA is not used in Finland as adequate information for regulatory purposes is considered to be received from level 2 PRA already. When assessing the lessons learnt from some operating experience or accidents and possible safety improvement needs at different NPPs (e.g. the Fukushima Dai-ichi accident), it is important to have an overall picture of the plant safety. Sometimes there might be even some more significant plant specific improvement needs that should be handled first instead of limiting the assessment only on for example in seismic hazards and flooding.

Timely implementation of safety improvements is also an important aspect. The justified safety improvements should be implemented as soon as reasonably practicable. On the other hand all plant modifications need careful planning and assessment of possible risks caused by planned modifications (configuration management).

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

The general rules for comprehensive and systematic periodic safety assessments at existing nuclear facilities are presented in the Finnish Nuclear Energy Act in order to identify reasonably practicable and achievable safety improvements which shall be implemented in a timely manner. Section 7 a also requires that “The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology.”

More detailed regulations require a periodic safety review (PSR) to be carried out at least every ten years which includes the licensee’s assessment of the current safety status of the nuclear facility, potential safety improvements, and the maintaining of safety in future (see question 10). Practice has been that the licensee is obliged to demonstrate that the safety of the operations can be ensured and improved also during the next 10 years. In general, PSR process includes licensee’s assessment, how the modern safety standards can be fulfilled as far as reasonably practicable. In Finland, this process is covered by the process of implementation decisions of revised regulatory guides written always for new nuclear facilities (see below). PSR is then an overall safety assessment of the site hazards, plant design, its current condition and licensee’s activities where the implementation decisions of the recently updated regulatory guides can be referenced, the planned safety improvement measures are listed and decisions concerning some further safety improvements can be made.

Finnish regulatory guides (YVL Guides) are applied as such for new nuclear facilities. For the existing facilities and facilities under construction, separate facility specific implementation decisions are made by Radiation and Nuclear Safety Authority (STUK). In case of non-compliances the licensee has to propose plans for improvements which can include plant modifications or modifications of the licensee’s activities to reach the required safety level. STUK can also approve exemptions from new requirements if it is not technically or economically reasonable to implement respective modifications and if safety is justified and considered adequate. This is case by case decision.

Finnish regulations require also that licensees maintain an up-to-date and comprehensive plant-specific probabilistic risk assessment (PRA) and that they use the PRA to enhance nuclear facility safety, to identify and prioritise plant modification needs and to compare the safety significance of alternative solutions.

New urgent information from accidents, operating experiences and research might also lead to direct improvements measures. For example, some of the plant safety modifications carried out at the Finnish NPPs are originating from the lessons learnt from the Fukushima Dai-ichi accident.

Also plant modernisations are considered as opportunities to improve safety. When carrying out plant modernization projects, the possibilities to further improve safety are always analysed at the Finnish NPPs. For example, when the emergency diesel generators will be replaced at the Olkiluoto units 1 and 2 within the next few years, the new emergency diesel generators will be provided with alternative air and seawater cooling, while the existing diesels have only seawater cooling. In the renewal of the reactor coolant pumps at the Olkiluoto units 1 and 2 there is also a related safety improvement since a flywheel will be added to the reactor coolant pump shaft to ensure sufficient cooling of the nuclear fuel in case of a trip during which the electrical power is unavailable. The pump is currently shut down by means of electric control.

A generic lesson learned in Finland is that the closer NPPs get to the end of their design lifetime, especially due to the current market price of electricity, the more difficult it is for the licensees to make decisions to modernise or modify the NPPs. Instead of renewing a system or a component, modernisation may be rejected or a partial modification is planned resulting in ageing issues in the remaining parts. This is why improving safety should be a continuous process from the start of plant operations and not related only for example for plant's long-term operation.

Peer reviews are regularly carried out both at the regulatory body STUK (IRRS, EPREV, IPPAS, …) and the licensees (OSART, WANO, …).

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Finnish regulations and safety requirements are regularly updated considering operating experience and safety research and advances in science and technology. The revised regulatory guides (YVL Guides) are applied as such for new nuclear facilities. For the existing facilities and facilities under construction, separate facility specific implementation decisions are made. The revision of the whole regulatory guide system was finalised in 2013. It took into account the updated international guidance such as IAEA safety standards and WENRA (Western European Regulators' Association) safety reference levels for existing reactors and WENRA safety objectives for new reactors. In addition, the lessons learnt from the Fukushima Dai-ichi accident were taken into account. STUK participates actively in the work of WENRA and IAEA, so the YVL Guides published in 2013 are very well in line also with IAEA SSR-2/1 rev.1 and updated WENRA reference levels. In 2013 regulations and guides, DiD level 3 was divided into two: 3a for design basis accidents and 3 b design extension conditions without fuel melt. Also the interpretation on what is meant by WENRA safety objective O3 for new plants (accidents with core melt) was included in the Finnish regulations taking into account RHWG Report on Safety of new NPP designs. There is a requirements in STUK Regulation Y/1/2018 Section 9 (in English https://www.finlex.fi/fi/viranomaiset/normi/555001/42574) that "the levels of defence required under the defence-in-depth principle shall be as independent of one another as is reasonably achievable". Guide YVL B.1 Section 4.3.1 gives more detailed requirements for the independence of DiD levels (in English https://www.stuklex.fi/en/ohje/YVLB-1).

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

Several plant changes have been carried out at the both Finnish operating NPPs during their lifetime. The most important projects since the commissioning have been modifications made for protection against fires (mainly 1990's), modifications based on the development of the PRA models, development of severe accident management strategies and implementation of required measures (mainly 1990's), modifications based on the lessons learnt from the Fukushima Dai-ichi accident, reactor power upratings (end of 1990's), and construction of training simulators, interim storages for spent fuel and repositories for reactor operational waste. The licensees' action plans after the Fukushima accident (these are the main safety improvements following the publication of WENRA safety objectives and IAEA SSR-2/1 and are in line with the updated Finnish regulations in 2013) included for example:

- enhanced protection against high seawater level at the Loviisa NPP
- independent air-cooled cooling units for decay heat removal from the reactor core and from the spent fuel pools in case of the loss of sea as an ultimate heat sink at the Loviisa NPP (these cooling units were considered already before the Fukushima Dai-ichi accident in the previous PSR due to the increased risks related to transporting of oil on the Finnish Gulf)
- ensuring cooling of the reactor core in case of total loss of AC power systems at the Olkiluoto units 1 and 2; A new steam turbine driven high pressure emergency injection system will be installed in 2018. The new system is planned to be as independent of the existing plant electric and automation systems as possible. Besides the high pressure emergency injection system, there is the possibility to inject water to the reactor after the depressurisation of the coolant system from the fire-protection system via emergency inlets as a manual operation.
- ensuring operation of the auxiliary feed water system pumps independently of availability of the sea water systems at the Olkiluoto units 1 and 2
- diverse cooling of the spent fuel pools at the Loviisa and Olkiluoto NPPs

As a result of the studies made after the TEPCO Fukushima Dai-ichi accident, no major changes at the plants were considered necessary for severe accident management since the backfitting measures were already carried out during 1980's and 1990's based on the lessons learnt from the TMI accident (see question 7.6).

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.)you require from this level of PSA:

PSA level 1: Yes. According to YVL YVL A.7 (in English https://www.stuklex.fi/en/ohje/YVLA-7) "the design of a nuclear power plant unit shall be such that the mean value of the frequency of reactor core damage is less than 10E–5/year". For the review of the nuclear power plant's construction licence application, the licence applicant shall submit to STUK

- for approval the Level 1 and Level 2 design phase probabilistic risk assessments, the computerised PRA model included.
- for information the PRA peer review report and a report on how the results of the review have been or are intended to be taken into account in the PRA.

For the review of the nuclear power plant's operating licence application, the licence applicant shall submit to STUK

- for approval the Level 1 and Level 2 probabilistic risk assessments, the computerised PRA model included.
- for information the PRA peer review report and a report on how the results of the review have been or are intended to be taken into account in the PRA.

The licensee shall keep the PRA updated and extend it, where necessary, during plant operation. And for example the licensee shall submit to STUK risk assessments related to plant modifications.

The licensee shall assess the risk relating to a nuclear power plant that is due for decommissioning. Provision shall be made for the nuclear power plant's decommissioning and the associated risks shall be identified already during the plant's design phase. The corresponding risk assessment shall be submitted to STUK during the construction licence application phase. The risk assessment of decommissioning shall be submitted to STUK for approval in good time before the plant's power operation is ended.

- PSA level 2: Yes. According to YVL A.7 the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq shall be less than 5·10 -7/year. See previous answer concerning the licensing stages, requirements are the same for levels 1 and 2.
- PSA level 3: No. Level 3 PRA is not used in Finland as adequate information for regulatory purposes is considered to be received from level 2 PRA already. Uncertainties related to level 3 PRA are also considered high.

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)? According to YVL A.7 "the licence applicant shall update the computerised PRA model and documentation as the plant's detailed design and construction proceeds and in case of significant modifications in particular. The updated Level 1 and Level 2 computerised PRA model, PRA documentation and justifications for the modifications shall be submitted to STUK for information as the construction proceeds, at least once a year."

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

According to YVL A.7: In the PRA, the following shall be analysed as initiating events: the plant's internal failures, disturbances and human errors, loss of off-site power supply, fires, flooding, hoisting of heavy loads, abnormal weather conditions, seismic events and other environmental factors as well as external factors caused by human activities. Requirements for the security of nuclear power plants and the use of the PRA in the risk analysis for nuclear security are set forth in Guide YVL A.11. PRA shall cover all operating states.

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

YVL A.7 is applied also to spent nuclear fuel storage in pools adjacent to the reactor and spent nuclear fuel storage in separate storages and to nuclear power plants at which power operation has ended but still contain spent nuclear fuel. The requirements for risk analyses concerning spent nuclear fuel encapsulation and final disposal are given in Guide YVL D.3 to the spent nuclear storages. For example, Olkiluoto unit 3 PRA includes nuclear fuel storage pools. Also there is a PRA for a separate spent fuel storage at the Olkiluoto site. And there is a requirement for the licensee of Loviisa NPP to provide a PRA for their spent fuel storage by the end of year 2018. This requirement was set when STUK made the implementation decision concerning the new YVL A.7 published in 2013. For spent fuel storage pools outside the containment the criterion is the same as for PRA level 2 for large releases.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

The dependencies between the plant units (for example electricity systems at Olkiluoto 1/2 or additional emergency feedwater system at Loviisa 1/2) has to be taken into account in the PRA models. However, the PRAs are plant unit specific and the criterions for levels 1 and 2 are plant unit specific. STUK follows the international discussion and research currently ongoing concerning multiple unit PRAs but so far we have not seen necessary to update the Finnish requirements for PRA.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

Yes. There are many requirements for risk-informed applications. According to YVL A.7

- The PRA shall be used in the risk-informed development of the Operational Limits and Conditions (OLC) to assess their coverage and balance. The PRA shall be used to determine the surveillance test intervals and allowed outage times of systems and components important to safety. The Operational Limits and Conditions and allowed outage times applied on structures, systems and components shall be separately analysed for every plant operational state. The PRA shall also be used to analyse failures where the change of the operational state may cause a greater risk than repairing the failure without changing the operational state.

- The PRA shall be used in the development of a risk-informed ageing management programme in accordance with Guide YVL A.8.
- The PRA shall be used in the risk-informed development of the in-service inspection programmes of Safety Class 1, 2 and 3 as well as Class EYT system piping in accordance with Guide YVL E.5.
- The PRA shall be applied to determine the safety classification of structures, systems and components. It shall be ensured by the PRA that the safety classification of every structure, system and component corresponds to its safety significance.
- The PRA shall be used in the risk-informed development of on-line preventive maintenance programmes carried out during power operation for systems and components important to safety.
- Personnel training shall address PRA-identified safety-significant actions relating to maintenance and operation as well as the most significant disturbances and accidents. The planning of simulator training for control room operators shall also ensure that the most important accident sequences and risk-significant operator actions are covered by the training.
- The most significant event sequences analysed in the PRA shall be used to support the development of the abnormal and emergency operating procedures.
- The frequencies of initiating events specified in the PRA shall be used as support in determining to which event category specified by the STUK regulation the various initiating events belong based on their probability.
- The PRA shall be used as support in deciding which severe accident event sequences are analysed in accordance with Guide YVL B.3 for radiation effects (releases and doses) caused by an accident and also in deciding

which accident sequences are used in emergency planning in accordance with Guide YVL C.5.

- In developing commissioning test programmes, the PRA shall be used to assess the coverage and balance of the programmes as well as to reduce the risks arising from the commissioning tests.

- The PRA shall be used to manage risks relating to maintenance outages, refuelling outages and the related operational states as well as the transfers between the operational states.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

According to YVL A.7 the PRA shall be constantly updated and specified taking into account operating experience feedback, plant modifications, new research results and the advancement of computation methods.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so: There is one research reactor in Finland, i.e., 250 kW Triga Mark II type open pool reactor which is in final shutdown stage and in licensing phase for decommissioning. There has not been any requirements to produce a PRA for this research reactor during its operation. For the decommissioning licence, there is a requirement to do a risk assessment which is not so detailed as for NPPs. For any possible new research reactors, a PRA would be required according to YVL A.7 (however, there are currently no plans to build new research reactors in Finland) .

- What levels of PSA are performed?

- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?

- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?

- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

DEC without fuel melt is considered in level 3b of DiD. Design basis accidents are considered in level 3a. Guide YVL B.1 provides more detailed requirements (in English https://www.stuklex.fi/en/ohje/YVLB-1). At level 3b, the objective is to control design extension conditions, meaning:

- anticipated operational occurrences and Class 1 postulated accidents that involve a common cause failure in the system designed for coping with the event concerned; (=DEC A)
- combinations of failures selected on the basis of a probabilistic risk assessment; and (=DEC B)
- rare external events that are unlikely to occur but nevertheless considered possible, such as extreme weather phenomena or the impact of a large aircraft. (=DEC C)

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Guide YVL B.3 gives more detailed requirements for deterministic safety analyses (in English https://www.stuklex.fi/en/ohje/YVLB-3). "Analyses pertaining to the plant's behaviour as well as releases of radioactive substances and radiation doses shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents. Examples of the events to be analysed are given in [IAEA GSR Part 4 and IAEA SSG-2]. According to YVL A.7 the frequencies of initiating events specified in the PRA shall be used as support in determining to which event category specified by the STUK regulation the various initiating events belong based on their probability. According to YVL B.3 "in connection with periodic safety assessments, the licensee shall evaluate the scope of and need for updates in transient and accident analyses, and update the analyses for the final safety analysis report, where necessary".

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

A loss of cooling in the fuel storage resulting in severe damage to the spent nuclear fuel is one of the events to be practically eliminated according to YVL B.1. "Events to be practically eliminated shall be identified and analysed using methods based on deterministic analyses complemented by probabilistic risk assessments and expert assessments. Practical elimination cannot be based solely on compliance with a cut-off probabilistic value. Even if the probabilistic analysis suggests that the probability of an event is extremely low, all practicable measures shall be taken to reduce the risk." See also answer to 6.4 for fuel storage facilities and rare external events .

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt? According to YVL B.3: For DEC A accidents (see DEC categories in answer to question 6.1), the most penalising single failure shall be assumed in one of the systems whose operation is required to accomplish a safety function in the event in question. For DEC B and C accidents, a single failure need not be assumed. The consequences of an initiating event shall be assumed in the analyses. Loss of the external grid need not be combined with other initiating events in design extension condition analyses unless it is the likely consequence of an initiating event. In design extension condition analyses, best estimate methods can be applied concerning assumptions of the plant's initial state and the performance of operating subsystems. In design extension condition analyses, application of the best estimate method need not be complemented with an uncertainty analysis.

- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?

According to YVL B.1, in multiple failure events (DEC B) and in rare external events (DEC C), it shall be possible to accomplish decay heat removal from the reactor to outside the containment and control of reactivity in such a way that the limits set forth for fuel integrity, radiological consequences and overpressure protection in design basis category DEC are not exceeded.

It shall be possible to accomplish decay heat removal and reactivity control in rare external events (DEC C) without relying on power supply from transportable sources for at least eight hours without any material replenishments or recharging of the DC batteries. In addition, a sufficient inventory of water and fuel and capability to recharge the DC batteries shall exist at the plant site to enable decay heat removal for a period of 72 hours.

The nuclear power plant shall have in place arrangements that can guarantee sufficient cooling for the fuel placed in fuel storage facilities during rare external events in accordance with requirement 450. These arrangements shall make it possible to supervise the water level in the spent fuel pools for a minimum of eight hours without recharging the DC batteries. Furthermore, it shall be possible to keep the fuel reliably submerged during the loss of the plant's internal electricity distribution system in accordance with requirement 451. A sufficient inventory of water and fuel and capability to recharge the DC batteries shall exist at the plant site to maintain these arrangements for a period of 72 hours.

- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

According to STUK regulation, the levels of defence required under the defence-in-depth concept shall be as independent of one another as is reasonably achievable. The loss of any single level of defence may not impair the operation of the other levels of defence. According to YVL B.1, such independence shall be based on the adequate application of functional isolation, the diversity principle and physical separation between the levels of defence. Due consideration shall be given to the dependence on the auxiliary systems supporting safety functions at different levels of defence in depth. Any dependence shall not unnecessarily impair the reliability of the defence-in-depth concept. The systems, structures and components required for each postulated initiating event shall be identified, and it shall be shown by means of deterministic analyses that the systems, structures and components required for implementing any one level of defence in depth are sufficiently independent from the other levels. The adequacy of the achieved independence shall also be judged by probabilistic analyses.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

There are three safety classes in Finland. Guide YVL B.2 (in English https://www.stuklex.fi/en/ohje/YVLB-2) gives the more detailed requirements. Systems shall be grouped into Safety Classes 2 and 3 as well as Class EYT (non-nuclear safety) based on their significance for the reliability of safety functions from the viewpoint of the management of initiating events. Systems accomplishing safety functions shall be assigned to Safety Class 2 if they are designed to provide against postulated accidents to bring the facility to a controlled state and to maintain this state for as long as the prerequisites for transfer to a safe state can be ensured. Safety Class 3 shall include systems accomplishing safety functions that … accomplish the diversity principle and are designed to ensure the bringing of the facility into a controlled state in case of the failure of systems primarily taking care of a corresponding safety function (=DEC A). In addition to decay heat removal systems (meeting N+2 failure criterion), the nuclear power plant shall have a system that complies with the diversity principle and is capable of removing the decay heat from the reactor and containment following an initiating event of any anticipated operational occurrence or Class 1 postulated accident in such a way that the limits set forth for fuel integrity, radiological consequences and overpressure protection in design basis category DEC are not exceeded. The decay heat removal system that complies with the diversity principle shall satisfy the (N+1) failure criterion and the 72-hour self-sufficiency criterion.

Systems belonging to Class EYT (non-nuclear safety) shall be allocated to Class EYT/STUK (system modifications sent for information to STUK) if the system… is necessary for bringing the facility to a controlled state in case of an event involving a design basis category DEC combination of failures (DEC B) or a rare external event (DEC C). According to YVL B.1, it shall be possible to accomplish decay heat removal and reactivity control in rare external events (DEC C) without relying on power supply from transportable sources for at least eight hours without any material replenishments or recharging of the DC batteries. In addition, a sufficient inventory of water and fuel and capability to recharge the DC batteries shall exist at the plant site to enable decay heat removal for a period of 72 hours. For DEC B/C no requirements for redundancy, environmental qualification for the conditions that they are used.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

DECs without fuel melt were introduced in Finnish regulations in 2008 (DiD level 3) based on WENRA reference levels after which they have been introduced both for in existing and new NPPs (requirements updated in 2013, see previous answers, DiD level 3 divided into 3a and 3b). So in principle the requirements are the same but there might be slight differences in implementation at the existing facilities related for example the safety classification or separation principle of systems. Mainly the existing NPPs fulfil the YVL Guide requirements concerning DEC but some detailed exemptions have been approved.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

In Finland DEC is considered in DiD level 3b and only without fuel melt. Severe accidents are dealt in DiD level 4.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

According to YVL B.3, analyses pertaining to the plant's behaviour as well as releases of radioactive substances and radiation doses shall cover the nuclear power plant's normal operational states, anticipated operational occurrences, postulated accidents, design extension conditions and severe reactor accidents. Examples of the events to be analysed are given in [IAEA GSR Part 4 and IAEA SSG-2]. In analysing severe reactor accidents, best estimate methods can be applied concerning assumptions of the plant's initial state and the performance of operating subsystems. However, the more essential the function, the better assurance for its successful accomplishment shall be provided. In severe accident analyses, application of the best estimate method need not be complemented with an uncertainty analysis. In severe reactor accident analyses, the most penalising failure according to the failure criterion presented in chapter 4.3 of Guide YVL B.1 shall be assumed for systems designed for severe reactor accident management. Consequences of the initiating event shall also be taken into account. The time needed for actions required for the severe reactor accident management strategy and other factors relating to the implementation of the actions (e.g. accessibility of locally operated equipment) shall be justified. Analyses justifying the hydrogen management strategy shall separately evaluate cases in which the hydrogen generation rate increases. Severe reactor accident analyses shall cover all actions required for the plant's severe reactor accident strategy and the phenomena associated with the strategy.

According to YVL A.7, the frequencies of initiating events specified in the PRA shall be used as support in determining to which event category specified by the STUK Regulation the various initiating events belong based on their probability. The PRA shall be used as support in deciding which severe accident event sequences are analysed in accordance with Guide

YVL B.3 for radiation effects (releases and doses) caused by an accident and also in deciding which accident sequences are used in emergency planning in accordance with Guide YVL C.5.

According to YVL B.3 “in connection with periodic safety assessments, the licensee shall evaluate the scope of and need for updates in transient and accident analyses, and update the analyses for the final safety analysis report, where necessary”.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

No. A loss of cooling in the fuel storage resulting in severe damage to the spent nuclear fuel is one of the events to be practically eliminated according to YVL B.1. “Events to be practically eliminated shall be identified and analysed using methods based on deterministic analyses complemented by probabilistic risk assessments and expert assessments. Practical elimination cannot be based solely on compliance with a cut-off probabilistic value. Even if the probabilistic analysis suggests that the probability of an event is extremely low, all practicable measures shall be taken to reduce the risk.” See also answer to 6.4 for fuel storage facilities and rare external events.

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt? See the answers for 7.2
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt? Yes. There shall be dedicated and independent severe accident management (SAM) systems (YVL B.6, in English https://www.stuklex.fi/en/ohje/YVLB-6) .
- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states? Yes, for reaching and maintaining the controlled state. According to STUK Regulation Y/1/2018 Section 9 the levels of defence required under the defence-in-depth principle shall be as independent of one another as is reasonably achievable. And according to Section 11 the systems

needed for reaching and maintaining a controlled state and the monitoring of the progress of an accident and the plant's status in severe reactor accidents in a nuclear power plant shall be independent of the systems designed for normal operation, anticipated operational occurrences and postulated accidents. The leaktightness of the containment during a severe reactor accident shall be reliably ensured. The nuclear power plant shall be designed so that it can be reliably brought into a safe state after a severe reactor accident.

Controlled state following a severe reactor accident refers to a state where the removal of decay heat from the reactor core debris and the containment has been secured, the temperature of the reactor core debris is stable or decreasing, the reactor core debris is in a form that poses no risk of recriticality, and no significant volumes of fission products are any longer being released from the reactor core debris. Safe state following a severe reactor accident refers to a state where the conditions for the controlled state of a severe reactor accident are met and, in addition, the pressure inside the containment is low enough that leak from the containment is minor, even if the containment is not leak-tight.

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

According to YVL B.2, Safety Class 3 shall include systems accomplishing safety functions that … are designed for severe reactor accident management. We are currently clarifying the YVL B.2 requirements so that the systems needed for reaching and maintaining a controlled state and the monitoring of the progress of an accident and the plant's status in severe reactor accidents need to be in Safety Class 3. Systems needed to reach a safe state after a severe reactor accident are allowed to be non-safety-classified.

According to YVL B.1, the following systems performing functions relevant to safety shall satisfy the (N+1) failure criterion: … active components of the systems designed to control severe reactor accidents. The power supply (electricity, compressed air, etc.) to the systems designed for managing severe reactor accidents shall be independent of all the other power supply units and power distribution systems of the plant. According to YVL B.6, combustible gases shall be primarily managed by systems and components that are located inside the containment and do not require an external power supply.

According to YVL B.1, the systems, structures and components important to safety shall be qualified for their intended use. The qualification process shall demonstrate that the systems, structures and systems are suitable for intended use and satisfy the relevant safety requirements. Aside from the assurance of the correctness of the design bases and the sufficiency of the quality management of design and implementation, the qualification process shall also include environmental qualification. This means that SAM systems structures and components need to be qualified for severe accident environmental conditions.

Hazards included in the design basis of SAM systems are design basis hazards (in Finland that means that for example design basis earthquake frequency of occurrence of stronger ground motions is less than once in a hundred thousand years (1E–5/y) at a median confidence level). According to YVL B.7, exceptional external events and conditions with an estimated frequency of occurrence less than 1E-5/year shall be considered design extension conditions (DEC C events). In DEC, the objective is to avoid fuel melt, so SAM systems are not designed using DEC C hazard assumptions.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

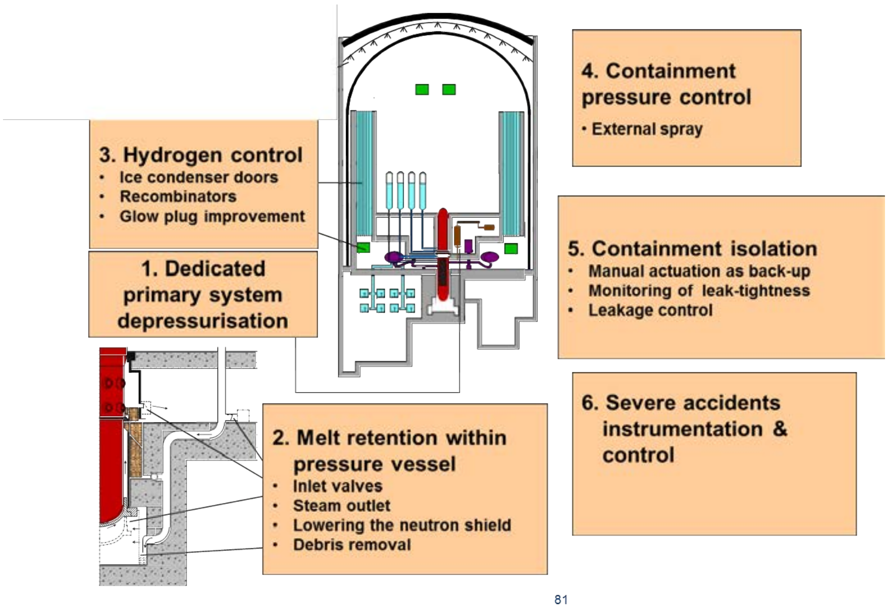
Requirements related to severe accident management (SAM) and the dedicated SAM systems were introduced in Finland for new nuclear power plants in 1982 after the Three Mile Island (TMI) accident. Separate regulatory decisions were made for existing NPPs after the Chernobyl accident. Utilities started planning the implementation of the measures in 1980's and first SAM systems were installed at the plants in 1989.

A comprehensive severe accident management strategy has been developed and implemented at the operating Finnish NPPs during 1980's and 1990's after the accidents in TMI and Chernobyl. These strategies are based on ensuring the containment integrity which is required in the national regulations. Level 2 PRA was also used for developing the strategies and led to some additional modifications at the plants. The means for managing severe accidents had to be adjusted to the existing design, and so an optimal implementation of all chosen solutions was not possible.

The Loviisa severe accident management programme, which includes plant modifications and severe accident management procedures, was initiated in the end on 1980's in order to meet the requirements of STUK. Loviisa NPP's SAM approach focuses on ensuring the following top level safety functions:

- depressurisation of the primary circuit,
- absence of energetic events, i.e. hydrogen burns and steam explosions,
- coolability and retention of molten core in the reactor vessel,
- long term containment cooling ensuring subcriticality, and
- ensuring containment isolation.

The developed SAM strategy lead to a number of hardware changes at the plant (see Fig.) as well as to new SAM guidelines and procedures. The dedicated primary system depressurisation valves were installed at the same time with the renewal of the pressuriser safety valves in 1996. A new hydrogen management strategy for Loviisa was also formulated and plant modifications included installation of autocatalytic hydrogen recombiners, modifications in the igniters system and a dedicated system for opening the ice-condenser doors to ensure air circulation in the containment. The modifications were completed in 2003.



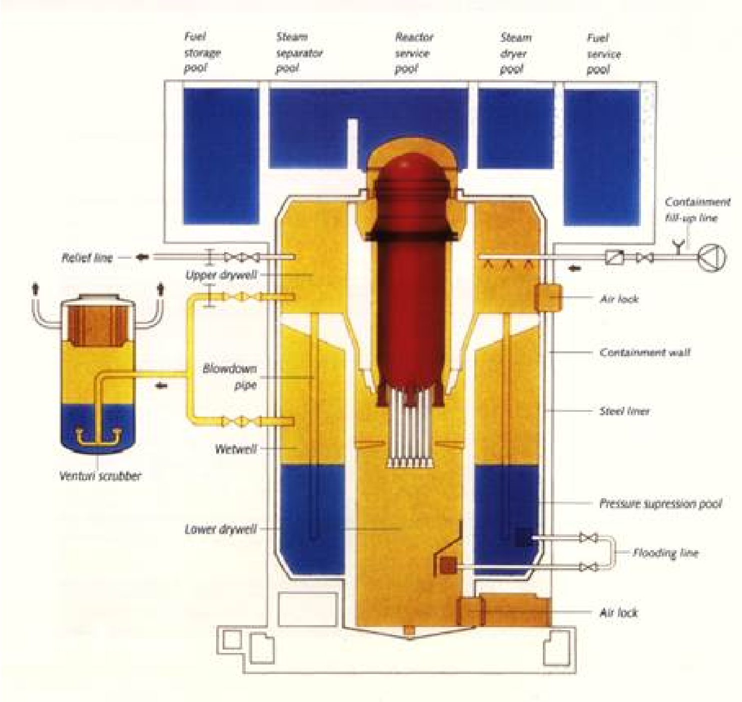
The cornerstone of the SAM strategy for Loviisa is the coolability of corium inside the reactor pressure vessel (RPV) through external cooling of the vessel. Due to in-vessel retention of molten corium all the ex-vessel corium phenomena such as ex-vessel steam explosions, direct containment heating and core-concrete interactions can be excluded. Some of the plant's design features make the in-vessel retention concept possible. Those features include the low power density of the core, large water volumes both in the primary and in the secondary side, no penetrations in the lower head of the RPV, and ice condensers which ensure a passively flooded cavity in most severe accident scenarios. An extensive research programme regarding the thermal aspects was carried out by the licensee. The modifications were completed in 2002. The most laborious one of them was the modification of the lower neutron and thermal shield such that it can be lowered down in case of an accident to allow free passage of water in contact with the RPV bottom.

The studies on prevention of long-term overpressurisation of the containment showed that the concept of filtered venting was not feasible at the Loviisa NPP because the capability of the steel liner containment to resist subatmospheric pressures is poor. Instead, an external spray system was designed to remove the heat from the containment during a severe accident when other means of decay heat removal from the containment are not operable. Autonomous operation of the system independently from plant emergency diesels is ensured with dedicated local diesel generators. The active parts of the system are independent from all other containment decay heat removal systems. The containment external spray system was implemented in 1990 and 1991.

The SAM strategy implementation included also a new, dedicated, limited scope instrumentation and control system for the SAM systems, a dedicated AC-power system and a separate SAM control room which is common to both units and to be used in case the main control room has to be abandoned during a severe accident. These were implemented mainly during 2000-2002.

The main provisions for severe accident management were installed at the Olkiluoto units 1 and 2 during the SAM project which was completed in 1989.The measures implemented were (see Fig.):

- containment overpressure protection (used in case of failed containment pressure suppression function before the core damage),
- containment filtered venting,
- lower drywell flooding from wetwell,
- containment penetration shielding in lower drywell,
- containment water filling from external source,
- containment instrumentation for severe accident control, and
- Emergency Operating Procedures for severe accidents.



One of the most significant deficiencies at the Olkiluoto plant containments, from the standpoint of controlling severe accidents, has been the small size of the containment, which may cause the containment to pressurise due to the hydrogen and steam generation during an accident (common feature for BWRs). Another deficiency is the location of the reactor pressure vessel inside the containment, which is such that the core melt erupting from the pressure vessel may expose the structures and penetrations which ensure the tightness of the containment, to pressure loads and thermal stresses. To eliminate these deficiencies, the containment was e.g. provided with a filtered venting system. To improve the possibilities for retaining organic iodine in the filtered venting system, chemicals have been added to the water in the scrubber tank of the system. To minimise the formation of organic iodine, it is possible to control the pH of the containment water volume by a specific system.

The part of the containment underneath the reactor pressure vessel can be flooded with water in order to protect the containment bottom and penetrations from the thermal effect of core melt. Some penetrations of the containment have been protected from the direct effect of core melt also by structural means. To ensure the cooling of reactor debris, the plant units are also provided with a water filling system, by the means of which the water level inside the containment can be raised all the way to the same level with the upper edge of the reactor core. A lot of research has been done on the possibility of steam explosions. The results show that the core melt discharged through the pressure vessel cools down as it travels through the water pool and cannot create a steam explosion. However, the structures of the lower equipment hatch have been enforced to decrease the risk for loss of containment integrity due to loads caused by limited steam explosions.

So in practice the severe accidents are analysed similarly in new and operating NPPs. The provisions are of course easier to implement when taken into account in the original design as in Olkiluoto unit 3 under commission compared to operating NPPs where the backfitting measures have been implemented after the commissioning. One of the biggest differences in the design of Olkiluoto unit 3 is the core catcher. Otherwise the Olkiluoto unit 3 has similar SAM systems as in operating plant units designed to fulfill its own severe accident management strategy (dedicated depressurisation valves of the primary circuit, dedicated containment spray system, filtered containment venting system and passive autocatalytic recombiners) .

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

The current Finnish requirements related to ensuring the containment building integrity state (STUK Regulation Y/1/2018):

- the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident conditions;
- pressure, radiation and temperature loads, radiation levels on plant premises, combustible gases, impacts of missiles and short-term high energy phenomena resulting from an accident shall be considered in the design of the containment; and
- the possibility of containment leaktightness becoming endangered as a result of reactor pressure vessel fracturing shall be extremely low.

A nuclear power plant shall be equipped with systems to ensure the stabilisation and cooling of molten core material generated during a severe accident. Direct interaction of molten core material with the load bearing containment structure shall be reliably prevented. More detailed requirements are given in YVL B.6.

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

According to YVL B.6, containment design pressure and temperature as well as the corresponding allowed leakage in postulated accidents shall be determined. The containment is considered to be leaktight, when the leakage is less than the allowed leakage. The containment design pressure and temperature in postulated accidents are determined by the containment analyses performed in compliance with Guide YVL B.3 by selecting the postulated accident exerting the highest load on the containment as the limiting case. A 10% safety margin shall be added to the maximum pressure (gauge pressure) obtained from the analyses to compensate for the uncertainties associated with the calculation methods and the calculation case. The containment shall be dimensioned so as to ensure that it retains its leaktightness in a severe reactor accident even if 100% of the easily oxidising reactor core materials react with water. Containment pressure and temperature limits shall be determined within which the containment retains its leaktightness in severe reactor accidents. The leaktightness of the containment in severe reactor accidents shall be demonstrated using the containment temperature and pressure obtained from the severe accident analyses performed in compliance with Guide YVL B.3 by increasing the maximum pressure (gauge pressure) by a 50% safety margin and by pressure increase due to hydrogen burn calculated according to the AICC principle. A containment based on the pressure suppression concept shall be designed to ensure that an accident involving the loss of the pressure suppression function will not lead to the loss of containment structural integrity. Any accident situation involving the loss of the pressure suppression function shall be analysed as a design extension condition (DEC B). The assumptions to be used in the analyses of design extension conditions are presented in Guide YVL B.3.

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

According to YVL B.6, the design shall allow for the testing of the leaktightness of the containment, its penetrations, access locks and hatches. Containment penetrations and access locks and hatches shall be designed to withstand the same temperature and pressure loads as the containment itself. Location, structure, protection and sealing materials of containment penetrations, access locks and hatches, and isolation valves shall ensure their operability and leaktightness during normal operation, anticipated operational occurrences and accidents. Containment penetrations shall withstand the loads exerted by piping movements and accidents. Equipment hatches shall be provided with double seals capable of being leak-tested. The equipment hatch of the containment shall be kept closed. The equipment hatch may only be opened in circumstances where it can be closed quickly enough to prevent releases resulting from potential transients or accidents under such circumstances. A containment pressure test shall be performed prior to the commissioning of the plant to demonstrate the structural integrity of the containment. The overpressure used in the pressure test shall be at least 1.15 times the containment design overpressure. The requirements for the containment pressure and leak test plans are set out in Guide YVL E.6. Regular leak tests shall be performed on the containment as well as its penetrations, access locks and hatches to ensure that the leaktightness of the containment remains at an acceptable level throughout the service life of the plant. The leak test shall be performed at a pressure equivalent to the maximum pressure in the postulated accident exerting the highest load on the containment. The leak test shall be performed at intervals that enable reliable monitoring of containment leaktightness. The containment shall be so designed that the test pressure of the periodic leak test will not endanger the operability of the containment and the structures and components within the containment, or significantly shorten their service life.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

According to YVL B.1, events that may result in a release requiring measures to protect the population in the early stages of the accident shall be practically eliminated. Events to be practically eliminated shall be identified and analysed using methods based on deterministic analyses complemented by probabilistic risk assessments and expert assessments. Practical elimination cannot be based solely on compliance with a cut-off probabilistic value. Even if the probabilistic analysis suggests that the probability of an event is extremely low, all practicable measures shall be taken to reduce the risk. As an example events to be practically eliminated include:

1.a rapid, uncontrolled increase of reactivity leading to a criticality accident or severe reactor accident;

2.a loss of coolant during an outage leading to reactor core uncovery;

3.a load jeopardising the integrity of the containment during a severe reactor accident (e.g. reactor pressure vessel breach at high pressure, hydrogen explosion, steam explosion, direct impact of molten reactor core on containment basemat or wall, uncontrolled containment pressure increase); and

4.a loss of cooling in the fuel storage resulting in severe damage to the spent nuclear fuel.

Question 9.2: Do you have requirements related to practical elimination?

See previous answer.

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

The probabilistic safety goals discussed previously apply as such to new NPP units. For operating units, the SAHARA principle and the principle of continuous improvement are applied. The large release frequency has been decreasing over the years at the Finnish NPPs also after the SAM modifications mainly due to the decrease of the core damage frequency. Olkiluoto units 1 and 2 don't fulfil the early release criteria either (in about 30% of the reactor core damage frequency, accident sequences lead also to an early containment bypass sequence). There is not much opportunities at the Olkiluoto units 1 and 2 to improve the situation anymore at the plant, because the bypass sequences are mainly related to outages when the containment is open. However, the licensee is still assessing the possibilities to inert the containment earlier after the outage. At Loviisa units 1 and 2, in about 2% of the reactor core damage frequency, accident sequences lead also to an early containment bypass sequence. This fulfils the goal of a small contribution to the reactor core damage frequency but the licensee still needs to continue assessing possibilities to decrease the risk of early release according to SAHARA principle.

The only research reactor in Finland is in final shutdown state and licensing for decommissioning is ongoing. For any possible new research reactors, the same requirements for practical elimination would apply as for new NPPs.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

See answer for question 7.6

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

See answer for question 9.1

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Yes, see answer for question 9.1

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Finland has applied periodic safety reviews (PSR) for the operating NPPs. According to Nuclear Energy Act, the safety of the nuclear facility shall be assessed as a whole ate least every ten years. Detailed requirements for PSR are given in YVL A.1 and they are in line with IAEA SSG-25 and WENRA reference levels. The licensee is obliged to demonstrate that the safety of the operations can be ensured and improved also during the next 10 years.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

In general, PSR process includes licensee's assessment, how the modern safety standards can be fulfilled as far as reasonably practicable. In Finland, this process is covered by the process of implementation decisions of revised regulatory guides written always for new nuclear facilities (see previous answers). PSR is then an overall safety assessment of the site hazards, plant design, its current condition and licensee's activities where the implementation decisions of the recently updated regulatory guides can be referenced, the planned safety improvement measures are listed and decisions concerning some further safety improvements can be made.

According to YVL A.1, for the purpose of conducting a periodic safety review, STUK shall be provided with safety-related reports including the licensee's assessment of the current safety status of the nuclear facility concerned, potential improvements, and the maintaining of safety in future. Requirements pertaining to these documents are specified in Section A.4 of YVL A.1, including (not a comprehensive list):

- licensee shall provide a summary of the most significant changes to the licensing documents since the previous operating licence was granted and a description of the up-to-dateness of the documents

- description demonstrating compliance with the requirements of Government Decrees and YVL Guides. As regards compliance with the requirements specified in the YVL Guides, any implementation decisions for the guides issued in respect of the nuclear facility concerned shall be reviewed and the implementation status of the measures defined in connection with the decisions specified. As regards any deviations/exemptions from the fulfilment of the requirements set out in the guides observed in the implementation decision stage, any changes made or foreseen in the design or operational organisation of the facility to satisfy the requirement concerned shall be specified.

- Description of the reassessment of the design bases of the facility site. The licensee shall assess the site-specific design bases concerning external threats and the potential need for updating them in connection with the periodic safety review. The description shall take due account of the advancement of the methods used to determine external threats. If the design bases need to be updated, this shall be taken into account in the update of the safety analyses. Guide YVL B.7 addresses external threats and provides detailed requirements for making provision against them.

- Description of the facility's ageing and ageing management. In the description, the licensee shall provide a summary of the ageing management programme concerning the operating licence period applied or remaining for the facility. The description may draw upon the annually submitted ageing management follow-up report by extending the description of ageing management to also cover the next safety review or renewal of the operating licence.

- Description of the environmental qualification of equipment. In the description, the licensee shall provide a summary of the equipment qualification procedures concerning the operating licence period applied or remaining for the facility, specifying how the qualifications are maintained and what the current status of the qualification is.

- Summary of renewed safety analyses. The transient and accident analyses, strength analyses, failure mode and effect analyses, probabilistic risk assessments as well as any other essential analyses concerning the facility shall be reviewed in connection with a periodic safety assessment. The analyses shall be updated where necessary and submitted to STUK. The summary shall provide a description of the up-to-dateness of the analyses, the conclusions drawn from the results of the renewed analysis, and the steps taken based on them, if any.

- Summary of the plant's safety indicators. The description shall provide a summary of the safety indicators monitored at the nuclear power plant and their development trends since the operating licence was granted or the previous periodic safety review was carried out.

- Description of the licensee's safety culture and safety management. The description shall specify the assessment methods, the conclusions drawn in respect of the current state and their implications for the subsequent or remaining operating licence period, and the steps taken to improve safety culture. The assessment and improvement of safety culture shall draw upon the expertise in organisational studies and nuclear safety practices.

- Summary of plant procedures. The summary shall specify the structure of the plant procedures, describing their up-to-dateness and any development projects currently underway or foreseen.

- Summary of the plant's radiation protection arrangements. The description shall provide a summary of the radiation protection of plant workers, the monitoring of radioactive releases, and the results of the environmental radiation monitoring programme. The description shall also provide a summary of the procedures by which the occupational radiation exposure of plant workers and radioactive releases are kept as low as reasonably achievable. Furthermore, an assessment shall be provided as to how the limitation of radioactive releases to and radiation levels in the environment is implemented employing the best available techniques.

- Summary of the plant's operating experience feedback and research activities and plant improvements. The description shall provide a summary of the plant's internal and external operating experience feedback activities and the uses made of research results to improve safety. The description shall also provide a summary of the plant improvements implemented since the previous operating licence was granted.

- Summary of the periodic safety review and action plan for improving plant safety. The description shall provide a summary of the periodic safety assessment results; an overall assessment of the safe operation of the plant following the previous periodic safety review; an assessment of the current state of the plant; and the preconditions for continuing its safe operation until the next periodic safety review. An action plan for plant improvements carried out pursuant to Section 7 a of the Nuclear Energy Act, complete with timetables, shall be provided as a summary of the periodic safety review.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

Finnish regulations and safety requirements are regularly updated considering operating experience and safety research and advances in science and technology. The revised regulatory guides (YVL Guides) are applied as such for new nuclear facilities. For the existing facilities and facilities under construction, separate facility specific implementation decisions are made. Before an implementation decision is made by radiation and nuclear safety authority (STUK), the licensees are requested to evaluate the compliance with the new guide. In case of non-compliances the licensee has to propose plans for improvement and schedules for achieving compliance. After having heard those concerned, STUK makes a separate decision on how a new or revised YVL Guide applies to operating nuclear facilities, or to those under construction. STUK can approve exemptions from new requirements if it is not technically or economically reasonable to implement respective modifications and if safety is justified and considered adequate. This is case by case decision.

In general, PSR process includes licensee's assessment, how the modern safety standards can be fulfilled as far as reasonably practicable. In Finland, this process is covered by the process of implementation decisions of revised regulatory guides written always for new nuclear facilities. PSR is then an overall safety assessment of the site hazards, plant design, its current condition and licensee's activities where the implementation decisions of the recently updated regulatory guides can be referenced, the planned safety improvement measures are listed and decisions concerning some further safety improvements can be made.

The last PSR of the Loviisa NPP was carried out in 2015-2017, and the Olkiluoto units 1 and 2 PSR in 2016-2018. Key issues in the last Loviisa NPP PSR and Olkiluoto NPP PSR have been ageing management, organisational issues and deterministic and probabilistic safety analyses and the status of the planned or ongoing safety improvements. The implementation of the revised regulatory guides was carried out during 2015 as a separate project but the results were utilised also in the PSRs. Loviisa NPP action plan concerning the safety related issues for the next period was approved by STUK in 2017 as a part of the PSR including:

- I&C renewal project ELSA (2016 -2018) and ageing management of I&C components
- Updating some deterministic analyses (DBA/DEC/SAM) 2016-2018
- Updating some PRA analyses (PRA model for both units and spent fuel storage)
- Increase of the embrittlement margins of Loviisa unit 2 RPV; action plan was submitted to STUK 12/2016, updating of the probabilistic (2018) and deterministic (2023) analyses
- Development of safety classification; new area seismic classification, seismic walkdowns
- Development of FSAR
- Development of the management system and human performance tools
- Finalising the on-going flooding protection improvements
- Decreasing the risk related to heavy load drop accident.

The STUK's safety assessment concerning the latest Olkiluoto NPP units 1 and 2 included organisational issues, performing the primary system pressure test, ageing management of the I&C systems and updates of some deterministic safety analysis. In the previous Olkiluoto NPP PSR carried out in 2007-2009, one of the safety improvements discussed between the licensee and STUK was related to emergency control rooms. Pursuant to a STUK Regulation Y/1/2018 (previously Government Decree 733/2008), a nuclear power plant shall have a supplementary control room independent of the main control room, and the necessary local control systems for shutting down and cooling the nuclear reactor, and for removing residual heat from the nuclear reactor and spent fuel stored at the plant in a situation where operations in the main control room are not possible. There is an exemption for this requirement for existing NPPs but according to the Nuclear Energy Act continuous safety improvement rule (Section 7 a), the licensee was required to assess and propose plant modifications to fulfil the safety goal as far as reasonably practicable. The licensee has now constructed separate emergency control rooms for the Olkiluoto units 1 and 2. The emergency control rooms have been redesigned and relocated to provide better coordination and control for plant shutdown and safety function monitoring. Plant units can now be brought to stable state solely by the controls from the emergency control room. Cooling the reactor down to a cold state can be carried out after the shutdown by using emergency control room and some local control posts.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

STUK Regulation Y/1/2018 Section 25 states that when designing, constructing, operating and decommissioning a nuclear facility, a good safety culture shall be maintained. Safety shall take priority in all operations. The decisions and activities of the management of each organisation participating in the abovementioned activities shall reflect its commitment to operational practices and solutions that promote safety. Personnel shall be encouraged to perform responsible work, and to identify, report, and eliminate factors endangering safety. Personnel shall be given the opportunity to contribute to the continuous improvement of safety. More detailed requirements are given in YVL A.3. This Guide is in line with IAEA GS-R-3 and is now updated to be in line with IAEA GSR Part 2.

There are no implicit requirements concerning the safety culture of the regulator. Safety is emphasized in STUK's management system. To meet the suggestion given in the IRRS mission carried out in 2012, STUK has continued the development activities related to regulator's safety culture.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

STUK regulation Y/1/2018 Section 21 (taking operating experience and safety research into consideration in order to improve safety) states that safety-significant operational events shall be investigated for the purpose of identifying the root causes as well as defining and implementing the corrective measures. For further safety enhancement, operating experience from the facility and from other nuclear facilities, the results of safety research and technical developments shall be regularly monitored and assessed. Opportunities for improvements in technical and organisational safety, identified from operating experience, safety research and technical developments shall be assessed and implemented to the extent regarded as justified on the basis of the principles laid down in Section 7 a of the Nuclear Energy Act (continuous safety improvement principle).

Detailed requirements concerning operating experience feedback of a nuclear facility are given in YVL A.10 (in English https://www.stuklex.fi/en/ohje/YVLA-10).

Arrangements in place at the Finnish NPPs are in line with the requirements and details were discussed when making the implementation decision concerning YVL A.10.

Appendix 5: Answers to the questionnaire - Germany

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

Revision R2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, Germany

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

Answer:

Firstly, it is important to establish that in Germany, according to § 7 para. 1, sentence 2 of the Nuclear Energy Act (AtG), no further licences will be issued for the construction and operation of installations for the fission of nuclear fuel for commercial generation of electricity or of facilities for the reprocessing of spent nuclear fuel. Accordingly, for nuclear installations, nuclear licensing procedures are only performed for essential modifications (§ 7 para. 1 AtG) and their decommissioning (§ 7 para. 3 AtG). The granting of such licence is regulated in § 7 AtG, according to which a licence may only be granted if

- there are no known facts giving rise to doubts as to the reliability of the applicant and of the persons responsible for the erection and management of the installation and the supervision of its operation, and the persons responsible for the erection and management of the installation and the supervision of its operation have the requisite qualification,
- it is assured that the persons who are otherwise engaged in the operation of the installation have the necessary knowledge concerning the safe operation of the installation, the possible hazards and the protective measures to be taken,
- the necessary precautions have been taken in the light of the state of the art of science and technology to prevent damage resulting from the erection and operation of the installation,
- the necessary financial security has been provided to comply with the legal liability to pay compensation for damage,
- the necessary protection has been provided against disruptive action or other interference by third parties, and
- the choice of the site of the installation does not conflict with overriding public interests, in particular in view of its environmental impacts.

In addition to the continuous regulatory supervision, comprehensive periodic safety reviews are performed every ten years. Since 2002, the obligations to conduct the safety reviews and to submit the results on specified dates are also regulated by law in § 19a AtG.

After having consulted the *Länder*, the BMUB publishes regulatory guidelines (in the form of requirements, guidelines, criteria and recommendations). In general, these are regulations passed in consensus with the competent licensing and supervisory authorities of the *Länder* on the uniform application of the AtG in the federal system of Germany.

Germany has many processes (e.g. analysis of operational experience feedback, participation in IAEA, OECD/NEA and European working groups, insights from bi-lateral and multi-lateral contacts, insights from international peer reviews, insight from research projects, etc.) in place to follow and determine the most recent state of science and technology in nuclear safety. In case of generic safety issues, the national regulatory framework will be continuously improved.

In relation to the licence holders of the nuclear installations, the regulatory framework becomes binding by being taken into account in nuclear licences or orders of the nuclear supervisory body.

Furthermore, according to § 7 para. 2 subpara 3 AtG in conjunction with § 104 para 1 Sentence 3 StrlSchV it is guaranteed that, at the time of their licensing, nuclear installations meet the latest technical standards. Based on a legal presumption, the necessary precautions against damage have been taken in the light of the state-of-the-art of science and technology to prevent damage resulting from the erection and operation of the installation if the design of the installation was based on those accidents that were to be taken into account due to the latest nuclear regulations (“Safety Requirements for Nuclear Power Plants” and other regulatory guidelines).

Following the principle of dynamic precaution, the necessity and proportionality of additional precautionary measures has to be checked, especially whenever new safety-relevant findings are available.

The most important nuclear regulations are the “Safety Requirements for Nuclear Power Plants”, including their “Interpretations”. These contain fundamental and overriding safety-related requirements within the framework of the non-mandatory guidance instruments which serve for putting in concrete terms the precaution in line with the state of the art in science and technology against damage caused by the construction and operation of the plant as stipulated in § 7 para. 2, subpara. 3 AtG. During the development of the German “Safety Requirements for Nuclear Power Plants” the Reactor Safety Commission (RSK, an advisory committee to BMU regarding nuclear safety) has published a statement on “RSK’s understanding of safety philosophy”, which is described below.

“*According to § 1, No. 2 of the Atomic Energy Act (Atomgesetz – AtG), life, health and real assets are to be protected against the hazards of nuclear energy and the harmful effects of ionising radiation. This is the fundamental safety objective. To achieve this safety objective, it must be ensured in particular that precautions have been taken as are necessary according to the state of the art in science and technology to prevent damage caused by the construction and operation of a nuclear facility. The facility must be designed and operated such that it can be reliably shut down and kept in shutdown state, residual heat can be removed and radiation exposure to personnel and the environment can be kept as low as possible even if below the dose limits specified by the provisions of the Atomic Energy Act and the statutory ordinances promulgated on the basis of this Act at any time during specified normal operation and design basis accidents. Moreover, organisational and technical measures are to be provided to the extent appropriate to mitigate the consequences of beyond design basis plant states.*

The safety philosophy described below is intended to support a consistent interpretation of the technical and organisational requirements specified in the “Safety Requirements for Nuclear Power Plants” and a coherent classification of future new requirements according to the defence-in-depth concept.

For events that may affect the effectiveness of barriers and retention functions or the compliance with the main safety functions directly or indirectly, measures and installations are to be provided to prevent and control such events to the extent necessary. In this respect, the principle of “the more ... the lower” is to be applied, i.e. the more often an event is to be expected, the lower should the resulting potential radiological impact be or, the more severe the consequences of an event may be, the lower should be its frequency of occurrence. Here,

- *plant operation with as little occurrences as possible is to be ensured by principles of de-sign, manufacture and operation that enhance reliability, and deviations from normal operation are to be detected at an early state and largely limited so that operational occurrences are prevented,*

- *operational occurrences that occur nonetheless are to be controlled, if possible, thus pre-venting the occurrence of design basis accidents,*

- *the reliable control of design basis accidents postulated nonetheless is to be ensured, thus preventing the progression of a design basis accident to a beyond design basis plant state, and*

- *for the case of a beyond design basis plant state (severe accident) occurring nonetheless, the consequences are to be mitigated and event sequences with early or large releases of radioactive material to be excluded.*

A general safety objective of the concept of successive levels of defence is to enable the control of event sequences that might be uncontrollable at a certain level of defence at the next level of de-fence.

The events and plant states that may lead to safety-relevant deviations from normal operation are to be classified according to event classes and levels of defence together with the respective initial and boundary conditions and postulates to be considered and acceptance targets to be complied with. A distinction is to be drawn between the following event classes and associated levels of defence. The frequencies specified here are to be understood as guidance values:

- *Specified normal operation (level of defence 1)*

- *Anticipated operational occurrences (level of defence 2)*

  - *approx. 10-2/a*

- *Design basis accidents (level of defence 3)*

  - *< approx. 10-2/a to > approx. 10-5/a*

- *Beyond design basis accidents (level of defence 4)*

  - *< approx. 10-5/a*

For internal and external hazards not assigned to a level of defence in the “Safety Requirements for Nuclear Power Plants”, requirements apply in terms of a plant- and site-

specific assignment of the hazard-related parameters to the levels of defence, using the guidance values.

For weather-related external hazards, assignment of the requirements to level of defence 3 is sufficient up to a frequency of 10^{-4} 1/a since for lower frequencies and thus a tendency to greater impacts, these can either be identified at an early stage and thus measures at level of defence 4 can be initiated at an early stage or the potential for damage is limited."

As Germany has no dedicate regulatory framework for research reactors, regulations and standards are appropriately applied to research reactor taking the lower hazard potential into account (i.e. application of a graded approach), This is always a case by case decision because the German research reactors are quite different covering a range of thermal power from 100 mW to 20 MW. Safety has always been the highest priority when testing and evaluating research reactors. Thus, there is no fundamental difference in supervision between research reactors and nuclear power plants.

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

Answer: In Germany, issuing licences for construction of new nuclear power plants are prohibited by the Atomic Energy Act (§ 7 AtG). In Germany, the state of the art in science and technology in nuclear safety has to be applied when making regulatory decisions to grant a licence according to § 7 para. 2, subpara. 3 AtG, irrespectively if it is a new or existing nuclear installation.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

Answer:

The same radiological objective as defined in IAEA SSR 2/1 “Safety of NPPs:Design” and in the WENRA Safety Objectives for new reactors is applied for existing Reactors in Germany. In the German “Safety Requirements for NPPs” it is required that any releases of radioactive materials into the environment of the plant, caused by the early failure or bypass of the containment and requiring measures of the external accident management for the implementation of which there is not sufficient time available (early release) or any releases of radioactive materials into the environment of the plant requiring wide-area and long-lasting measures of the external accident management (large release)shall be either excluded or their radiological consequences shall be limited to such an extent that measures of the external accident management will only be required to a limited spatial and temporal extent.

To achieve this radiological objective several technical safety concepts have to be fulfilled. This are the defence-in-depth concept as the overarching concept which is supported by the concept of the multi-level confinement of the radioactive inventory (barrier concept), the achievement of the three fundamental safety functions and the protection concept against internal and external hazards as well as against very rare human-induced external hazards. In addition, for all plant states and levels of defence-in-depth radiological objectives have been formulated as well as technical acceptance criteria and targets.

"Level of defence 1:

The objective of level of defence 1 is to ensure normal operation (undisturbed, specified normal operation) and to avoid abnormal operation.

Level of defence 2:

The objective of level of defence 2 is the control of operational occurrences and the avoidance of abnormal operation. The level of defence is characterised by the undisturbed, specified normal operation.

At the second level of defence, particular importance is attached to the limitation systems that pre-cede the reactor protection system. There are three types of limitation systems that are classified according to task and requirement. In case of anticipated operational occurrences, the limitations shall automatically limit the process variables to defined values in order to increase the availability of the installation (operational limitations) and to maintain initial conditions for the accidents to be considered (limitations of process variables). Furthermore, safety variables are brought back to values at which continuation of specified normal operation is permissible (protective limitations). Operational limitations are instrumentation and control systems with increased reliability which, for the rest, are comparable with the control systems.

The overall objective is to reach a high degree of automation for relief of man from short-term measures and comprehensive preventive measures to counteract the development of anticipated operational occurrences into accidents and a high tolerance against human failures. The require-ments for comprehensive, reliable and user friendly process information systems also provide technical support for personnel actions. The objective is to enable man to fulfil his safety task with-in the overall system in an optimal manner.

Level of defence 3:

The objective of level of defence 3 is the control of design basis accidents and the prevention of multiple failure of engineered safety features safety. For this purpose, highly reliable safety systems and the reactor protection system are used.

Level of defence 4a:

The objective of level of defence 4a is the control of events with postulated failure of the reactor scram system (ATWS).

Level of defence 4b:

The objective of level of defence 4b is the control of events with multiple failure of safety systems to prevent accidents with severe core damage.

Here, preventive measures of accident management (level of defence 4b) are used which are to maintain or restore core cooling and transfer the installation into a safe state.

Level of defence 4c:

Subsection 2.1 (3b) of the "Safety Requirements for Nuclear Power Plants" stipulates that on level of defence 4c "mitigative measures of the internal accident management shall be provided for accidents involving severe fuel assembly damages for the purpose of maintaining – by using all available measures and equipment – the integrity of the containment for as long as possible, excluding or limiting releases of radioactive materials into the environment according to Subsection 2.5 (1), and achieving a long-term controllable plant state."

The mitigative measures of level of defence 4c are provided in order to practically exclude events that could lead to any releases of radioactive materials caused by the early failure of the containment or any releases of radioactive materials requiring wide-area and long-lasting measures of off-site emergency preparedness, or to limit their radiological consequences to such an extent that off-site emergency preparedness measures will only be required to a limited spatial and temporal extent. For the nuclear installations in operation, the practical exclusion of events with early or large releases is proven by the interaction of plant operation, high reliability of the safety system and a comprehensive accident management." [CNS 7. National Report]

For levels of defence in depth 2 to 4a acceptance targets and acceptance criteria are defined quantitatively as well as qualitatively. An overview is provided in Annex 2 of the "Safety Requirements for Nuclear Power Plants".

In addition, radiological safety objectives defined in the "Safety Requirements for Nuclear Power Plants":

"*On levels of defence 1 and 2*

- *radiation exposure of the personnel shall be kept as low as achievable for all activities, even below the limits of the Radiation Protection Ordinance (20 mSv per year for* *radiation exposed workers, 1 mSv for non-radiation exposed workers), taking into account all circumstances of each individual case,*

- *any discharge of radioactive materials with air (dose to the public max. 0.3 mSv) or water (dose to the public max. 0.3 mSv) shall be controlled via the specially provided discharge paths; the discharges shall be monitored as well as documented and specified according to their kind and activity, and*
- *any radiation exposure or contamination of man and the environment by direct radiation from the plant as well as by the discharge of radioactive materials shall be kept as low as achievable, even below the limits of the Radiation Protection Ordinance, taking into account all circumstances of each individual case.*

On level of defence 3

- *the maximum radiation exposure limits for the personnel in connection with the planning of activities for the control of events, the mitigation of their effects or the removal of their consequences shall not exceed the relevant limits of the Radiation Protection Ordinance (),*
- *the maximum design limits for the plant for protecting the population against any release-induced radiation exposure shall not exceed the relevant accident planning levels of the Radiation Protection Ordinance (50 mSv),*
- *any release shall only happen via specially provided release paths; the release shall be monitored and shall be documented and specified according to its kind and activity; and*
- *the on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.*

On level of defence 4

- *the planning of activities to control events of level of defence 4a as well as for the planning of activities in connection with internal accident management measures shall be based the relevant requirements of the Radiation Protection Ordinance regarding the anticipated radiation exposure of the personnel,*

- *the monitoring of releases of radioactive materials from the plant according to their kind and activity shall be ensured and*
- *the on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.*

Taking into account the measures and equipment for the internal accident management provided on levels of defence 4b and 4c,

- *any releases of radioactive materials into the environment of the plant, caused by the early failure or bypass of the containment and requiring measures of the external accident management for the implementation of which there is not sufficient time available (early release), or*
- *any releases of radioactive materials into the environment of the plant requiring wide-area and long-lasting measures of the external accident management (large release) shall be excluded, or their radiological consequences shall be limited to such an extent that measures of the external accident management will only be required to a limited spatial and temporal extent.*"

For more information on the protection concept against internal and external hazards as well as against very rare human-induced external hazards see answers to questions 3.4.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

Answer: Question not applicable for Germany.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Answer: In the German nuclear rules and regulations, external hazards are addressed on various levels. Essential requirements with respect to the protection of NPPs against external hazards are stipulated in Section 2.4 (see below) and 4.2 of the Safety Requirements for Nuclear Power plants. These basic requirements are supplemented by more detailed general and hazard-specific requirements in Annex 3 of the same document.

"*Section 2.4 (1) of the "Safety Requirements for Nuclear Power Plants" requires that all equipment that is necessary for shutting the reactor down safely, for maintaining it in shutdown condition, for removing the residual heat or for preventing a release of radioactive materials shall be designed such and be able to be maintained in such a condition that they fulfil their safety-related functions even in the case of internal and external hazards as well as very rare human induced external hazards. In this respect, the following hazards have to be considered in particular:*

- *Natural external hazards, as far as to be considered site-specifically, such as earthquake, flooding, extreme meteorological conditions (e.g. high or low temperatures of outside air or cooling water, storm, snowfall, icing, lightning stroke) or biological impacts*
- *Man-made external hazards, such as aircraft crash, plant-external blasts, impact of dangerous substances and other man-made hazards (e.g. impact of flotsam, loss of cooling water due to failure of a river barrage downstream, consequences of shipping accidents)*

In the nuclear rules and regulations, accidental aircraft crash, blast wave and the impact of hazardous substances are referred to as very rare human-induced external hazards or man-made hazard conditions. Man-made hazard conditions are controlled by means of specially protected emergency equipment. For these, less stringent redundancy requirements apply than for the systems for accident control (level 3 of the defence-in-depth concept) which have to control the single failure and the simultaneous maintenance case in the event of a hazard-induced impact." [CNS 7. National Report]

Protection against accidental aircraft crash is based on a site-independent impact load time diagram corresponding to the impact of a fast-flying military aircraft of the "Phantom" type. It is furthermore specified, amongst other things, that the impacts of debris and of kerosene fires as well as the vibrations induced by the impact of the aircraft have to be taken into account in the design.

"*The protective measures against pressure waves from accidents outside the installation are based on the assumption of a maximum pressure of 0.45 bar at the site and that a certain safety distance is kept to potential blast or release locations. They are regulated in detail in the guideline for the protection of nuclear power plants against pressure waves from chemical reactions by means of the design of nuclear power plants with regard to strength and induced vibrations and by means of the adherence to safety distances.*" [CNS 7. National Report]

Detailed requirements regarding the leading natural hazards, e. g. earthquakes, flooding and lightning, are specified in the corresponding Safety Standards of the Nuclear Safety Standards Commission (KTA) and recommendations of the Reactor Safety Commission (RSK) ["Lightning with parameters exceeding the standardised lightning current parameters" and "Minimum value of 0.1g (approx. 1.0 m/s²) for the maximum horizontal ground acceleration in an earthquake" (in German only)]. As examples, the basic approaches with respect to flooding and earthquakes are described below.

"*According to nuclear safety standard KTA 2207 "Flood Protection for Nuclear Power Plants", permanent flood protection measures shall be provided. Under special boundary conditions, protection against the difference between the water levels of the flood with an exceedance probability value of 10^{-2}/a and the design basis water level of 10^{-4}/a may also be provided by temporary measures.*"[CNS 7. National Report] In addition to the requirements of KTA 2207, the RSK recommends a systematic assessment of aleatoric and epistemic uncertainties in the determination of the design basis flood and a comparison of the design basis flood with flood events observed in the region ["Aspects of the determination of the site-specific design basis flood"].

The design against earthquakes is based on safety standard KTA 2201.1 "Design of Nuclear Power Plants against Seismic Events; Part 1: Principles". The design basis earthquake is determined on the basis of deterministic and probabilistic analyses. For both methods, wider surroundings of the site have to be considered. The deterministic determination of the design basis earthquake is to be based on an earthquake with the maximum seismic impact assumed for the site – taking into account events that have occurred in the past – that can be expected according to scientific knowledge. The probabilistic determination of the parameters of the design basis earthquake has to take an exceedance probability of 10^{-5}/a (median) into account. The design basis earthquake will then be conclusively defined taking into account the results of both analyses.

In the aftermath of the nuclear accidents at the Fukushima Daiichi site in Japan, RSK has initiated a systematic robustness analysis to check the effectiveness of the vital safety functions for beyond design basis external or internal hazards ["Plant-specific safety review of German nuclear power plants in the light of the events in Fukushima-1 (Japan)" and "Recommendations of the RSK on the robustness of the German nuclear power plants"]. Based on the results of this review RSK has issued several statements confirming the existence of safety margins for a broad spectrum of external hazards ["Plant-specific safety review (RSK-SÜ) of German nuclear power plants in the light of the events in Fukushima-1 (Japan)", "Evaluation of the implementation of RSK recommendations in response to Fukushima" (in German only), "RSK summary statement on man-made hazards, aircraft crash - Reference report: definition of load assumptions and assessment of Konvoi plants" (in German only) and "Evaluation of the implementation of RSK recommendations in response to Fukushima" (in German only)]. For extreme weather conditions RSK has recommended to review the robustness of NPPs with respect to design basis weather conditions with a return frequency of 10^{-4}/a or to demonstrate a high level of robustness deterministically using engineering judgement In addition, it was suggested to consider conditions beyond these conditions in terms of robustness with engineering assessments for the determination of safety margins. ["Assessment of the coverage of extreme weather conditions by the existing design"]. Besides the RSK, also the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) addressed lessons learned from the Fukushima Daiichi nuclear accidents in an Information Notice ["Auswirkungen des Tohoku-Erdbebens an den japanischen Kernkraftwerksstandorten Fukushima Dai-ichi (1) und Dai-ni (11)am 11.03.2011 und desNiigataken Chuetsu-Oki-Erdbebens am japanischen Kernkraftwerksstandort Kashiwazaki-Kariwa am 16.07.2007" (in German only)] recommending, e. g., measures to ensure electrical power supply and essential cooling water supply under extreme plant conditions.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

Answer: The terms "early radioactive release" and "large radioactive release" are defined in the radiological safety objectives in the German "Safety Requirements for Nuclear Power Plants".

In Germany the following definitions apply:

- "*An early release is any releases of radioactive materials into the environment of the plant, caused by the early failure or bypass of the containment and requiring measures of the external accident management for the implementation of which there is not sufficient time available.*
- *"A large release is any releases of radioactive materials into the environment of the plant requiring wide-area and long-lasting measures of the external accident management.*"

In the "Safety Requirements for Nuclear Power Plants" a demand can be found that both early and large releases must be avoided. It is required, that by "*taking into account the measures and equipment for the internal accident management provided on levels of defence 4b and 4c,*

- *any releases of radioactive materials into the environment of the plant, caused by the early failure or bypass of the containment and requiring measures of the external accident management for the implementation of which there is not sufficient time available (early release), or*
- *any releases of radioactive materials into the environment of the plant requiring wide-area and long-lasting measures of the external accident management (large release)*

shall be excluded, or their radiological consequences shall be limited to such an extent that measures of the external accident management will only be required to a limited spatial and temporal extent."

By deterministic severe accident analyses the utility has to show that early and large releases can be avoided with a high level of confidence and finally the results have to be approved by its regulator. In principle, this is done in the frame of Level 2 PSA which is performed within the periodic safety review of the plants.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Answer:
In Germany, the decision if a possible safety improvement is reasonably practicable and if it is implemented timely, is based on a case-by-case analysis and subject to the overarching principle of proportionality. This concept is based in the Grundgesetz (German Constitution) and is applicable for any administrative decision in Germany. Thus, any measure has to be suitable, necessary, and appropriate. When applying this test, the economic impact of a measure will not be assessed. This means that the intensity of a safety measure to be carried out solely depends on the nuclear risk that is associated with the installation or activity. The same is true for the factor of timely implementation.

However, the decision-making is dependent of the scenario.

(1) Firstly, it needs to be established that according to § 7 para. 2 No. 3 AtG, the necessary precautions in the light of the state-of-the-art of science and technology to prevent damage resulting from the erection and operation of the installation have to be taken. If a safety deficiency leads to the conclusion that this precaution for safe operation is not fulfilled and the licensee cannot provide new evidence which convinces the authorities of the safety of the installation, the licensee needs to implement the necessary backfitting measures. Until these are implemented, the regulatory authority can order the interim closure of the installation. Failure to implement the necessary backfitting measures can lead to the final closure of the installation.

(2) Secondly, in Germany, PSRs as demanded within the framework of the “Vienna Declaration on Nuclear Safety” have been performed since the 1990s already. In 2002, the obligation of a decennial safety review of each nuclear installation in power operation was anchored in § 19a AtG. This obligation guarantees the continuous improvement of nuclear safety. This provision requires the license holders of nuclear installations to improve the nuclear safety of the nuclear installation continuously on the basis of the PSR.

Further, the assessment of the safety of the nuclear installations is continuously reviewed by the competent Land authorities within the framework of the nuclear supervisory procedure. If there are any new safety-relevant findings, the need for the implementation of safety-related improvements is examined. This is done by reviewing documents on site at the nuclear installations.

As part of nuclear supervision, safety assessments conducted by the licence holders are reviewed both continuously and discontinuously by the nuclear licensing and supervisory authorities of the Länder, as are the special periodic safety reviews stipulated by §19a AtG.

The license holder shall submit the review results and the associated verifiable documents to the supervisory authority and shall identify areas of improvement (from his perspective). The supervisory authority reviews the documents and assessments submitted and decides which measures are to be implemented for the continuous improvement of the nuclear safety of the installation. Necessary safety-enhancing measures or upgrades are in most cases implemented by the licence holders on a voluntary basis. If not, the regulatory authority can require the licence holder to do so in accordance with the principle of proportionality.

(3) Thirdly, the licensees of plants for the fission of nuclear fuels for the commercial generation of electricity are required, according to § 7d AtG, to ensure that safety precautions are implemented in accordance with the progressing state-of-the-art of science and technology. These safety precautions must be developed, suitable and appropriate in each case in order to make not only a minor contribution to further prevention against risks for the general public.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

Answer: Continuous improvement of nuclear safety is a concept followed in Germany since the late 1970ies. As indicated above (Answer to Question 3.6), The idea of continuous improvement is implemented in the German Atomic Energy Act and nuclear safety has to follow the state-of-the-art in science and technology. This is one of the prerequisites defined in the Atomic Energy Act which has to be fulfilled before the authority can issue a license (→ § 6 and § 7 para. 2 No. 3 Atomic Energy Act). Furthermore, after a license has been issued the licensee is obliged to implement the necessary safety improvements, except this measure will have only a negligible effect on the further reduction of risk (→ § 7d Atomic Energy Act). In the "Safety Requirements for Nuclear Power Plants", priority to safety is further specified as follows:

- The licensee shall give priority to safety over all other business objectives.
- The prime objectives of the integrated management system (IMS) are specified as:
  - the guarantee of safety,
  - the continuous improvement of safety, and
  - the promotion of safety culture.

"If, in the course of regulatory supervision, there are any new safety-related findings, their applicability to other nuclear installations and the need for any possible backfitting measures is examined. BMU keeps continuously up to date with the developments in the area of nuclear safety by taking an active part in the work of international committees and working groups (IAEA, OECD/NEA, committees resulting from bi- and multilateral agreements and treaties, etc.). The results of the work of these committees and working groups as well as of the research programmes and research and development projects sponsored by the Federal Government at international level influence the constant improvement of the requirements for the safety of the nuclear installations in accordance with the state of the art in science and technology. The BMUB also requests its advisory commissions RSK, ESK and SSK to comment on selected developments and events in the area of nuclear safety and to make recommendations. The expert organisation GRS supports the BMU and carries out its own

research on the safety of nuclear installations.from a generic point of view by request of the BMUB. GRS evaluates events that have occurred in German and also in foreign nuclear installations with regard to their safety significance and applicability to other installations and prepares recommendations in the form of information notices (WLNs)." [CNS 7. National Report]. Insights from the different process described below are used to determine the most recent state-of-the-art in science and technology in the field of nuclear safety. These findings are continuously benchmarked against the national regulatory framework to identify potential needs to improve the existing regulations and to update the German regulations accordingly.

- Germany has a well-established system for operating experience feedback. Every licensee is obliged to report events occurred in his plant to the authority. Criteria for reporting are established in the Nuclear Safety Officer and Reporting Ordinance (AtSMV).

- Germany takes active part in various peer review missions, like IRRS-Missions, WANO-Missions, OSART-Missions or benchmarks on the European level like the ENSREG stress test or the required Topical Peer Reviews. Findings are carefully assessed and potential improvements for the regulatory system as well as for the NPPs will be discussed and implemented whenever necessary to further improve nuclear safety. In addition, Germany takes part in further self-assessments and benchmarking processes, like the RHWG-Benchmark on implementation the updated WENRA Reference Level published 2014.

- Germany is actively engaged in and continuously follows the development of international safety standards by continuously performing the following tasks:

    - "*Active involvement in all IAEA safety standards committees (CSS, NUSSC, RASSC, WASSC, TRANSSC);*

    - *Secondment of technical experts for the development and revision of IAEA safety standards;*

    - *Formal public participation in the process of providing comments on IAEA safety standards by the member states. For this purpose, the relevant drafts are published in the Federal Gazette with an invitation to submit comments;*

- *Preparation of annual summary reports on the work of the IAEA on safety standards. This has been done by the BMU since 2006;*
- *Participation in the development and revision of the "WENRA Safety Reference Levels" and Safety Objectives for new nuclear power plants.*" [CNS 7. National Report]

- Since the beginning of the 1990s, safety reviews (SÜs) have been carried out every ten years according to standardized national criteria. They consist of a deterministic safety status analysis, a probabilistic safety analysis and a deterministic analysis of the physical protection of the installation. Since 2002 the licensee is obliged to perform a periodic safety review every 10 years in accordance with § 19a of the Atomic Energy Act. During the process the nuclear power plant under consideration is benchmarked against the latest state-of-the-art in nuclear safety. In such cases, where safety improvements have been identified, the licensee will propose an action plan to implement the safety improvements.

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Answer: While developing the German "Safety Requirements for Nuclear Power Plants" the most recent state of the art in science and technology has been considered. The WENRA Safety Objectives, WENRA Refernce Levels, IAEA Safety Standards (in particular SSR 2/1 and SSR 2/2) have been consulted and implemented in the German regulations. In addition, lessons learned from the accident at the Fukushima Daiich-NPP were considered. The confinement of the radioactive materials presents in the nuclear power plant as well as the shielding of the radiation emanating from them have to be ensured. In order to achieve this objective, the German safety concept comprises measures and equipment that are allocated to different levels of defence. These comprise normal operation, anticipated operational occurrences (like transients), design basis accidents and very rare events (beyond design basis accidents). The prevention of events and the control of events is provided by measures and equipment for all these levels of defence as well as against internal and external hazards including very rare human induced external hazards.

Furthermore, additional measures and equipment to identify and limit the consequences of plant conditions that are not allocated to the above-mentioned levels of defence (up to level of defence 4a) due to their low probability of occurrence are provided as a precaution. Therefore, additional measures and equipment of the internal accident management are installed or planned, respectively, on levels of defence 4b and 4c of the defence-in-depth concept. These levels of defence are characterised by the following plant conditions:

- Level of defence 4b: events involving the multiple failure of safety equipment
- Level of defence 4c: accidents involving severe fuel assembly damages.

The most important measures taken after the Fukushima accident are described in the answer to question 4.2. [CNS 7. National Report]

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

Answer: After the issue of the WENRA safety objectives for new reactors and the IAEA SSR-2/1 the main changes in German NPPs have been made due to the action plan after the Fukushima accident. These actions were also based on the latest revisions of the German Atomic Act and the "Safety Requirements for Nuclear Power Plants". However, many safety improvements have been already implemented in 1990ies years. Important safety improvements mainly based on research results and operating experience feedback have been implemented in cooperation with the licence holders. For German PWRs and BWRs the following safety improvements have been implemented:

- Emergency manual (PWR and BWR)
- Secondary-side bleed (PWR)
- Secondary-side feed (PWR)
- Primary-side bleed (PWR)
- Primary-side feed (PWR)
- Diverse emergency HPCI system (steam-driven pump) (BWR)
- Additional RPV injection and makeup system (BWR)
- Containment isolation (PWR and BWR)
- Inerting of containment with nitrogen (BWR)
- Diverse RPV pressure limitation (BWR)
- Filtered containment venting (PWR) / Filtered venting (BWR)
- Passive autocatalytic recombiners (PWR)
- Filtering of control room air (PWR and BWR)
- Emergency power supply by neighbouring unit (PWR and BWR)
- Sufficient battery capacity (PWR and BWR)
- Re-establishment of the external electrical energy supply (PWR and BWR)
- 3rd grid connection (underground cable) (PWR and BWR)
- Containment sampling system (PWR and BWR)

The main improvements to German NPPs in the aftermath of the accident at the Fukushima Daiichi NPP have been:

- Further measures against station blackout
  - Residual heat removal, accident overview measurements, necessary lightning for 10 hours with batteries only
  - Additional power generators to be operable after 10 hours
  - Service water supply with independent power supply and independent auxiliary systems
- Additional ultimate heat sink
- Independent emergency water resources
- Mobile emergency equipment
- Supply of emergency power and water from outside of the containment
- Emergency fuel pool cooling
- Full implementation of the Severe Accident Management Guidelines (SAMG)

Another important measure undertaken by the Federal Environment Ministry (BMU) following the analysis conducted in June 2011 into what happened in Fukushima was commissioning the Commission on Radiological Protection (SSK, an advisory committee to BMU regarding radiation protection) with a review of the statutory regulations governing the off-site emergency response. The course of events in Japan differed greatly from that of Chernobyl, allowing new experiences to be gained in practically every field of emergency preparedness. The existing analyses conducted by the Japanese government and the International Atomic Energy Agency into the accident, the RSK safety review series as well as experience and observations made by the SSK crisis unit were taken into account during the review of the statutory regulations. In February 2015, the SSK published more than 70 recommendations on the advancement of emergency response based on experience gained in Fukushima. [EU stress test National Report Germany]

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- PSA level 1:
- PSA level 2:
- PSA level 3:

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

Answer: PSA is required for all existing nuclear power plants in Germany according to the high level German nuclear regulation “Safety Requirements for Nuclear Power Plants”. These “Safety Requirements for Nuclear Plants” require:

“5 (5a) “To supplement the deterministic safety demonstrations, the balance of the safety-related design shall be verified by probabilistic safety analyses (PSAs).

5 (5b) To supplement the deterministic safety demonstrations, probabilistic safety analyses shall also be done to assess the safety significance

of modifications of measures, equipment or the operating mode of the plant, as well as

of findings that have become known from safety-relevant events or phenomena that have occurred and which can be applied to the nuclear power plants in Germany that are referred to in the scope of application of the "Safety Requirements for Nuclear Power Plants"

for which a significant influence on the results of the PSA can be expected.

5 (5c) Compared with the unchanged condition of the plant, modifications of measures, equipment or the operating mode of the plant must not lead to an increase in the average core damage frequency and the average frequency of large and early releases, neither for power *operation nor for low-power and shutdown states, considering all plant-internal events as well as all internal and external hazards as well as very rare human induced external hazards."*

According to the Atomic Energy Act, PSA is mandatory to be performed in the frame of the (Periodic) Safety Reviews (SR).

Guidance for the application of PSA in the frame of the SR is given in "Safety Review for Nuclear Power Plants pursuant to § 19a of the Atomic Energy Act - Guide Probabilistic Safety Analysis - of 30 August 2005, Edition 08/05" and its technical supplements on "Methods for Probabilistic Safety Analysis of Nuclear Powers Plants" [1], "Data for Quantification of Event Sequence Diagrams and Fault Trees" [2], and on "Methods and Data for Probabilistic Safety Analysis of Nuclear Power Plants" [3] (published in 2016).

For covering PSA outside the SR as required in par. 5 (5c) of the "Safety Requirements for Nuclear Power Plants", another guidance document "PSA Applications Outside Periodic Safety Reviews" has been published in 2018. PSA is also applied in the frame of precursor analyses

"*The methods and data applied for the PSA are described in technical documents ("Methods" and "Data" volumes for the probabilistic safety analysis for nuclear power plants)*

[1] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: August 2005, BfS-SCHR-37/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, Oktober 2005, in German,http://doris.bfs.de/jspui/handle/urn:nbn:de:0221-201011243824

[2] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Daten zur Quantifizierung von Ereignisablaufdiagrammen und Fehlerbäumen, Stand: August 2005, BfS-SCHR-38/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, Oktober 2005, in German, https://doris.bfs.de/jspui/handle/urn:nbn:de:0221-201011243838

[3] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden und Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: Mai 2015, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, BfS-SCHR-61/16, September 2016, in German, https://doris.bfs.de/jspui/bitstream/urn:nbn:de:0221.../3/BfS-SCHR-61-16.pdf

[3] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden und Beispiele für die probabilistische Bewertung sicherheitsrelevanter Fragestellungen außerhalb der SÜ, Stand: Mai 2015, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, BfS-SCHR-03/18, Januar 2018, https://doris.bfs.de/jspui/bitstream/urn:nbn:de:0221-2018013014519/3/BfE-SCHR-03-18_FAK%20PSA.pdf.
Englisch: PSA Applications Outside Periodic Safety Reviews, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, January 2018 (in German only, will be published in English by end-2018)

supplementing the "Guide Probabilistic Safety Analysis" and were first published in 1996 and updated in 2005.

Since 1990, the licence holders operating the German nuclear installations have performed Level 1 PSAs as part of the periodic safety review for all German nuclear installations. Level 2 PSAs also Article 14 - 104 - exist for all nuclear installations in power operation. The Level 1 PSAs in particular have led to technical and procedural improvements at the nuclear installations.

Since 2005, a Level 1 PSA has comprised

- *plant-internal initiating events for all operating states (power operation and low-power and shutdown states),*
- *for power operation, common-cause initiators such as fire, internal flooding,*
- *postulated site-specific external hazards such as*
  - *aircraft crash,*
  - *blast wave,*
  - *flooding, and*
  - *site-specific earthquake with an intensity of more than 6 on the MSK (MedvedevSponheuer-Karnik) scale.*

A Level 2 PSA has to be performed for internal initiating events for power operating conditions.

The FAK PSA (Facharbeitskreis Probabilistische Sicherheitsanalyse für Kernkraftwerke) technical committee established by the BMU and coordinated by the BfE, is a committee of experts in the field of PSA. The FAK PSA works out proposals for the updating of technical documents on PSA methods and data according to the established state of knowledge. A revision and updated version of the methods and data volume of the PSA Guide was presented to the Technical Committee for Nuclear Safety (Fachausschuss Reaktorsicherheit – FARS) of the Federal/Länder Committee for Nuclear Energy (LAA) for approval and resolution in 2015 and is to be adopted in 2016. It contains supplementary documents on the topic areas "Level 2 PSA", "PSA for low-power and shutdown states", "Consideration of the human factor in a PSA" and "PSA for external hazards", which need to be looked at in more detail to be in line *with the state of the art in science and technology. Since according to the 13th AtG amendment only two of the nine nuclear power plants in operation have to perform probabilistic safety analyses within the framework of the required safety review, a revision of the PSA Guide is no longer planned.*" [CNS 7. National Report]

The most recent technical document "Methods and Data for Probabilistic Safety Analysis of Nuclear Power Plants" also provides methods and data for Level 2 PSA during low power and shutdown phases.

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

Answer: The Federal German regulatory body has promulgated the "Safety Requirements for Nuclear Power Plants" covering also requirements for PSA. These include only qualitative criteria, requiring PSA methods up to Level 2 PSA for full power (FP) as well as low power and shutdown states (LPSD) covering also PSA for internal and external hazards. The assessment shall also include the spent fuel pool (SFP). Multi-unit aspects shall be taken into account; however, so far no detailed risk metrics are available. Quantitative threshold values and criteria are not provided in Germany. More details on the German approach can be found in the answer to question 5.1.

Detailed guidance is provided in the German Guide "Probabilistic Safety Analysis" and the supporting technical guidance documents[4,5,6]. In addition, as a minimum the requirements of WENRA, in particular WENRA Safety Reference Levels, and IAEA safety Standards, typically

[4] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: August 2005, BfS-SCHR-37/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, Oktober 2005, in German,http://doris.bfs.de/jspui/handle/urn:nbn:de:0221-201011243824

[5] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Daten zur Quantifizierung von Ereignisablaufdiagrammen und Fehlerbäumen, Stand: August 2005, BfS-SCHR-38/05, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, Oktober 2005, in German, https://doris.bfs.de/jspui/handle/urn:nbn:de:0221-201011243838

[6] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke: Methoden und Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, Stand: Mai 2015, Bundesamt für Strahlenschutz (BfS), Salzgitter, Germany, BfS-SCHR-61/16, September 2016, in German, https://doris.bfs.de/jspui/bitstream/urn:nbn:de:0221.../3/BfS-SCHR-61-16.pdf

SSG-3, SSG-4, TECDOC 1229, Safety Report Series No. 25 and TECDOC 1511, are considered as well.

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Answer: According to the "Safety Requirements for Nuclear Power Plants", the probabilistic assessment shall cover both the reactor and the spent fuel pool (SFP). Details see also answer to question 5.1. So far, detailed requirements for PSA of non-reactor radioactive sources do not yet exist. Developments are however ongoing to develop approaches for a complete site-level PSA covering multiple installations (multi-unit, multi-source) within PSA.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Answer: Multi-unit aspects shall be taken into account; however, so far no detailed risk metrics are available. Because of the German nuclear phase-out even at German two-unit sites only one reactor is still in commercial operation. The second reactor unit is either already under decommissioning or in the post-commercial safe shutdown phase with the fuel elements in the spent fuel pool (SFP).

Developments are ongoing to develop approaches for a complete site-level PSA covering multiple installations (multi-unit, multi-source) within PSA.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

Answer: In the recent past, a new guidance document "PSA Applications Outside Periodic Safety Reviews" has been developed covering PSA applications out of the SR. Guidance is provided on how to apply PSA in case of specific requests by the regulatory authority in case of license and plant modifications according to this document.

Furthermore, Level 1 models are used for precursor analyses, the information provided by Level 2 PSA is used as a basis for fast source term prognosis tools by GRS on behalf of the regulatory body.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Answer: According to the Atomic Energy Act, PSA is mandatory to be performed in the frame of the (Periodic) Safety Reviews (SR) every ten years. Updates outside the safety reviews maybe needed according to the “Safety Requirements for Nuclear Power Plants”. Details see answer to question 5.1.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed?
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

Answer: After discussions among the regulatory authorities of the federal states in Germany it was agreed that in the frame of periodic safety reviews (SR)for two German research reactors with the highest thermal power (FRM-II and BER-II) a PSA for external hazards should complement the deterministic assessment. However, for FRM-II a full scope PSA up to Level 2 for all plant operating states has already been performed by the licensee in the frame of the last safety review. This PSA is still under review by the regulatory authority, details have not yet been published.

Moreover, activities are ongoing for developing a Level 1 PSA approach for an existing research reactor covering already plant internal initiating events. In the near future, this methodological approach will be extended to cover all types of internal and external hazards including hazard combinations. Extensions to Level 2 PSA are also planned. According to the new guidance document for PSA applications outside the SR[Error! Bookmark not defined.], applications to license and plant modifications are possible as well as to renewals of existing reactors and safety improvements to be licenced.

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

The following answers are only dedicated to existing German nuclear power plants as new buildings of NPPs are forbidden in Germany.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

Answer: DEC without fuel melt are assigned to level of defence in depth 4. Anticipated transients without scram belong to sublevel 4a and multiple failure of safety systems belong to sublevel 4b.

The following safety-enhancing design, manufacturing and operating principles are required for the measures and equipment on levels of defence-in-depth 1 to 4a by the German “Safety Requirements for Nuclear Power Plants”:

a) *“well-founded safety factors in the design of components depending on their safety significance; here, established rules and standards may be applied with regard to the case of application;*
b) *preference to inherently safe-acting mechanisms in the design;*
c) *use of qualified materials and manufacturing and testing methods and of equipment that has been proven by operating experience or which has been sufficiently tested;*
d) *maintenance- and test-friendly design of equipment, with special consideration of the radiation exposure of the personnel;*
e) *ergonomic design of the workplaces;*
f) *assurance and maintenance of the quality features during manufacturing, construction and operation;*
g) *execution of regular in-service inspections to an extent that is necessary from a safety-related point of view;*
h) *reliable monitoring of the relevant operating states in the respective operational modes;*

i) preparation and implementation of a monitoring concept with monitoring systems to detect and control operation- and ageing-induced damages;

j) recording, evaluation and safety-related use of operating experience.

Events with multiple failure of safety systems (level of defence-in-depth 4b) are controlled by means of plant internal accident management. This equipment must impair neither normal specified operation nor the use of safety and emergency equipment as specified by their design. Their compatibility with the safety concept hast to be ensured. The accident management measures rest on specially dedicated measures and equipment including equipment that is not permanently installed (mobile) as well as on the flexible use of available safety equipment, operating systems and emergency equipment. The measures and equipment of the accident management shall remain effective even in case of internal and external hazards as well as in case of human-induced external hazards if these hazards may lead to multiple failures of safety equipment that is necessary in these situations and if these measures and equipment contribute to the mitigation of the effects of the respective hazards and human-induced external hazards." [Safety Requirements for Nuclear Power Plants]

Differences exist concerning the safety analysis of postulated initiating events on levels of defence-in-depth 3, 4a and 4b:

"Level of defence 3 (design basis accident):

The initial plant conditions to be assumed for a safety demonstration shall bound the worst case for the different operational modes with regard to the respective acceptance criterion, or shall be realistic parameter values, taking into account their uncertainty range. The single-failure concept shall be applied. A loss of station service power supply occurring simultaneously or - depending on the event - with a time lag shall be postulated for all measures and equipment necessary for accident control if this will have an adverse effect on the event sequence. Emergency power supply shall be considered in the analysis according to the switch-on programme of the devices supplied with emergency power. In case of loss-of-coolant accidents the worst leak or break location for the different safety demonstrations, respectively, shall be determined and postulated for the range of leak and break sizes to be considered. In addition to the assumed loss of functions of the single-failure concept, safety demonstration shall also take into account accident-induced consequential loss of functions of measures and equipment with an adverse effect on the accident with regard to the acceptance target.

Level of defence 4a (ATWS)

In the analysis of anticipated transients without scram realistic initial and boundary conditions can be chosen. The initial condition of the reactor core, however, shall assume the power operation at the most unfavourable point in time of the cycle (with xenon equilibrium) loading- and event-specific. Additionally, with regard to reactivity feedback effects, values shall be applied that cover existing uncertainties. All measures and equipment that have not failed due to the postulated event may be assumed to be available.

Level of defence 4b (events involving the multiple failure of safety equipment):

For the analysis of the effectiveness of preventive accident management measures on level of defence-in-depth 4b realistic models and realistic initial and boundary conditions can be used for the event sequences on which they are based." "Safety Requirements for Nuclear Power Plants"

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Answer: A dedicated event list for DEC without fuel melt (beyond design basis accidents) is not available for German NPPs. But, in the "Safety Requirements for Nuclear Power Plants" dedicated lists of events are given for both transients and DBAs. These lists cover full power and shutdown states and have been derived for PWR and BWR plants as well as for the spent fuel pools of such NPPs. From these list, beyond design accident sequences shall be derived. The requirements for the selection of events for preventive actions are defined in the "Safety Requirements for Nuclear Power Plants".

"*For the determination of the representative event sequences for the planning of preventive measures of the internal accident management, the results of deterministic and probabilistic safety analyses, operating experience as well as results of reactor safety research and international recommendations shall be referred to within the framework of an overall survey. Event sequences shall be considered which according to the results of probabilistic safety analyses make a dominant contribution to the core melt frequency and especially those that may lead to a direct release of radioactive materials into the environment.*

The plant-specific spectrum of event sequences on which the planning of preventive measures of the internal accident management shall comprise at least events from the following groups of events:

- *transients,*
- *loss-of-coolant accidents inside the containment as a result of the maximum postulated leaks in the reactor coolant system,*
- *loss-of-coolant accidents with containment bypass, and*
- *external and internal hazards if these hazards can lead to multiple failures of safety equipment.*

Based on a postulated multiple failure of safety equipment, the representative event sequences to be referred to for the planning shall be defined.

For the planning of preventive measures of the internal accident management aimed at the restoration and maintenance of fuel cooling in the spent fuel pool event sequences involving the complete loss of the systems provided on levels of defense 1 to 3 for heat removal from the spent fuel pool as well as event sequences involving a loss of coolant from the spent fuel pool and a drop below the minimum level required for the operation of the systems for heat removal shall be postulated.

Regarding the event sequences mentioned above, the possibility of the complete loss of one each of the safety functions necessary for the control of events on level of defense 3 shall be analyzed when planning preventive measures of the internal accident management. Here, the failure of the required safety equipment and, on the other hand, the loss of one each of the supply functions that may be necessary for the safety equipment shall be analyzed separately."
"Safety Requirements for Nuclear Power Plants"

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer: A dedicated event list for beyond design basis accidents in spent fuel pools is not available for German NPPs. But, in the “Safety Requirements for Nuclear Power Plants” a dedicated list of events inside SFP is given for both transients and DBAs. That lists cover full power and shutdown states and have been derived for PWR and BWR plants. From these list, beyond design accident sequences for the SFP shall be derived by the postulation of additional failures of safety systems. Exemplarily, initiating DBA events of that list are:

- Loss of two trains of the spent fuel pool cooling system for a longer period (> 30 min.),
- Loss of coolant from the spent fuel pool due to leaks with a cross section > DN25 up to the largest connecting pipe,
- Leak at the refuelling cavity or setdown pool with opened refuelling slot gate,
- Internal leak in heat exchangers of the spent fuel pool carrying coolant,
- Loss of offsite power for more than 10 hours,
- Geometry changes due to external hazards (spent fuel pool, dry storage facility for non-irradiated fuel), etc.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?
- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

Answer: For beyond design accidents their deterministic analyses should be performed in a best-estimate manner regarding both boundary conditions and applied simulation tool.

For DECs without fuel melt dedicated provisions (components and/or systems, like e. g. secondary and primary side bleed & feed (PWR), diverse depressurization valves (BWR), mobile equipment (PWR and BWR), passive injection (PWR and BWR), etc.) shall be considered in the plant which are able to prevent the damage of core and/or fuel assemblies inside SFP. The progression of the BDBA into a level of defence 4c sequence should be avoided.

The provisions of accident management measures for prevention should be independent from the components/systems of other plant states (transients, DBAs) and they shall be designed being effective for a broad spectrum of events. But, suitable respective measures and equipment of level of defense 1 to 3 may also be used for accident management.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

Answer: In Germany, DECs are those plant states associated with the levels of defence in depth 4a (ATWS), 4b (multiple failure of safety systems) and 4c (severe accidents). As Germany operates only existing nuclear power plants, this are plant states beyond the design basis. The ATWS can be handled by the plants with EOPs for DBAs. Events of Level 4b and 4c are controlled by plant internal severe accident management measures (prevention and mitigation). This relies on fixed installed equipment, portable equipment, and SAMG measures. No requirements on redundancy exists for equipment provided for severe accident management (Levels 4b and 4c), as there is no demand for the application of the single failure concept for accident management measures. The dedicated components for the prevention of core damage (e. g. like valves for secondary and primary side depressurization) and instrumentation are classified like the equipment which is used for Design Basis Accidents.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer: Question not applicable for Germany.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

The following answers are only dedicated to existing German nuclear power plants as new buildings of NPPs are forbidden in Germany.

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

Answer: In Germany, accidents with severe fuel degradation are distinguished from those accidents beyond the design basis accidents until the onset of fuel melt (see also answer to question 3.2). For severe accident with core damage mitigative severe accident management measures are foreseen in German NPPs.

"*In addition to the existing EOPs, mitigative measures like e. g. filtered containment venting system, passive autocatalytic recombiners, filtered control room ventilation have already been realized in the 1990ties. Furthermore, plant-specific SAMGs have meanwhile been introduced at all German nuclear power plants for their crisis management teams as part of the National Action Plan after the Fukushima accident. The procedures and strategies contained in these manuals ("emergency operating manual" and "Handbook for mitigative SAM Measures") comply with the international recommendations on EOPs and SAMGs.* " [CNS 7. National Report]

German nuclear power plants have been back fitted several times (already started in the 1990ies) to mitigate the consequences of severe accidents, primarily to ensure the integrity of the containment as long as possible. The last optimization of the mitigative SAM measures has been realized after the Fukushima accident by the implementation of a “Severe Accident Management Guidelines (SAMG)” concept which is documented in the “Handbook of mitigative SAM Measures (HMN)” which is part of the plant documentation of all German NPPs in operation.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Answer: Relevant severe accident scenarios with core damage are determined from dedicated event lists documented in the “Safety Requirements for Nuclear Power Plants” for both transients and DBAs. These lists cover full power and shutdown states and have been derived for PWR and BWR plants as well as for the spent fuel pools of such NPPs. For selected initiating events of these lists, beyond design accident sequences with core damage should be derived by the assumption of additional multiple failures of systems / safety systems. The selection of the relevant initiating event sequences is supported by PSA Level 1 results.

“*For the design of mitigating measures of the internal accident management to cope with severe accidents on level of defence 4c, a spectrum of events shall be postulated that takes into account all relevant phenomena of accidents with severe fuel assembly damages. In this context, special attention shall be paid to those phenomena that put containment integrity and, if the spent fuel is stored in a fuel pool outside the containment, the integrity of the structure around the fuel pool at risk. Furthermore, the phenomena that have an effect on the release of radioactive materials and on possible release paths to the environment shall be considered.*”
“Safety Requirements for Nuclear Power Plants”

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer: See answer to Question 6.3. The initial events listed under Question 6.3 can also progress into a severe accident with fuel assembly melting by the postulation of further failures of components and safety systems.

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?
- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

Answer: For beyond design accidents, their deterministic analyses should be performed in a best-estimate manner regarding both boundary conditions and applied simulation tool.

For DECs with fuel melt dedicated mitigative provisions (components and/or systems, like e.g. filtered containment venting system, passive autocatalytic recombiners, and mobile equipment) are considered in the plant which are able to mitigate the damage of the containment as the last barrier against the release of radionuclides. The release of radionuclides into the environment of the BDBA into a level of defence 4c sequence should be avoided.

The dedicated provisions of accident management measures for mitigation, like filtered containment venting and passive autocatalytic recombiners should be independent from the components/systems of other plant states (transients, DBAs) and they shall be designed being effective for a broad spectrum of events. But, suitable respective measures and equipment of level of defence 1 to 3 may also be used for SAMG. This is especially part of the SAMG concept.

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Answer: In Germany, DECs are those plant states associated with the levels of defence in depth 4a (ATWS), 4b (multiple failure of safety systems without core damage) and 4c (severe accidents with core damage). For Germany only existing nuclear power plants have to be considered. Thus, plant states under consideration here are beyond the design basis accidents. The ATWS can be handled by the plants with EOPs for DBAs. Events of Level 4b and 4c are controlled by plant internal severe accident management measures (prevention and mitigation). This relies on fixed installed equipment, portable equipment, and SAMG measures. No requirements on redundancy exists for equipment provided for severe accident management (Levels 4b and 4c), as there is no demand for the application of the single failure concept for accident management measures.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer: Question not applicable for Germany.

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Answer: As one of the technical safety concepts the concept of multi-level confinement of the radioactive inventory (barrier concept) is required. For each level of defence in depth specific requirements exists for the barriers and confinement functions. As the question is related to accident conditions, the answer is provided of levels of defence in depth 3 (design basis accidents), 4a (ATWS), 4b (multiple failure of safety systems) and 4c (severe accidents).

Level of defence-in-depth 3:

"*For the confinement of the radioactive materials in the reactor core:*

1. *the fuel rod cladding, unless their failure is postulated as initiating event and not in event of a large-break loss-of-coolant accident,*
2. *the reactor coolant pressure boundary, unless the reactor coolant system has been opened intentionally and its failure is postulated as initiating event,*
3. *the containment, unless it has been opened intentionally. If the containment has been opened intentionally, it shall be ensured that the barrier function of the containment can be restored in due time to the necessary extent or that effective and reliable retention functions are available so that an inadmissible release of radioactive materials is prevented or stopped in time.*

for the handling and storage of fuel assemblies:

1. *the fuel rod cladding, not considering event-specific postulated cladding failures) as well as*
2. *the containment, unless it has been opened intentionally. If the containment has been opened intentionally, it shall be ensured that the barrier function of the containment can be restored in due time to the necessary extent in the case of events involving releases of radioactive materials within the containment. If spent fuel assemblies are handled or stored outside of the containment, the lack of this barrier shall be compensated by retention functions. The achievement of the radiological safety objectives with regard*

to radioactive materials elsewhere in the plant shall be ensured in all operational modes by retention functions." [Safety Requirements for Nuclear Power Plants]

Level of defence-in-depth 4a:

1. "*the fuel rod cladding to the extent necessary for achieving the applicable acceptance targets,*
2. *the reactor coolant pressure boundary,*
3. *the containment.*" [Safety Requirements for Nuclear Power Plants]

Level of defence-in-depth 4b:

"*At least one of the still existing barriers shall be maintained by the planned measures of the internal accident management to achieve the radiological safety objectives to exclude large and early releases. For the confinement of the radioactive materials in spent, stored fuel assemblies, the integrity of at least one barrier shall be ensured on level of defence 4b. If spent fuel assemblies are handled or stored outside of the containment, the lack of this barrier shall be compensated by retention functions.*" [Safety Requirements for Nuclear Power Plants]

Level of defence-in-depth 4c:

"*Mitigative measures of the internal accident management shall be provided for the purpose of maintaining - by using all available measures and equipment - the integrity of the containment for as long as possible, excluding or limiting releases of radioactive materials into the environment to exclude large and early releases, and achieving a long-term controllable plant state.*" "Safety Requirements for Nuclear Power Plants"

If spent fuel is stored in the spent fuel pool outside the containment, mitigative measures of the internal accident management shall be provided for the purpose of maintaining - by using all available measures and equipment - the integrity of the surrounding structural cover for as long as possible, excluding or limiting releases of radioactive materials into the environment according to subsection 2.5 (1), and achieving a long-term controllable plant state.

In addition, specific requirements for the containment regarding the confinement function during accidents (DBAs and BDBAs) can be found in the German "Safety Requirements for Nuclear Power Plants".

"*The nuclear power plant shall have a containment system consisting of the containment and the surrounding building as well as of the auxiliary systems for the retention and filtering of any possible leakages from the containment. The containment system shall fulfil the retention function such that the release of radioactive materials into the environment is kept as low as possible and the limits specified for levels of defence 1 to 3 are not exceeded.*

Under the operating conditions in which it is closed according to schedule, the containment shall fulfil its safety functions on levels of defence 1 to 3 as well as during transients involving the failure of reactor scram (level of defence 4a) and in the event of internal and external hazards as well as under very rare human- induced external hazards.

Reliable, sufficiently fast and adequately long-lasting isolation of the containment penetrations shall be ensured. The required leak-tightness for the containment shall be quantified by a maximum permissible leak rate for the operational modes in which the containment is closed.

For accidents involving severe fuel assembly damages (level of defence 4c), the following shall apply additional to the requirements of subsection 2.1(3b):

- *It shall be ensured by internal accident management measures that there will be no overpressure failure of the containment due to a steady pressure increase. If containment venting is provided as an intentional accident management measure, this shall be effective under the expected severe accident conditions and shall provide efficient filters for aerosol and iodine retention. Containment failure due to negative pressure as a result of venting shall be avoided.*
- *In the event of an accident involving severe fuel assembly damages, it shall be achieved by internal accident management measures that there will be no deflagration processes of gases (H_2, CO) inside the containment that will put containment integrity at risk.*
- *In the event of an accident involving severe fuel assembly damages in the spent fuel pool, it shall be achieved by internal accident management measures that there will be no combustion processes of gases (H_2,) that will put containment integrity or the integrity of the structure surrounding the spent fuel pool at risk.*" [Safety Requirements for Nuclear Power Plants]

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

Answer: The design pressure and temperature of the PWR containment has been determined from the envelope load/bounding case scenario “Large Break LOCA” plus the evaporation of the inventory of the secondary side of one steam generator and the inventory of its main steam line up to the containment isolation, which is initiated by a postulated leak in the main steam line inside the containment. Further main requirements had to be considered for the design of PWR containment:

- Normal power state of the plant,
- decay heat equals to 1.2 times of the ANS standard,
- volume of the primary circuit was enlarged by 2 %,
- free Volume of the containment was shortened by 2%,
- stored heat of the solid structures of the primary circuit has been added to the containment atmosphere,
- additional charge of max 0.3 bar for a design pressure up to 7 bar in order to consider thermodynamically disequilibrium of the two phases (water and steam),
- energy from metal-water reaction considered in the containment atmosphere,
- Design pressure is at least 1.15 times of the maximal pressure resulting from the worst LOCA event.

To maintain the integrity of the PWR containment during severe accident sequences with core damage a filtered containment venting system has been installed which is operated at the design pressure of the containment at the latest.

For the BWR, the design pressure of the containment has been determined for the maximal pressure resulting from the spectrum of the LOCAs to be considered for the plant. The design pressure of the BWR containment is determined in combination with the necessary suppression system of the plant. Following main requirements has been considered for the design of the BWR containment in addition:

- Normal power state of the plant,
- decay heat equals to 1.2 times of the ANS standard,
- volume of the reactor circuit was enlarged by 2 %,

- free Volume of the containment was shortened by 2%,
- stored heat of the solid structures of the reactor circuit has been added to the containment atmosphere,
- volume pressure chamber increased by 2%, volume of suppression chamber decreased by 2%,
- energy from metal-water reaction considered in the containment atmosphere,
- design pressure is at least 1.15 times of the maximal pressure resulting from the worst LOCA event.

To maintain the integrity of the BWR containment during severe accident sequences with core damage a filtered containment venting system has been installed which is operated at the design pressure of the containment at the latest.

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

Answer: The containment penetrations shall be realized by a dedicated design appropriate to minimize the total leakage of the containment. A leak rate of the containment has been fixed (typical value 0,25 vol%/d) and approved during the design of the plants. The leak rate of the containment (including penetrations, isolation valves, hatches, etc.) has been determined at the design pressure of the containment for both PWR and BWR plants respectively. The leak tightness of each containment had to be shown before the commissioning of the plants by an integral measurement. In addition, the integral leak flow has to be checked at the end of every in-service inspection with an overpressure of at least 0.5 bar. Furthermore, periodical pressure tests (each 4 years) has to be done (nuclear standard KTA 3401.4).

After Fukushima, the hydrogen treatment outside the containment during severe accident sequences with core damage has been discussed in the German Reactor Safety Commission (RSK). A note issued by RSK contains requirements for both PWR and BWR reactor type to handle hydrogen outside the containment. The goal of those measures is to avoid strong hydrogen explosions and detonation.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Answer: In Germany, there is a definition of practical elimination**:** "*The occurrence of an event or event sequence or a state can be considered as excluded*

- *if it is physically impossible to occur or*
- *if it can be considered with a high degree of confidence to be extremely unlikely to arise.*" [Safety Requirements for Nuclear Power Plants]

Question 9.2: Do you have requirements related to practical elimination?

Answer: The very low probability of such events/states have to be shown by the utility of the plant and finally to be approved by its regulator. The very low probability can also be achieved by both dedicated technical equipment as well as organizational measures realized in the plant.

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?

  Answer: Following examples for DBAs are given, which have been taken from the event lists of the German nuclear regulatory framework:

  PWR: Avoiding the faulty injection of deionat into rector circuit by organizational measures like locked valves.

  PWR: Avoiding safety relevant flooding in the reactor building annulus by separated compartments in the annulus.

  BWR: Avoiding loss of tightness between drywell and wetwell by taking pre-cautionary measures so that impermissible leaks between drywell and wetwell, in particular during restart of the plant and after maintenance measures can be prevented.

  SFP: Avoiding re-criticality by a dedicated design of the racks (appropriate pitches of the fuel assemblies to avoid re-criticality), absorber material enclosed in the racks, borating of the water inside spent fuel pools.

SFP: Avoiding leaks at the SFP in the lower part of the pool by avoiding connections of pipes and an appropriate design against earthquakes.

- For new power reactors?

  Answer: Not applicable for Germany

- For existing research reactors?

  Answer: German research reactors have a much lower radioactive inventory compared to NPPs. For research reactors, the implementation of the concept of practical elimination is applied to meet the radiological safety objective to avoid large or early releases. For example, activation of the containment isolation by the reactor safety system is an implemented measure to practically eliminate a large release of radioactivity from a research reactor. As no pressure increase within the containment will occur, no additional measures like containment venting, is necessary. Another example is checking the pool water level in the long term and, if necessary, a make-up can be supplied via operational systems by accident management measures.

- For new research reactors?

  Answer: Currently, no new research reactors are planned in Germany.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

Answer: To avoid early releases a main issue is the by a high-quality design of the barriers like reactor circuit and containment and appropriate equipment (depressurization of reactor circuit, passive autocatalytic recombiners to avoid H2/CO explosions/detonations, containment venting etc.) to protect the barriers against their early failure, leading to relevant grace periods.

The avoidance of large releases can be managed by a high-quality design of the barriers like reactor circuit and containment and appropriate equipment for the protection of the containment, like passive autocatalytic recombiners, filtered containment venting, staggering of low pressure levels in the containment, etc.).

For research reactors see the answer to the previous question.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

Answer: The demonstration of the practical elimination of accident sequences, which has to be done by the vendor/utility of a plant, is a combination of deterministic and probabilistic methods. The deterministic analyses shall be done to make a safety assessment of systems/components which are used to make an accidental sequence very unlikely or physically impossible. If necessary, the assessment of the systems/components shall be supported by deterministic event analyses to show that an initiating incident does not progress into such an accident sequence which should be avoided or be made very unlikely. The probabilistic method shall be applied for the quantification of the probability of such a sequence to get a feeling if the sequence can be treated as practically eliminated. The results can be used as a basis for the decision of the regulator if the measures of the utility can be approved for the elimination of the dedicated accident sequence.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Answer: Yes, it is. See examples for spent fuel storage pools under Question 9.3.

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Answer: Since the amendment of the Atomic Energy Act of April 2001, the performance of PSR every ten years has been mandatory, with the date of the first safety review laid down for every installation. The obligation to present the safety review results is lifted if the licence holder makes the binding declaration to the licensing and supervisory authority that he is definitively going to terminate power operation at the installation no later than three years after the final date for submission of the safety review mentioned in the Atomic Energy Act.

A focal point for the deterministic safety status analysis is the consideration of the accidents compiled in Appendix A of the guideline for the safety status analysis, the deterministic safety status analysis and a spectrum of beyond-design-basis plant conditions for which the existence of accident management measures has to be shown.

For nuclear installations in the post-operational phase, the federal authority and the regulatory authorities decided that the licence holder has to perform a safety analysis for this phase. Corresponding details are specified in a “Checklist for the performance of an assessment of the safety status of the installation for the post-operational phase”.

The scope of the periodic safety Review is clearly defined in the “Guideline to perform Periodic Safety Reviews for Nuclear Power Plants”.

For the results achieved so far it can be stated that on the basis of the analyses performed, it was demonstrated that the German nuclear installations fulfil the safety requirements that are necessary for compliance with the protection goals, referred to as “fundamental safety functions” in the IAEA safety standards.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

Answer: The details are laid down in German guideline for periodic safety reviews "Guides for the Periodic Safety Review of Nuclear Power Plants" that is currently under revision. This Guideline implicitly includes the assessment of the 14 safety factors as required by WENRA RL P2.2.

The deterministic safety status analysis is performed according to the procedure stated in the guide "Safety Status Analysis". According to this guide, the procedure is provided as follows:

- a deterministic review of engineered safety features of the plant according to the protection goal-oriented requirements of nuclear regulations and the accident spectrum to be considered,
- description of the equipment and measures for special, very rare events as well as of the severe accident management concept,
- description of operational management and evaluation of operating experience.

The description of operational management and evaluation of operating experience is an additional essential part of the safety status analysis. Essential subjects are: technical knowledge and operational organisation, periodic testing and in-service inspections, maintenance, experience feedback, radiation protection and emergency preparedness. The evaluation of operating experience is to concentrate on safety aspects of normal operation, anticipated operational occurrences and accidents to be considered. The results are to be judged with regard to operating experience important to safety of plant equipment. Assessment of the results is to demonstrate whether the systems engineering requirements stated for levels 1 and 2 of the safety concept are met, and whether reliable operation with regard to accident prevention is ensured and can also be expected for the future.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

Answer: In Germany, the licensee is obliged to respect the state of the art in science and technology. This includes the recent developments in international standards – which present the current state of the art in science and technology - in his analyses and, consequently, to identify and determine safety improvements. Safety reviews are required by the “Safety Requirements” and described in the “Guideline for PSR”. These are based on the Atomic Energy Act that requires the precaution according to the state of the art in science and technology. The “Safety Requirements for Nuclear Power Plants” contain fundamental and general safety-related requirements within the framework of the non-mandatory safety standards and rules that serve for substantiating the precaution that pursuant to § 7 para. 2 no. 3 of the Atomic Energy Act is necessary according to the state of the art in science and technology to prevent any damage caused by the construction and operation of the plant as well as the requirements of § 7d of the Atomic Energy Act Regarding the nuclear power plants operated in Germany, this concerns modification licences.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Answer: The German "Safety Requirements for Nuclear Power Plants" requires the company management of the licensee to develop, introduce and continually improve an integrated, process-oriented management system (IMS) and to define and implement the company policy and business objectives in which the company commits itself to a high level of safety and to a strengthening of the safety culture. One of the prime objectives of the IMS is the promotion of safety culture.

Safety Standard KTA 1402 "Integrated Management System for the Safe Operation of Nuclear Power Plants" further elaborates that safety culture is furthered and supported by the management system in that it is required of this system that it

a) "*supports a mutual understanding of the key aspects of safety culture within the corporation,*

b) *provides the means with which the corporation supports individuals and groups to perform their tasks safely and successfully taking the interactions between man, technology and organizational aspects into account,*

c) *strengthens the basic attitude for learning and questioning at all levels of the corporation, and*

d) *provides the means by which the corporation continuously supports the development and improvement of safety culture.*" [KTA 1402]

According to KTA 1402, the plant management shall among other things conduct in regular intervals a self-assessment of the safety culture and an independent assessment of the safety culture and implement improvement measures, in order to maintain a high safety culture and to continuously improve it.

Specifically, it is stated in KTA 1402 that plant management and executives shall normally perform process-independent assessments of the management system on the basis of, e.g., plant walk-throughs and work observations, in order to identify improvement possibilities regarding work behaviour and safety culture. Self-assessments and independent assessments of the safety culture for the entire plant shall be conducted regularly under the participation of technical experts. The results of the assessments shall normally be made accessible within the organization of the plant for all employees on all levels. For the assessment of the safety culture it is recommended to combine quantitative (such as employee surveys, indicators, etc.) and qualitative (such as interviews, observations, etc.) methods. The goal for the use of several methods for the evaluation of the safety culture is to balance strengths vs. weaknesses, so that the strengths of one method may offset the weaknesses of the other methods. In this context, plant management should also initiate carrying out national and international reviews.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

Answer: According to Safety Standard KTA 1402 “Integrated Management System for the Safe Operation of Nuclear Power Plants”, plant-internal and plant-external experience shall be used to improve safe operation through learning, training and implementing procedures. A systematic exchange of information of safety-related operating experience shall be organized. The distribution of information, information feedback, and how information is to be processed and documented shall be specified. In this context, the plant-internal experience and the experience of other power plant operators shall be assessed and communicated. The plant-internal and plant-external operating experience shall be assessed from a technical standpoint by the particular organizational unit in charge and, from a general standpoint, by an independent organizational unit. New scientific and technological findings shall be considered and evaluated within the framework of experience feedback.

KTA 1402 further elaborates that regarding the plant-internal experience feedback, operational procedures shall be installed to ensure a consistent mutual exchange of information with respect to operational events. These operational procedures shall specify the ways in which electronic, written or verbal information can be used for this exchange. The plant-internal feedback of experience involves, e.g., the failure alarm procedure, daily work conferences as well as procedures for passing on information about almost-events. Further processing of these input data shall be specified in accordance with their safety relevance. Cause analyses shall be performed taking all aspects of man, technology, organization as well as their interaction into account. The procedures for performing these analyses shall be specified. In this context, criteria shall be specified regarding the participation of independent organizational units and regarding a further in-depth analysis. In-depth analyses shall be performed by an independent organizational unit. Modification measures shall be specified on the basis of the analysis results. The implementation of these measures shall be surveilled and their effectiveness evaluated through suitable methods. The causes determined from these analyses shall be subjected to a trend analysis in order to enable an early detection and prevention of a possibly occurring accumulation of these causes.

KTA 1402 goes on that with regard to external experience feedback, the experience and findings made by other power plant operators and other institutions (e.g., authorities, authorized experts, manufacturers) shall be taken into consideration and the transferability for the own power plant evaluated. In this context, the following source of information shall be utilized:

a) "*reportable events,*

b) *operation experience as reported in national and international information systems,*

c) *exchange of the plant-internal experience with other power plants,*

d) *exchange of information in working groups,*

e) *exchange of information with manufacturers.*" [KTA 1402]

In case the experience is transferrable to the own power plant, the measures required shall be specified and their implementation closely followed.

Appendix 6: Answers to the questionnaire - Hungary

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

RevisionR2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name:

Organization:

Position:

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

In general, the safety approach is defined by the legal system and is much wider than to cover only issues contained in the 8a – 8c articles of the NSD (Nuclear Safety Directive). On the highest level, it is defined in the Act on atomic energy (Act CXVI of 1996, modified and supplemented numerous times). The Act regulates not only the nuclear facilities containing reactors, but all uses of the atomic energy, and even for these facilities the approach to maintaining safety covers other articles of the NSD as well.

A substantial part of this philosophy is stated in the "Basic principles" section of the Act. It is worthwhile to mention from these the followings (not all are exact quotations):

- *Atomic energy shall be utilized the way that it does not harm the life of humans, health and living conditions of present and future generations, the environment and material goods beyond the risk level acceptable, and tolerable along all other economic activities, by the society;*
- *When utilizing atomic energy safety has preference compared with any other viewpoints;*
- *Along utilization of atomic energy un-regulated and un-controlled chain reaction shall be excluded and the professional and public doses from all sources shall not exceed the dose limit set by the related safety regulation taking into account the newest validated results of the science and guidance from the international and national expert organizations, complying with the ALARA principle and the releases shall be regulated accordingly;*
- *The extraordinary events shall be preventable, the consequences shall be mitigated according to elaborated plans and the harmful effects of escaping radioactive materials and radiation shall be decreased to lowest rationally achievable level;*
- *User of atomic energy shall maintain that radwaste stemming from the activity be ALARA;*
- *Safe use of atomic energy, including response to emergencies, and solution of related research and development tasks shall be promoted by developing science and technic,*

harmonized organization of research, practical implementation of results of domestic and international scientific research, training and retraining of specialists;

Furthermore the fundamental safety principle and (with small modifications) the 10 safety principles from the IAEA SF-1 are declared by the Act, establishing the Hungarian safety philosophy within the legal framework.

Additional basic principles are stipulated to require

- *regular overview and modernisation of nuclear safety requirements based on scientific results and international experience,*
- *utilization of atomic energy exclusively according to law and under authority control, while conditions are determined by authorities continuously taking into account of law, science and technical results,*
- *independence of authorities involved in utilization.*

The Act excludes altering or nullifying authority decisions by any supervising organization and sets responsibility for safe use of atomic energy and adherence to safety requirements on utilizing person.

The Government decree (118/2011 amended several times) specifies the detailed safety requirements for nuclear facilities and also the related authority procedures (licensing, approval, inspection, assessment and enforcement). They also cover some additional general safety approach issues, such as

- *the licensee is obliged to demonstrate for the authority complete adherence to the prescribed requirements;*
- *definition of 3 safety goals:*
  - *the general nuclear safety goal (defence against ionizing radiation by realization and adequate maintenance of effective safety measures),*
  - *the radiation protection goal (radiation doses for professionals and population being ALARA and always below the prescribed limits in all phases of the life-cycle of the nuclear facility, what shall be maintained for events belonging to design basis conditions and – as much as rationally practicable – for events belonging to design extension conditions),*
  - *the technical safety goal (occurrence of incidents shall be avertable or precluded by high reliability, possible consequences shall be within the acceptable margins for all postulated initiating events taken into account in*

design of the nuclear facility, probability of the accidents shall be as low as required);

- *the above safety goals shall be effective in all phases of the life-cycle, such as design, siting, fabrication, construction, commissioning, operation and decommissioning;*
- *formulation of the defence in depth principle and its 5 levels, together with goals vested in it;*
- *requirement on elaboration and implementation of a safety policy by the licensee listing the main required characteristics of it;*
- *requirement of continuous improvement of the safety level (including modifications for any reason) taking into account the internal and external experience.*

The Government decree formulates as well the contents of the procedures of the safety authority, including the following policy related elements:

- *continuous authority supervision – including periodic safety reviews,*
- *issuance of prescriptions in case, when the risk is more than the value taken into account earlier,*
- *formulation of the authority decisions based on the overall and detailed evaluation of facts from point of view of the nuclear safety of the facility as a whole, taking into account all available relevant information.*

In addition, there exists an internal policy document of the Hungarian safety authority, titled "Safety policy of the HAEA and its authority code of conduct". From this document, it is worthwhile mentioning as well some moments relevant to the NSD, and specifically to the articles 8a to 8c, and which are complementary to above elements originating from legal documents (and many of which are repeated in this Safety policy as well). These are listed below:

- *Fundamental goal of our authority is that the population, the environment and the utilizer of the atomic energy shall not suffer harm from the dangers caused by nuclear installations;*
- *The guarantee for achieving this goal is that the licensees completely execute their duties deriving from their responsibility. The authority accomplishes its supervisory activities primarily based on technical facts and considerations …, using the means of <u>prevention</u> to hinder the deviation from rules and requirements, <u>detecting the deviations</u>*

as early as possible, and prescription of countermeasures to eliminate the materialized deviations;

- *the authority accomplishes its activities to promote prevention of deviations and problems, and should they occur, recognition and elimination of those;*
- *the principles followed by the authority accomplishing these activities include:*
    - *principle of the substantiation (comparison of risks and benefits),*
    - *principle of international co-operation,*
    - *principle of coherence and transparency,*
    - *principle of utilization of proven technical solutions,*
    - *principle to follow the scientific and technical progress,*
    - *principle of formation of safety culture;*
- *when formulating proposals for juridical acts or safety requirements as well as authority decisions clear, unambiguous texts shall be used, but the prescriptive character should be avoided, expediently prescribing "what" to be done, instead "how" to be done.*

Practical implementation of this approach is complex. It is materialised by continuous efforts for adequate staffing of the nuclear safety authority (HAEA), training of its personnel, fulfilling all tasks of the authority (authorization, inspections [planned comprehensive, for specific predefined activities and reactive, both announced and un-announced], analysis and evaluation of the licensees' safety performance, enforcement of requirements, regular review and renewal of safety requirements and of safety guidance documents, reporting to the government, parliament and the public, distribution information for the public, international co-operation including wide variety of peer reviews, etc. Activities of the HAEA are governed by a certified management system, with all required attributes of that.

The safety requirements are differentiated for the existing and planned NPP units, at the same time requiring through periodic safety reviews the implementation of the best international safety practices for the sake of the continuous safety improvement.

It should be stated as well that Hungary took obligation to adopt the WENRA safety reference levels. In this relation, the actual standing of this work was peer-reviewed by the WENRA RHWG; the results of the peer-review could be find at the URL http://www.wenra.org/archives/wenra-statement-implementation-its-2014-safety-ref/. Here is an action plan as well to cover the missing items, what was fully implemented in April 2018, by amendments to the Code. Furthermore it is worthwhile to mention, that the IAEA IRRS mission in 2015 stated as well, that the Hungarian regulatory system is basically in line with the IAEA Safety standards see the report at URL:

http://www.oah.hu/web/v3/HAEAportal.nsf/03EF76109A67E9C5C1257EE400194F93/$FILE/Final%20IRRS%20Report%20Hungary.pdf. Later the follow-up mission in 2018 concluded, that Hungary made substantial progress in the implementation of recommendations and suggestions.

As it could be seen from the above policy elements, our general approach – in a broad sense – contributes to complying (among others) to following parts of the NSD:

- *Introductory text of paragraph 1. of Article 8a is covered by the above outlined "Basic principles" of the Act and the first 3 bullets of additional issues related general safety approach taken from the 118/2011 Governmental decree; while points (a) and (b) are not specifically covered on policy level, instead they are covered by safety requirements in the Nuclear Safety Code – further the Code – (11 volumes constituting Attachments to the Governmental decree).*
- *Paragraph 2. of Article 8a is covered in general at policy level, but differentiation between new and existing nuclear facilities given by the two dedicated volumes of the Code – one containing the design requirements for the existing NPPs (Volume 3) and the other for the new ones (Volume 3a).*
- *The points (a) to (e) of paragraph 1. of Article 8b are basically covered by the "Basic principles" of the Act, however the detailed requirements of the protection against natural and man-made hazards, as well as the specific requirements about the five levels of defence in depth are given in the Governmental decree, especially in the design related volumes of the Code. The point (f) is covered in volume 2 of the Code (Management system of nuclear facilities) and in volumes containing requirements to operation of these facilities.*
- *The introductory text of paragraph 2. of Article 8b is covered in general at policy level (see above), while the specific requirements – points (a) to (d) of this paragraph – are covered in Volume 2 of the Code.*
- *The basic principles in relation to Article 8c are covered in the Act by requiring the fulfilment of all safety requirements. The detailed requirements for demonstrating the safety in order to obtaining construction and operational licenses, as well as the requirements for periodic safety reassessment (including the content, criteria, methodology, time-frame, etc.) are set in the Volumes 1, 3, 3a and 5 of the Code.*

3 Safety objectives for new reactors and for existing ones

General information: *Hungary has committed to a new NPP construction project in 2014, with the objective to build two VVER-1200 type plants next to the existing units at Paks site. The new NPP project is currently in its early phase, so far only the site license has been issued, thus availability of technical and design information for the regulator is limited. In line with that, information on how the requirements of Directive 2014/87/Euratom are to be implemented in practice is not currently available. Because of this, information presented below regarding the new units is based on the legally binding regulatory requirements that the license applicant has to comply with, and duly demonstrate in the construction licensing application.*

There are two research reactors in Hungary. The licensed power rate of the training reactor of the Budapest Technical and Economy University is 100 kW_{th}, and therefore it is out of scope of this questionnaire, while to the Budapest Research Reactor with its nominal thermal power rate of 10 MW is given due consideration in following answers.

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

Detailed, technical level requirements for nuclear safety are set forth in Government Decree No. 118/2011 (VII. 11.) "On the nuclear safety requirements of nuclear facilities and on related regulatory activities" and in its annex, the Nuclear Safety Code (NSC). In this, design requirements for newly constructed NPPs are stricter than for existing NPPs. Some of the reasons for this approach was that:

  - *it represented a significant challenge to create design and assessment requirements, that are fully applicable for existing and new units at the same time without being either un-implementable (too strict) or too permissive;*
  - *the new units will have a design life of at least 60 years, so in order to meet future challenges all involved parties agreed to create requirements that are reasonably stricter ("future-proofing");*

- *the scope of design and analysis activities are different in case of an existing unit and a new unit (plant level initial design of new NPPs vs. modification of an existing plant/technology);*
- *the design/assessment codes and standards have significantly evolved (Gen II vs. Gen III+);*
- *some situations that are considered as "beyond design" for existing plants, are considered in the design of new plants (e.g. core melt) as design extension conditions;*
- *etc.*

In terms of quantitative requirements, the differences – among others – are summarized in:

- *Table 1 shows the probabilistic limiting values for screening out different initiating event types and hazards that have to be considered in the design basis of the NPPs; and*
- *Table 2 shows the probabilistic acceptance criteria for CDF and LERF for different NPPs.*

Plant type	*Screening limits of initiating events and hazards to be considered within the design basis*		
	Internal Initiating Events and Internal Hazards [1/year]	*Human induced External Hazards [1/year]*	*Natural Hazards [1/year]*
New NPP	10^{-6}	10^{-7}	10^{-5}
Existing NPP	10^{-5}	10^{-7}	10^{-4}

Table 1.: Screening limits for initiating events and hazards to be considered within the design

Plant type	*Acceptance criteria*	
	CDF [1/year]	*LERF* [1/year]*
New NPP	*< 10^{-5}*	*< 10^{-6}*
Existing NPP	*< 10^{-4}*	*< 10^{-5}***

**Cumulated for all plant conditions (DBC and DEC - taking into account internal initiating events, internal and external hazards) with exclusion of malicious acts and of earthquakes*

***With implementation of all reasonable modifications and interventions advances shall be made towards the value of 10^{-6}/ry*

Table 2.: Probabilistic acceptance criteria for NPPs

It should be noted that the Hungarian requirements use the term Large OR Early Release (LER) and Large OR Early Release Frequency (LERF), which is a bit different than Large Release and Large Early Release terms that are most commonly used in the international practice. The exact definition of the term Large OR Early Release is provided in the answer of question 3.4.

The practical result of this differentiation can be shown through the following example:
The existing and new NPP units will be located at the same location (Paks) as neighbouring sites. In terms of the Design Basis Earthquake values, the existing units are designed for 0,25 g (PGA), but the new units will be designed for 0,35 g (PGA).

In addition, the NSC has requirements aimed to ensure, that safety objectives are met during the design and construction phases. NSC Volume 9 has specific management system requirements for the following areas (among others):

- *Quality management of design activities:*
  - *Rules for the development of a quality management program for design;*
  - *Configuration- and change management;*
  - *Use of codes and standards;*
  - *Incorporation of construction and operational experience;*
  - *Design check and review;*
  - *Design adaptation;*
  - *Independent third party design check and review;*
- *Supply chain management:*

  - *"Intelligent customer" capability of the licensee;*
  - *Management system requirements for the suppliers*
- *Quality management of construction and manufacturing activities:*
  - *Rules for the development of a quality management program for construction and manufacturing;*
  - *Detailed design requirements;*
  - *Management of design changes and nonconformities during the construction phase;*
  - *Use of independent third parties for construction and manufacturing supervision;*
  - *Storage and conservation;*
  - *etc.*

This question is not applicable to the research reactors, as no new research reactors are foreseen, thus no specific requirements are elaborated.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

There are formulated objectives to the 5 levels of DiD, but these are not different for the existing and the new NPP units to be constructed. At the same time, the associated plant states (operational states and accident conditions) are formulated differently for existing and new NPP units (see this and the next answer below).

Table 3. contains the DiD levels for existing NPP units, the associated objectives, plant state definitions, their abbreviations in relation to the design basis and the radiation acceptance criteria set by the Code:

Level of DiD	***Objective***	***Associated plant condition/accident categories***	***Associated design category***	***Radiological acceptance criteria***
1	*Prevention of abnormal operation and failures*	*Normal operation*	*DBC1*	*Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year)*

2	*Control of abnormal operation and detection of failures*	*Anticipated operational occurrences*	*DBC2*	*Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year) Initiating events resulting in shall not cause doses exceeding 1 mSv/event outside the controlled area of the nuclear power plant, within its operational areas authorized for human staying*
3	*Control of accidents within the design basis*	*Design basis Accidents*	*DBC4*	*1., Dose for the reference group of the population is not exceeded 5 mSv/event 2., Initiating events resulting in DBC4 may not cause a dose exceeding 10 mSv effective dose or 100 mGy dose for the thyroid outside the controlled area of the nuclear power plant ~~and~~ within its operational areas authorized for human presence*
4	*Control of severe conditions, including prevention of accident progression and mitigation of consequences of severe accidents*	*Complex accidents*	*DEC1*	*In the case of DEC1, radioactive releases shall be minimized to the extent reasonably achievable.*
		Severe accidents	*DEC2*	*In the case of DEC2 the release of radioactive materials shall be limited both in time and quantity (exact definition is provided further bellow within the answer of question 3.4)*
5	*Mitigation of radiological consequences of significant releases of radioactive materials*	*Very severe accidents*		*Shall be practically eliminated*

Table 3.: Defense-in-depth levels, the associated safety objectives and radiological requirements for existing units

Additional technical requirements are given in the Code as follows below. These are typically not numerical criteria, but result oriented, requiring adequate justification of the design by analysis:

- *The design shall confirm by deterministic safety analyses that the initiating events resulting in DBC2 operating conditions will not lead to the loss of function of any barrier even with the assumption of a single failure.*
- *The integrity of the reactor pressure vessel against brittle fracture shall be maintained by ensuring that the realistic actual transition temperature of the critical elements of the pressure vessel is less than its maximal critical transition temperature derived from proper analysis of initiating events resulting DBC1 to 4 conditions i.e. discontinuities of*

material in the structure may not increase during scenarios resulting in DBC2-4 operating conditions.

- *Following initiating events resulting in DBC2 to 4 plant states the control and safety devices controlling reactivity, the nuclear fuel assemblies, as well as the structural elements of the nuclear reactor shall not be damaged or deformed to such extent that the movement of control and safety devices to terminate the fission chain reaction becomes impossible.*

- *Following initiating events resulting in DBC2 to 4 plant states the nuclear fuel assemblies, the primary circuit of the nuclear reactor, and the connected systems shall remain in such a condition that the short- and long-term cooling and management of the irradiated nuclear fuel can be ensured, furthermore the systems necessary for heat removal shall be able to perform their function both on short- and long-term.*

- *For events resulting in DBC2 state the criteria ensuring the integrity of the fuel rods shall be defined during design by defining limits for the temperature of the nuclear fuel, the critical heat flux and the temperature of the cladding. For DBC4 design basis accidents to fulfil criteria for long-term cooling and fuel management the acceptable maximum degree and type of fuel damage shall be determined.*

- *To perform a safety function, criteria for maximum pressure, maximum and minimum temperatures, thermal and pressure transients, degradation and stresses depending on the temperature range shall be defined for systems, structures and components confining radioactive releases or performing retaining physical barrier functions during their whole lifetime.*

- *To fulfil nuclear safety requirements criteria shall be defined for the temperature, pressure and leakage rate of the containment throughout its lifetime.*

As it was stated above, the 5 levels of the DiD and the associated safety goals are general for all nuclear installations. However, in case of research reactors the DiD levels are not combined with facility states, and even the terms of DEC (as well as the former BDBA) are not introduced as mandatory requirements. Definition of operational and accident states is left to the designer, requiring due substantiation and approval by the regulator.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

Table 4 below contains the DiD levels for new NPP units the different, the associated safety objectives, plant condition category definitions, its short abbreviations in relation to the design basis and the radiation acceptance criteria set by the Code. The differences may be evaluated by comparing the contents of Tables 3 and 4.

Level of DiD		Objective	Associated plant condition/accident categories	Associated design category	Radiological acceptance criteria
1		*Prevention of deviations from the normal operating conditions and prevention of failures*	*Normal operation*	*DBC1*	*Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year)*
2		*Management of deviations from the normal operating conditions and failures*	*Anticipated operational occurrences*	*DBC2*	*Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year) Initiating events resulting in DBC2 operating conditions shall not cause doses exceeding 1 mSv/event outside the controlled area of the nuclear power plant, in operational areas authorized for human staying*
3	*3.a.*	*Management of design basis accidents in order to limit radioactive releases and to prevent fuel melting*	*Low occurrence frequency DBA*	*DBC3*	*Dose for the reference group of the population is not exceeded 1 mSv/event and for initiating events resulting in DBC4 operating conditions, and may not cause a dose exceeding 10 mSv effective dose or 100 mGy dose for the thyroid outside the controlled area of the nuclear power plant and in operational areas authorized for human presence*
			Very low occurrence frequency DBA	*DBC4*	*Dose for the reference group of the population is not exceeded 5 mSv/event for initiating events resulting in DBC4 operating conditions, and may not cause a dose exceeding 10 mSv effective dose or 100 mGy dose for the thyroid outside the controlled area of the nuclear power plant and in operational areas authorized for human presence*
	3.b.		Complex accidents	DEC1	In the case of DEC1 operating conditions, radioactive releases shall be minimized to the extent reasonably achievable.

4	Management of accidents involving a fuel melting in order to limit offsite releases	Severe accidents	DEC2	In the case of DEC2 operating conditions, the release of radioactive materials shall be limited both in time and quantity (exact definition is provided further bellow within the answer of question 3.4):
5	Mitigation of radiological consequences of significant releases of radioactive materials	Very severe accidents	--	Shall be practically eliminated,

Table 4.: Defense-in-depth levels, the associated safety objectives and radiological requirements for new units

As no new research reactors are foreseen, this question is not relevant for the research reactors in Hungary.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

The DB level external hazards are differentiated on the basis of their type of origin (natural or human induced) but no such differentiation is made for external events exceeding the DB magnitude of the hazard, and any protection against the hazard has to be designed in a conservative approach.

Existing NPPs

The following external hazards or hazard magnitudes can be screened out from the scope of design basis:

- *hazards resulting from external human activities typical of the site, if the frequency of the occurrence is less than 10^{-7} /year, or if the hazard may only occur at such a distance, that it can be demonstrated that it is not able to cause an effect on the nuclear power plant unit; and*

- *external hazards of natural origin with magnitudes bellow, 10^{-4} /year occurrence frequency, or external effects of natural origin for which it can be demonstrated that they are not capable of causing any damage to the systems important to safety of the plant.*

The regularly up-dated Final Safety Analysis Report (U-FSAR) of the existing plants is elaborated corresponding to the above requirements.

New NPPs

The screening conditions for external hazards for new NPPs are:

- *hazards resulting from external human activities typical of the site, if the frequency of occurrence is less than 10^{-7}/ry , or if the hazard may only occur at a such a distance, that it can be demonstrated that it is not able to cause an effect on the nuclear power plant unit; and*
- *hazard magnitudes of natural origin, with a recurrence frequency below 10^{-5}/ry, or external effects of natural origin for which it can be demonstrated that they are not capable of causing any damage to the systems important to safety of the plant.*

External events outside of the scope of the design basis are requested to be assessed up the recurrence frequency of 10^{-7}/ry both for new and existing NPPs as a part of the probabilistic safety assessment. External events are within the scope of all PSA requirements both for the existing and for the new plants, therefore the 10^{-4}/ry CDF and 10^{-5}/ry LERF criteria for existing NPPs and 10^{-5}/ry CDF and 10^{-6}/ry LERF criteria for new NPPs shall be demonstrated with external hazards included. The external hazard assessment shall be extended to all realistic combinations of these hazards, as well.

For the existing NPPs the following external hazards were identified in the U-FSAR as the most severe/relevant:

- *Earthquake*
- *Extreme wind*
- *Extreme snow*

- *Extreme frost (freezing rain)*
- *Extreme hot, etc.*

Although it is required by the Code to consider, up to date no relevant combination of these hazards were identified, however the external hazard assessment in general is under review and the re-assessment is ongoing right now thus these conclusions may change in the near future. For the relevant event combinations both the deterministic and probabilistic assessments shall be considered.

It also should be noted that as new information occurs due to further site studies and/or the development of science and technology the list of external hazards that need to be taken into account in the analysis can be and shall be extended. At the moment several external hazards (such as lighting and tornado) is under re-assessment in order to ensure the comprehensiveness of the list of relevant EHs.

For the new NPPs this complete relevance analysis shall be part of the supporting material of the construction licence application.

For all NPPs (existing and new) within the framework of the DEC analyses it shall be justified that there exist adequate margins to avoid any cliff-edge effects endangering the fulfilment of the basic safety functions which may result from possible internal initiating events, internal or external hazards, including their combinations.

Research Reactors

It shall be demonstrated concerning all hazards having safety aspects identified during the design and analyses have been taken into account by the designer and the relevant criteria have been complied with. Exclusion of events from the design basis shall be based on their frequency, or it shall be demonstrated that the given hazard is appropriately distant and no effect is reasonably expected on the research reactor. Every hazard of external or internal origin affecting the research reactor shall be analysed and evaluated. It shall be assumed that the hazards and hazard factors occur during the most adverse normal service conditions of the research reactor. The analysis shall take into account: reasonably assumable combination of various, simultaneously occurring hazards, and that some hazard coincides with a single failure or maintenance activities.

The design basis earthquake shall be taken into account in the design of the research reactor. The probability of occurrence of the largest earthquake that may compromise the safety of the research reactor, depending on the power and purpose of use, shall not exceed the following values taken for the whole lifecycle: a) … (not relevant this time), b) in the case of a research reactor having 0.1-10 MW thermal power: $5*10^{-2}$, c) … (out of scope of the questionnaire).

Other hazard of natural origin shall also be taken into account in the design basis depending on the power and purpose of use of the research reactor, according to the above considerations, with an exceedance probability calculated for the whole lifetime. It shall be demonstrated regarding all the potential hazards that the requirements for design specifications are adequately complied with according to the design, analysis and probabilistic principles. Only those hazards can be screened out without any further assessment, about which it can be demonstrated that no effect can reasonably be expected on the safety of the research reactor.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires “preventing accidents and, should an accident occur, mitigating its consequences and avoiding” “early radioactive releases that would require off-site emergency measures but with insufficient time to implement them” and “large radioactive releases that would require protective measures that could not be limited in area or time”. Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

The Hungarian legal and regulatory framework uses a combined term called “large or early release” (LERF) in the requirements. The Code defines the large releases as releases when the protective measures outside the site cannot be limited in a planned way in space and in time. The early releases are defined as releases, when exist necessity to implement urgent protective measures outside the site, but there is no enough time to execute them.

<u>For existing NPPs</u>

The above regulatory definitions are translated into requirements the following way:

A radioactive release where off-site protective measures cannot be limited in space and time or releases that require urgent protective measures outside of the site but there is not enough time to implement them practically means:

- *Containment failure or severe fuel damage in the spent fuel pool has been occurred, and*

- *Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor;*

or

- *There is a need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor;*

or

- *There is a need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor.*

For new NPPs

Regulatory definition is transposed into the requirements as:

- *Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor;*

  or

- *There is a need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor;*

  or

- *There is a need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor;*

  or

- *There is a need for any long-term restriction on food consumption.*

(In rough simplification, it is estimated to be around 30 TBq release into the atmosphere.)

There is a requirement, that a situation with any of the above consequences shall be practically eliminated, i.e. the "Limited Environmental Impact" criteria shall be met.

After determining the scope of the necessary severe accident analyses and implementing them by combination of deterministic and probabilistic methods using engineering judgement as well, if the physical impossibility of the scenario cannot be demonstrated then the avoidance of large or early releases shall be demonstrated through probabilistic safety assessments for which the requirements are the following:

Existing NPPs

The overall LERF value (not including sabotage and earthquake) cannot be higher than 10^{-5}/ry for while all reasonable modification and improvement should aim for the 10^{-6}/ry overall target value. To fulfil this requirement even before the Fukushima accident were initiated and partly finished some severe accident mitigating measures (e.g. introduction of SAMGs – among *others aiming prevention of high pressure core damage scenarios, installation of additional severe accident autocatalytic hydrogen recombiners, introduction of external cooling of the reactor pressure vessels with the aim of in-vessel retention of the molten core). After the Fukushima accident as an outcome of the stress test additional measures were identified and implemented, or are in the phase of implementation. Among them such as enabling any safety diesel to feed all units by way of severe accident interconnections, implementation of additional external cooling of the confinements when all safety system branches fail, that way preventing slow over-pressurization and failure of the confinement. The safety benefit and fulfilment of requirements shall be demonstrated in the phase of regulatory approval of the modifications and in the process of periodic safety reviews.*

New NPPs

The overall LERF the value cannot be higher than 10^{-6} /ry but any single scenario resulting in LER should be practically eliminated. Fulfilment of the requirements will be evaluated in the process of the construction licensing, as a precondition to issue the license.

Research reactors

In case of the research reactors it is required that, the plant states of the nuclear facility shall be identified; the postulated initiating events shall be categorized. The categories shall cover normal service, anticipated operational occurrences and design basis accidents. Acceptance criteria shall be assigned to each category by observing the requirement that frequent initiating events may only entail very limited radiological consequences, while during design basis accidents of significantly lower frequency, the compliance with the release limits specified for accidents shall be ensure. According to this, not only DEC terms, but "early and/or large releases" as well are not introduced for the research reactors, although even the much earlier analyses made for BDB scenarios and included in the FSAR do show acceptable release results even from the point of view safety objectives formulated in the NSD.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Answer to this question is equally valid for the NPPs and research reactors.

Safety improvements may be initiated in principle according to different procedures, as follows:

a. *as a usual procedure of modifications initiated by any reason (licensee's own proposal or specifically required by the regulator on the basis of the experience feed-back – either international or domestic),*

b. *a partial case of the previous, when a major event initiates a set of safety improvement measures relatively urgently, even before the requirements are codified in mandatory legal documents (an example is the result of the "stress test" after the Fukushima accident),*

c. *change of legal requirements originated from any reason (periodic review – in Hungary it is required every 5 years, harmonization of international requirements [documents – e. g. EU directives or WENRA safety reference levels], new scientific knowledge or technology, new information from the operating experience),*

d. *within the periodic safety review of the nuclear facility (required every 10 years).*

Decision on the required time frame/deadline of the implementation is different for the cases above:

For a.: if the initiative came from the licensee there are no specific considerations, when modification is prescribed by the regulator, its decision already contains the deadline.

For b.: the time schedule of the implementation is based on the practical possibility of the good quality realization (including design, procurement – when this is required including the public procurement process –, etc.). The possibility of parallel realization of multiple measures is taken into consideration as well, while priority is determined mainly by safety significance.

For c.: when changes in requirements are introduced, transition clauses are applied. Usually these require submission of a gap analysis. The licensee shall report which of the new or changed requirements applicable to the licensee and whether the nuclear facility actually fully, or partially satisfies or completely unsatisfies the new requirements, together with the safety evaluation of the deviations. Based on this assessment the licensee puts forward a proposal for the time required to implement the necessary changes to comply with the fully or partially unfulfilled requirements and submit an *application for granting temporary exemption until the completion of the modification. The nuclear safety authority issues a resolution on granting temporal partial or complete exemption from fulfilment of the requested requirements and its approved duration while taking into consideration the extent of increased risk caused by the deviation from the requirements.*

What concerns timely implementation, the extent, the resource requirements and the realistic time of implementation are taken into consideration, and that the safety requirements should not unjustifiably differ within one site for different units. The exemption period may not extend beyond the date of the next Periodic Safety Review.

For d.: the method described for case c. is practised.

Authority supervision of the timely implementation, consistent with the authority decision is based on periodic progress reports submitted by the licensee, what is typically required in the resolution.

Evaluation on completeness of the gap analysis is part of the normal regulatory safety evaluation process in case of submissions. This and the technical evaluation of individual modification proposals in the framework of their regulatory authorization procedure are covering the issue to implement all reasonably practicable safety improvements.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

The principle of continuous improvement of nuclear safety is stated at the highest level of the legal system and at the lower levels the details are specified. (While the Act on atomic energy requires non decreasing level of safety, the Government decree 118/2011 in the part related to the requirements of licensees' safety policy, explicitly prescribes continuous improvement of safety, what shall be included in the safety policy). In practice, we use many different tools to achieve this goal.

1. According to the requirements, the Periodic Safety Review for the nuclear facilities is executed in every 10 years:

- *NPP (1996 [units No. 1 and 2] 1998 [units No. 3 and 4], 2007, 2017), - as the 4 units are very similar to each other, from 2007 PSR is made simultaneously for the 4 units*

- *Budapest Research Reactor (2003, 2013)*

Within this process among others, the analysis of international and domestic operating experience, scientific and technical advances, as well as the implementation of new safety objectives and requirements are compulsory topics.

2. Evolution of the safety standards

Due to the continuous development of IAEA Safety Standards, WENRA reference levels and other reference materials, the national regulations have been reviewed and revised even more frequently than Hungarian law requires it (every 5 years). Such review had been carried out based on:

- *National experience*
- *EURATOM Nuclear Safety Directive*
- *IAEA Safety Standards*
- *WENRA Reference Levels*
- *Changes in Hungarian regulatory system*
  - *Oversight of radiation protection (+ inclusion of new BSS)*
  - *Oversight of civil structures*

as these two areas ere additionally included in the HAEA authority portfolio.

3. International peer review missions

Since its commissioning, the management of Paks NPP has paid special attention to utilizing international experience and 38 international review missions have been invited since 1984. These included all kinds of applicable review missions organized by the International Atomic Energy Agency. The last OSART review was hosted in 2014; and its follow-up mission was held in 2016. Additionally, the World Association of Nuclear Operators (hereinafter referred to as WANO) performs regular reviews at Paks Nuclear Power Plant. The follow-up of the 3rd peer review was conducted in 2014, the 4th peer review and the follow-up of the WANO corporate level peer review conducted in 2016.

In relation of the research reactor, external review was invited in the time of full-scale reactor upgrade and for the HEU –LEU transition, among others. In the framework of Europe Advisory Safety Committee for Research Reactors (EURASC) an expert site walk down took place in 2015.

In 2012, the Hungarian Government invited the International Regulatory Review Service mission to compare the country's regulatory system and practices with relevant IAEA Safety Standards. The full scope review was carried out from 11 to 22 May 2015. The follow up mission took place in September 2018. The review team found that since 2015, the HAEA has taken positive steps. While additional work remains to ensure effective implementation, Hungary has demonstrated its dedication to continuous improvement and focus on safety.

4. Utilization of external operational experience:

It is of vital interest to Paks NPP to learn and make use of operating and other experience imparted by other installations and international information sources. The operator of the plant (MVM Paks Nuclear Power Plant Ltd.) takes part in the work of significant international nuclear organizations (e.g. International Atomic Energy Agency, OECD Nuclear Energy Agency). Closer co-operation exists by way of participating in the professional work of various groups comprising operators of nuclear power plants, such as the World Association of Nuclear Operators (WANO) and the Club of VVER-440 Operators. The closest cooperation may take place between the partner plants. Links such as these enable many kinds of mutually advantageous occasional or long-term activities to be identified, including joint projects (e.g. in field of fuel development), exchange of operational experiences, and data supply.

The research reactor staff is member of the Europe Advisory Safety Committee for Research Reactors (EURASC).

The licensees and the regulatory body are regularly attending the meeting organized by IAEA, USIE, Clearing House, WENRA Working Groups, etc.

Major international events of interest (e.g. Doel, Davis Bessie, etc.) are evaluated in domestic workshops by the operator and the authority as well, taking into account the results of different international fora dealing with these issues.

5. Benchmarks:

International benchmarks of regulatory activities are performed mainly in framework of participation of the different workgroups of OECD NEA (Work group of inspection practice of CNRA), WENRA (e.g. carbon segregation issues - Reactor Harmonization Working Group). Hungary also participated/participates in various task forces and ad-hoc working groups, such as on the regulatory safety culture of the OECD NEA CNRA, timely implementation of reasonably achievable safety improvements of WENRA, RR working group established to develop safety reference levels (RLs) for existing research reactors. Vendor specific groups (Forum of WWER regulators, WWER plant operators group) are utilized as well for exchange of safety related information and practices.

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and are in forced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

The answer to this question is based on a brief historical description of safety improvement measures already executed and ongoing in the Paks NPP (the existing 4 units were put into operation from 1982 to 1987). In relation of the research reactor(s) the safety approach practically was not updated yet after the publication of documents mentioned in the question.

After the Chernobyl accident the Soviet general contractor made a proposal on several safety improvement measures, e.g. on primary-to-secondary leak (PRISE) handling. At that time the so-called “extra budgetary” program of the IAEA and the genuine Hungarian AGNES project (Advanced General and New Evaluation of Safety) were ongoing. Although not all proposals from the Soviet side were accepted, combining with the results of the two other mentioned safety reviews, a major safety upgrade program was decided and completed around the turn of the century. Its elements today can be categorized as associated with the DiD level 3, i.e. the design basis accidents, and included such measures e.g. as:

- *PRISE management,*
- *prevention of clogging of confinement sump screens used for recirculation of ECC and confinement spray systems,*
- *exclusion of possible refilling of ECC tanks during recirculation (what could cause loss of the ECC),*
- *elimination of the immediate "artificial" de-energisation of electrical safety distributors in case of any emergency signal requiring operation of the ECC system and substitution their supply from emergency diesels,*
- *installation of hydrogen recombiners capable of dealing with H_2 resulting from DBCs.*

A lengthy process was carried out as well, during which the site-specific earthquake characteristics were comprehensively re-evaluated, resulting in new input data (with higher PGA values) for the design basis and following that complex earthquake resistance measures were implemented for on all relevant systems, components and structures.

As a result of the first PSR, it was decided to elaborate and introduce the symptom oriented emergency management procedures. It was implemented in two steps, first for the power operation states and then for the low-power and shutdown states of the reactors as well, and today can be categorized as associated with the DiD level 3a (deign basis accidents) and 3b (multiple failures).

As a result of the second PSR, it was decided to implement a set of measures to deal with accidents which today can be categorized as associated with DiD levels 3b and 4 (partly by multiple failures and partly by severe accidents). These included measures:

- *elaboration and introduction of SAM guidelines,*
- *placement of additional hydrogen recombiners in the confinement for DEC scenarios, including core melt,*
- *modification of the spent fuel pond cooling system to exclude loss of water from the pond due to big leaks in the cooling system,*
- *implementation of external cooling of the reactor pressure vessel (RPV) by flooding reactor cavity with the aim to prevent the vessel failure and make possible in-vessel retention of the corium after fuel melt,*
- *introduction of possibility of connecting part of I&C system and electrical valves needed in severe accident management to an independent electrical system fed by small mobile diesel generators,*
- *addition of a special severe accident measurement system,*

- *addition of blow-down possibility of the pressurizer to prevent high-pressure failure of the RPV.*

Implementation of these measures took several years, although after the Fukushima accident, it was accelerated, and all these measures were completed a few years after the accident (around 2014).

New formulation of the nuclear safety requirements (of the Code) was issued in a new Governmental decree in July, 2011 (preparations for which were well in advance already before the Fukushima accident). This already contained definition of the different levels of DiD.

Resulting from the European stress test initiated in 2011 an additional action plan was elaborated,
(for the updated version see:
http://www.oah.hu/web/v3/HAEAportal.nsf/16D92FB22428EFFCC12581FD0050F453/$FILE/NAcP_2017_E_végl_fedlap.pdf).
The measures within it are mainly related to DiD levels 3b and 4 (multiple failures and severe accidents) and 5, and the action plan includes among others the following actions:

- *addition of further independent permanent severe accident diesel generators capable to feed all pumps needed for emergency core cooling and confinement spray (level 3b and 4),*
- *addition of an independent confinement cooling system to prevent long term over-pressurization and failure of that (level 4),*
- *construction of capability to make up the spent fuel pools from outside of the building with borated water (level 3b).*

It should be noted that implementation of some modifications of the action plan will only be finished in the coming years

In spring 2012 amendments were introduced to the Code, which determined the (DBC and DEC) plant states for the existing and new reactors (with some differences), and additional requirements on how to apply the DiD levels in the design. This was based on the draft versions of updated reference levels for the existing power plants and the safety objectives for new power reactors some later published by the WENRA.

After publication of the modified NSD further refinement of the requirements were introduced into Hungarian legal system, most notable of these are related to

- *expressively require independence of different DiD levels – as much as reasonably achievable,*
- *requirements related to management system and safety culture were extended and reinforced,*
- *requirements how the deterministic and probabilistic methods shall be applied in safety assessments.*

When the IAEA SSR-2/1 (Rev. 1) requirements document was published, the legal framework was reviewed but it did not require any substantial changes.

Implementation of these requirements by the licensee of the existing NPP followed the mechanism described above in the answer to question 3.6, according to procedure c. In the case of the new build units as the set of requirements is in force, it shall be followed by the design and safety substantiation, and shall be presented for the authority in the forthcoming application for the construction license.

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

In our understanding this question relates solely to the NPPs. For the most important safety improvements executed and planned in the existing Paks NPP see the historical overview in the answer to question 4.1.

It may be added, that the last PSR of the 4 NPP units was finished in January 2019 by the decision of the authority. This decision reinforced the prescription to implement the remaining open issues from the post-Fukushima action plan and required execution of additional measures. Among these, there is a number of new analyses (e.g. systematic analysis of hazard combinations to be included into the design basis and PSA), after which the results could initiate design modifications to be implemented. Among other measures, for enhancement of the containment function, beside re-qualification and change of cable penetrations for some additional valves on tubing penetrating the confinement wall introduction of inspection possibility of leak-tight closure was prescribed as well.

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Note: In our understanding question 5.6 still refers to nuclear power plants (therefore we answered the question accordingly) and question 5.7 deals with research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- *PSA level 1: The PSA model and its documentation is required and shall be part of the construction license application, furthermore the licensee have to update it before commissioning and before operation. PSA is also needed in case of major plant modifications. In all licensing stages, the following end points are required: core damage frequencies, fuel damage frequencies (for spent fuel pools) complemented with sensitivity, importance measures, uncertainty analyses, list of major contributors, etc. The scope shall cover the full power, the low power and the shutdown initial states, and also all internal and external hazards.*

- *PSA level 2: The PSA model and its documentation is required and shall constitute part of the construction license application, furthermore the licensee have to update it before commissioning and before operation. PSA is also needed in case of major plant modifications. In all licensing stages, the following end points are required: release frequencies by categories (based on quantity, composition, timeliness) complemented with sensitivity, importance measures, uncertainty analyses, list of major contributors, etc. The scope shall cover the full power, the low power and the shutdown initial states, and also all internal and external hazards.*

- *PSA level 3: Not required for any licensing activity at the moment. Possible level 3 PSA requirements and methodologies are under research and development.*

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required /provided at the different stages (licensing, operation…)?

Yes. Although it should be noted that while a full scope PSA both for level 1 and level 2 is required for all major licensing phases, but e.g. in the construction licensing phase no local operational experience/data is/are available. Therefore, during this phase more generic data, *assumptions and some simplifications within the models are acceptable by the regulatory body, as long as it can be ensured with a high confidence that the data and assumptions used for the assessment supporting the Preliminary Safety Analysis Report is more conservative than what we can expect for the operation.*

Uncertainty and sensitivity should be assessed regardless of the type of licensing procedure actually performed.

For the existing units the PSA was prepared in phases and after many years of being in operation. Otherwise requirements are mostly similar for them and for the new-build NPPs (the required methodology remained the same but the quantitative criteria are lowered by one order of magnitude). The PSA model is regularly up-dated and also applied for supporting any modification request.

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

A full scope PSA (internal initiating events, internal hazards, external hazards in all operating states) is required by law both for operating and new NPPs including their spent fuel pools as well.

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Yes. A full scope PSA (internal initiating events, internal hazards, external hazards in all operating states) is required by law both for operating and new NPPs including their spent fuel pools as well.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

There are no exact requirements at the moment to develop a multi-unit PSA or an integrated PSA which covers multiple installations at the same site, although the effects of the units and nearby nuclear installations on each other is required to be assessed in general (not necessarily through PSA methods). At the moment there is no finished multiunit PSA submitted to the regulatory body (nor it is required within the current legal framework), but some activities are ongoing in this direction.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

There is a general safety requirement, that the PSA shall prove that the design is "balanced", i.e. there are no outstanding contributors among the initial events or the minimal cut-sets predominantly influencing the PSA results while importance measures shall also be taken into consideration to justify that the design is balanced.

The maintenance, testing and oversight programmes (e.g. test frequencies and durations) of systems and system components important to nuclear safety are determined taking into consideration the PSA results. In addition, PSA is used for the evaluation of significance of operational occurrences and in the justification of planned modifications of a system or a component, the Operational Limits and Conditions or of plant procedures.

PSA is also be used to support safety management, i.e. for identification whether modifications of the unit or procedures are necessary, including severe accident management guidelines. The results of the PSA are also used for the safety-related training programmes of the operating staff, especially for the development and validation of simulator training of control room personnel and to ensure that verification and testing programmes contain those aspects that are significant contributors to the risk.

The above listed requirements are implemented in practice, and PSA is also used for maintenance effectiveness monitoring, maintenance planning and evaluation (risk monitor – currently off-line), and for event analyses.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

PSA is updated and the new model version with its corresponding documentation is submitted to the Hungarian Atomic Energy Authority on a yearly basis by the licensee. Since the Hungarian legal and regulatory framework requires the licensee to justify that any and all modification proposed by the licensee do not decrease nuclear safety, the licensee have to perform a PSA and update and submit their models and calculations whenever it applies for a regulatory approval for a risk/safety significant modification. It is also required to maintain a living PSA; therefore, changes in the generic data (e.g. international experience-based reliability data of certain component types, such as valves) shall be introduced into the model(s) as well and submitted to the regulatory body with the annual version update.

Question 5.7: Are PSA required / developed for new and/or existing research reactors? If so:

According to Code requirements full PSA is not required, however fault-logic models for the research reactor shall be developed, which shall systematically include each of the possible operational states, system configurations and all the postulated initiating events, and which can serve as a basis for a future probabilistic safety assessment. Probabilistic safety assessments shall be performed, only if a reliable database can be produced. The licensee of the research reactor shall endeavour to collect all the data, which may be utilized in a future probabilistic safety assessment.

(It could be noted as well that on the turn of the 80-s and 90-s of the last century, at the time of the complete refurbishment of the research reactor of the KFKI, when almost all systems were changed and substantial additional safety features were implemented - e.g. reserve diesels and a passive ECC system were added - a level 1 PSA [only full power state, only internal initiating events] was developed to support the substantiation of safety. The data used for the assessment as well as most modelling techniques applied for it are outdated now and cannot be used as a basis for PSA based applications today. Research in this field is under planning and is part of the HAEA's strategy.)

- What levels of PSA are performed?

  No requirements for levels, however parts of level 1 PSA is available. (See above.).

- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?

  The requirements only define that each of the possible operational states, system configurations and all postulated initiating events must be covered in the fault-logic model.

- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?

At the moment PSAs are not used for supporting modifications and safety improvements.

- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

  There are no associated thresholds related in these requirements.

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis” as mentioned in Article 8b? If so, please explain.

In regulation and practice for new nuclear power plants DiD level 3a is associated with DBA3-4 and DiD level 3b is with DEC1 (DEC without fuel melt). For operating nuclear power plants and research reactors these are not implemented in the regulations, but in practice DiD level 3 is associated with DBA3-4 and DEC1.

For research reactor In accordance with the DiD levels, the design basis distinguishes three categories of operation, each of which must be provided with criteria to meet the safety objectives.

The three operating categories are:

• *normal operation (NO)*

• *anticipated operational occurrence (AOO),*

• *postulated accident (PA).*

In addition to the design basis, we also distinguish beyond design basis accidents (BDBA) and serious accidents (severe accidents). These categories are not subject to numerical criteria under the current regulations, but at the time of designing the reactor, the design, temperature and pressure conditions of the reactor and primary cooling circuit exclude the possibility of "dry" the entire active zone.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Operating NPPs

DEC states are separated into three categories and these categories are identified , listed and used in a somewhat different manner. These categories are the following according to the FSAR:

- *DEC1: Complex accident without core melt/core damage/etc. (occurrence frequency: 10^{-05}/ry – 10^{-07}/ry)*
- *DEC2: Severe accidents with core melts/core damages/etc. that does not lead to unacceptable releases. (occurrence frequency: 10^{-5}/ry – 10^{-7}/ry)*
- *Very severe accidents that lead to unacceptable radioactive releases. (<10^{-07}/ry, not including scenarios originating from earthquakes)*

In addition to the general requirements, a minimum list of event to be considering is given in the NSC (3.2.2.3900 and 3.2.2.391). This includes among others SBO, ATWS, extraordinary strength natural hazards, and etc. It should also be described that in the Hungarian practice the FSAR analysis at least one order of magnitude lower than the probability thresholds required by the NSC.

As a first step, scenarios that lead to DEC are identified on the basis of their occurrence frequency (ergo by PSA) and grouped based on their nature and transient behaviour. The probabilistic screening criterion for such accidents is 10^{-07}/ry, which is considered as the frequency threshold of practical elimination in the Hungarian practice, therefore accident scenarios beyond this threshold can be eliminated from further assessment. Whether a scenario belongs to DEC1 or DEC2 can only be determined after their end-state/outcome is known; therefore a representative scenario shall be chosen from the groups based on a conservative approach which will be subject of detailed analysis. These assessments/analyses are allowed to be based on best estimate approach.

The regulatory body reviews and approves all steps of this identification process, both for the probabilistic and the deterministic safety assessment parts. The up-to-date nature of this list is ensured through two main requirements from the Hungarian requirements:

- *The licensee shall follow and integrate any and all new scientific knowledge and technology relevant to its nuclear safety, as well as its own operating experience into its assessments continually, ergo if an information occurs that may contradict the assumptions or the results of the assessments on which the list of DEC1 scenarios were developed than it shall be reviewed and modified as necessary.*
- *The licensee have to carry out a comprehensive review and evaluation of its nuclear safety, which includes the review of the assumptions, analyses and results that justifies the DEC1 and DEC2 categorisation.*

New NPPs

The identifications of DEC1 scenarios for new NPPs are fairly similar (at least from a legal point of view, since we do not have a PSAR or FSAR at the moment from which we could provide the exact methodology) but the required frequency thresholds for DEC1 and DEC2 are one order of magnitude lower than for the operating NPPs.

Research Reactors

The investigation began with the mapping of all initial events that could be imagined at the reactor. Since the reactor is located relatively close to residential areas in a research facility, the upper limit of conceivable events has been drawn at events beyond the natural law and technical principles. Only these events are considered unthinkable. The expected incidence of individual incidents / accidents and the severity of the consequences should be determined. The following are the categorization procedure for the research reactor:

- Initial events with a frequency greater than 10^{-3}/cycle were categorized as AOO, with less frequencies in PA

- where no estimation of the frequency of occurrence was available, we decided on the category based on other considerations.

The regulatory body reviews and approves all steps of this identification process:

Yes, it is reviewed in the scope of PSR approval for every facility.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Yes, for the existing NPP units the Nuclear Safety Code paragraph 3.2.2.3900, point j) is requiring to consider and include accidents of the spent fuel pools into these lists:

"3.2.2.3900. For the extension of the design basis the following shall at least be considered, if it is not yet included in the design basis and if it is applicable for the specific nuclear power plant type:

j) loss of cooling of the spent fuel pool…"

An example in short: for a DEC1 scenario originating from the Spent Fuel Pool (SFP) is the 200% LOCA on one of the coolant loops of the SFP at an isolable section and the operator action or the automatic closure of the isolating valves are successful and the configuration changes to restore the cooling function takes place. The occurrence frequency of such an event is in the order of 10^{-07}/ry due to the very low initiating event frequency; ergo it falls into the DEC category. However the successful automatic or operator action means that the fuel damage will be avoided which means that the sequence belongs to DEC1.

For the new build NPPs similar requirement is contained in the Nuclear Safety Codes points 3a2.2.0300. a) and 3a.2.2.6300. j).

For research reactor it is also *requiring to consider and include accidents of the spent fuel pools, if such system exists. In the FSAR of Research Reactor it is analysed for loss of coolant, loss of flow, reactivity accident, fuel damage and external events scenarios.*

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?

The DEC1 scenarios shall be grouped and a representative scenario shall be chosen from the group based on a conservative basis. Once the representative scenario is chosen it can be assessed on a best estimate basis and the single failure criterion no longer has to be applied to it.

- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?

Provisions and operator actions for DEC1 scenarios are required to be included in the EOPs both for full power, low power and shut down states.

- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

For the operating NPPs the same systems are used to limit and mitigate the consequences of DEC1 as for DBA.

For the new NPPs the requirements are more strict and specific:

In the process of classifying the safety functions into levels, it shall be identified to which of the levels of the defence in depth the given function can be assigned, primarily in order to allow an evaluation of the independence of the levels of defence.

There are also specific requirements for certain DEC1 equipment such as:

It shall be ensured with adequate design that:

cc) Under a station blackout condition (DEC1) DC accumulators performing F1 safety function – independently from F1 accumulators used under DB2-4 conditions – shall be designed to operate and perform their F1 safety function for at least 6 hours without recharging.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

In the operating NPPs mostly the same SSCs are used for limiting the consequences of DEC1 scenarios as for DB accidents, therefore their safety classification remains the same (Safety Class 2-3).The requirements for these SSCs changes for DEC1, namely that the single failure criterion no longer applies to them, assumptions and considerations can be made on a best estimate approach and the 95/95 rule (95% probability at a 95% confidence level of success) does not have to be followed. Systems that play a role in limiting and mitigating the *consequences of DEC1 scenarios are required to endure the conditions that may arise during these scenarios.*

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

While we do not have a new NPP under licensing at the moment (construction license application is expected in the near/mid future) therefore we do not have detailed information on the SSCs dedicated to limit and mitigate the consequences of DEC1 accidents, meaning that we have not yet seen the licensee's proposal for the safety classification, we can assume that there will be such systems and under the current regulation these systems should fall at least into SC3 category. Their requirements remain the same as for the operating NPPs (no single failure, BE assumptions and calculations without 95/95, etc.). For the new NPPs the threshold value for DB4 is one order of magnitude lower than for the operating NPPs, therefore DEC1 systems are required to endure conditions at a one order of magnitude lower occurrence frequency.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

Yes. Design extension conditions or beyond design basis conditions are separated into three parts:

- *DEC1: Complex accident without core melt/core damage/etc.*

- *DEC2: Severe accidents with core melts/core damages/etc. that does not lead to unacceptable releases.*
- *Very severe accidents that lead to unacceptable radioactive releases and consequences.*

In the practice the FSAR of the NPP fully complies with these requirements.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

The scope of DECs with fuel melt conditions are defined through and ensured by the combinations of PSA and DSA methods. As it was described above, DEC scenarios (by definition) can be identified through a probabilistic approach but their consequences could only be determined on deterministic basis/through deterministic assessment or in the most obvious cases by engineering judgement (e.g.: reactor vessel rupture).

It is required to review and revise the scope of all DECs (just like all the DB cases) with and without fuel melt, as well as very severe accidents during the PSRs and if new technological or scientific knowledge or if new operating experience occurs that may contradict previous assumptions and assessments on which the scope of DECs were determined, new analyses have to be carried out.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Yes, it is required by law to include DECs with potential fuel melt in the SFP. One of the sequences with the highest contribution to the overall FDF is the following:

- *Loss of flow (e.g.: SFP coolant pump fails while running, the redundant system is under maintenance/non-available) during POS5 operational state, when the reactor core is totally and recently unloaded (the thermal output of the spent fuel in the pool is in the range of 700kW – 1 MW)*
- *The operator does not recognise the problem or fails to change system configuration to recover the cooling function as it is required by the EOP (e.g.: by starting the injection of water from the trays in the localization towers or redundant systems) nor does he/she*

recover the failed system before the coolant starts to boil and the coolant level starts to decrease.

- *The critical point for the sequence is the elevation of the cooling loops, once the water level drops below this point, there is no longer enough time to recover the lost cooling functions since the operator has to refill the SFP first.*
- *If the operator fails to refill the SFP and to recover the failed system or change the system configuration to recover the cooling function, the FD occurs.*

The occurrence frequency of this sequence is in the range of 10^{-7}/ry and leads to severe fuel damage;

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?

The consequences of DEC2 and very severe accidents are limited through level 1 and 2 PSA criteria that differ both in frequency limit and definition for operating and new NPPs. To our current knowledge, the definition of core damage is also somewhat different for the AES-2006 type reactors planned to be built at the Paks site and the VVER 440/213 type reactors currently in operation due to differences in the design and parameters of the fuel and the assemblies (different oxidation percentage, different radial enthalpy limits, etc.). Under the current regulation the exact definitions of core damage, the large or early release for the operating NPPs, and the core damage for the new NPPs should be proposed by the licensee and approved by the regulatory body. Meanwhile the exact/practical definition of large or early release for new NPPs is given by NSC Volume 3a.

Operating NPPs: The requirements are given above in Table 3.

For the operating NPPs in the Hungarian practice the goal of DEC2 deterministic analysis is to provide the best possible input for the operator to prepare for the most probable DEC2 scenarios, therefore the analyses of such accident scenarios are required to be done on a best estimate basis. Due to the very high number of possible DEC sequences, these scenarios are grouped and a representative member is chosen from the group on a best estimate basis for *further detailed assessment. The purpose of the analyses is to prove that the criteria in Tables 3 and 4 are met except if it can be proved that the frequency is less than 10^{-7} / year,*

- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?

The requirements have been shown above in Tables 3&4 in the lines for DEC2 and also have been discussed in more detail. Based on the results of these assessments/analyses provisions should be made mainly in a form SAMG recommendations to limit and mitigate the consequences of such events. The probabilities of the success paths are evaluated by the Level 2 PSA and it is to be shown that sum of the failure paths is below the probability limit of "practical elimination".

- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

Yes, it is required to develop independent procedures and guides for DEC2 conditions, and once a DEC2 condition is reached the operator is supposed to change from the EOPs to the SAMGs and there are dedicated DEC2 systems that play no role in DB1-4 or DEC1 conditions (e.g.: DEC2 power supply systems, severe accident measurement system, etc.)

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements(redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Under the current regulation safety systems dedicated limit and mitigate the consequences of DEC2 scenarios fall in SC3 category. There are no redundancy requirements for DEC2 systems, but they shall be designed to endure the conditions occurring under DEC2 scenarios.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

We do not have an answer for this yet, because we do not have the licensing documentation at the moment that would contain this information. However, the legal requirements are stricter for new reactors, thus it is expected that the consequences shall be lower.

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak-tightness, integrity, etc.)?

The nuclear safety requirements in Hungary are to a certain extent non-prescriptive; therefore, they require that in the process of design, and later in (periodic) safety reviews all the accident processes, requiring functioning of the containment shall be determined, and based on these analyses required numerical values of the technical parameters of the containment shall be calculated. Accordingly the containment shall be designed and validated for limiting dose consequences of incidents and accidents, which are numerically set in the legally binding design requirements.

The general requirement is, that as far as reasonably achievable, the events which challenge the integrity of the barriers (and as such of the containment) are prevented, the circumstances stemming from hazards are tolerated by barriers, concurring failure of more than one barrier is avoided, and that a barrier shall not fail due to the failure of another barrier or other component. Furthermore it shall be ensured that the building structures bear the loads (including environmental circumstances) occurring in DBC 1 to 4 and DEC 1 conditions in accordance with the criteria for the given condition.

At the same time, quantitative and qualitative requirements are set as:

- *It is required that leak tightness is limited for DBC 2-4 such, that the releases are as low as reasonably achievable and*
  - *for DBC 2 for reference group doses are within the dose constraint (90 µSv/yr) (both for existing and new units),*
  - *for DBC 3 for reference group doses are not more than 1 mSv/event, (only for new units),*
  - *for DBC 4 for reference group doses are not more than 5 mSv/event, (both for existing and new units);*
- *for DEC 1 releases are minimised as reasonably achievable for existing plants and for new units the DBC4 criteria are applicable;*

- *for DEC 2 it is ensured that the controlled release function of the containment ensures sufficient time is available should protection measures be necessary for the population, and long-term contamination of large areas can be avoided for existing units and for new units the "Limited Environmental Impact" criteria are applicable (See the answer for 3.5!);*
- *for all DBC and DEC conditions removal of heat, overpressure protection of the construction and handling of arising flammable gases are ensured, thus the containment integrity shall be maintained (both for existing and new units);*
- *Loss of structural integrity of the containment shall be practically eliminated.*

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

See the introductory part of the answer 8.1, furthermore, the nuclear safety code requires, that,

- *As part of the study of the initiating events, the special internal hazards, such as flooding, fire, explosion, high energy pipe break shall be identified, the occurrence of which may influence the performance of function of the isolating barrier.*
- *The containment shall be designed as a pressure retaining structure.*
- *Along design the operational conditions and mechanical loads, the load cycles including those effects caused by internal and external hazards shall be determined for pressure retaining components/structures.*
- *For systems, structures and system components confining radioactive releases or performing retaining physical barrier functions, criteria for the maximum pressure, and the minimum temperature, thermal and pressure transients, degradation and stresses depending on the temperature range shall be defined for their whole lifetime with the aim enabling to perform their safety function.*
- *For enabling the nuclear safety requirements for the whole life-cycle of the containment criteria shall be established for its temperature, pressure and leak tightness.*
- *It shall be ensured that the building constructions bear the loads, environmental conditions in all DBCs and in DEC 1, DEC 2 according to their safety classification and conformance criteria for the given design condition.*

Question 8.3: What is the approach for containment penetrations? Is their leaktightness regularly tested?

For existing and new units:

The general approach is that hermetization of pipelines penetrating the containment wall shall be ensured by a set of isolation valves (under the current regulation minimally two, one on the inside of the containment and one on the outside as close to the wall of the containment as possible). The number of penetrations shall be reasonably minimized and all penetrations shall fulfil the design requirements related to the hermetization and integrity of the containment. The penetrations shall be protected against reaction forces caused by movements of pipelines as well as by loads caused by accident event sequences, among others internal and external hazards, water jets, flying objects, pipe whipping, missiles.

Entering into containment shall be possible through interlocked air locks, maintaining that at least one of the doors remain closed in all DBC conditions, and functionality of the air locks for DBC 1 and 2 conditions shall be maintained. For transport of equipment in and out such hutches shall be designed doors of which could be quickly and reliably closed when isolation of the containment is needed.

Design of the containment shall make possible periodic testing of local leak-rate of penetrations. At the time of the operation, the maintenance-, test- and surveillance program of the containment integrity shall include check of the leak rate, examination of the penetrations' sealing, of closing valves, of air locks with the aim of justification of their leak-tightness, and as necessary of operability, as well as the general integrity of the whole construction.

For the existing units:

Isolation valves on pipelines penetrating the containment wall shall be operable from the distance or being locked in closed position. Design specification of the valves shall be determined taking into account all DBC conditions. They shall be located as close to the to the containment wall as possible. Open or closed position of the valves shall be signalized at the unit's command point(s). The units comply with these requirements according to the regular tests and the PSRs.

For the new units additionally:

The valves shall be quickly operable in order to limit the flow, furthermore the inner valve may be a check-valve if its position is controllable continuously. Additional requirements are, that in *case of the double-containment design its characteristics shall be accounted for, the pipe section between the isolation valves shall be provided by a leak-control system, and the isolation valves operated by auxiliary energy shall be fail-safe in relation of the loss of this energy supply as much as practicable.*

On the other hand for the new units it is acceptable to have only one valve outside of the containment in the case, but only if the pipe in question does not join directly to the boundary of the main cooling circulation or the inner air volume of the containment, and the integrity of the pipe section up to the closing valve is ensured.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Practical elimination is required for certain conditions but it is only indirectly defined for new NPPs. In Nuclear Safety Codes 3a.2.2.7200 practical elimination is defined for initiating events as physically impossible and/or its occurrence frequency is lower than 10^{-7}/ry at a high confidence level. In practice the same definition is used for the operating NPPs as and research reactors as well, but they are not defined on the level of requirements. The total frequency of event sequences resulting in large or early releases summarized for all initiating operating conditions and effects, excluding sabotage, shall not exceed 10^{-6}/ry. The fulfilment of the requirements shall be demonstrated by Level 2 probabilistic safety analyses.

Question 9.2: Do you have requirements related to practical elimination?

Yes, several requirement refers to practical elimination both for operating and new NPPs, e.g.:

Operating NPPs:

NSC 3.2.2.4310. In order to minimise uncertainties and to increase the robustness of the nuclear power plant unit, during the demonstration of practical elimination, demonstration based on physical impossibility shall be preferred to demonstration on a probabilistic basis.

NSC 3.4.5.0910. In the case of a nuclear power plant having more than one unit, direct electrical connection shall be provided between the units to the reasonably practicable extent in such a way that the spread of possible defects from one unit to another should be practically eliminated.

NSC 3.4.6.1210. The loss of the structural integrity of the containment shall be practically eliminated. To this end, equipment stored on or off site may also be used for controlling the conditions prevailing in the containment.

New NPPs:

NSC 3a.2.2.7200. At least the following events shall be practically eliminated by design solutions or the implementation of preventive accident management capabilities, i.e. it shall be demonstrated that their occurrence is physically impossible or the frequencies of their occurrence are less than 10^{-7} /year with high certainty:

a) rupture of the reactor vessel,

b) reactivity accidents with prompt criticality, including the cases of heterogeneous boric acid dilution,

c) all loads appearing in the short and long run, which may jeopardise the integrity of the containment, in particular, the dropping of a heavy load, steam and hydrogen explosion, interaction between the molten core and concrete loadbearing structures, and containment over-pressurization,

d) loss of cooling during the storage of irradiated fuel elements, which may lead to damage to the fuel elements, and

e) loss of coolant with open containment, which may cause core dry out.

NSC 3a.2.2.7300. In order to minimise uncertainties and ensure robustness of the safety of the nuclear power plant unit, demonstration of physical impossibility shall be preferred to demonstration of low probability when justifying practical elimination.

NSC 3a.2.4.0800. Events resulting in large or early releases shall be practically eliminated. The total frequency of event sequences resulting in large or early releases summarized for all initiating operating conditions and effects, excluding sabotage, shall not exceed 10-6 /year. The fulfilment of the requirements shall be demonstrated by Level 2 probabilistic safety analyses.

NSC 3a.4.5.1200. In the case of a nuclear power plant with more than one unit, direct electrical connection between the units shall be provided to the extent reasonably achievable in such a way that the propagation of possible failures from one unit to another can be practically excluded.

NSC 3a.4.6.1400. The loss of the structural integrity of the containment shall be practically eliminated. To this end, equipment stored on or off site may also be used for controlling the conditions prevailing in the containment.

NSC 3a.4.6.2000. The isolation of the containment shall be possible even in the case of DEC1 and DEC2 operating conditions. If an event leads to environmental releases by bypassing the protective shell of the containment, the consequences shall be mitigated. If an event leads to bypassing the containment, design solutions shall be provided that practically eliminate the possibility of damage to the fuel elements with high certainty.

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?

Yes, e.g.:

NSC 3.4.5.0910. In the case of a nuclear power plant having more than one unit, direct electrical connection shall be provided between the units to the reasonably practicable extent in such a way that the spread of possible defects from one unit to another should be practically eliminated.

- For new power reactors?

Yes, e.g.:

NSC 3a.2.2.7200. At least the following events shall be practically eliminated by design solutions or the implementation of preventive accident management capabilities, i.e. it shall be demonstrated that their occurrence is physically impossible or the frequencies of their occurrence are less than 10^{-7} /year with high certainty:

a) rupture of the reactor vessel,

b) reactivity accidents with prompt criticality, including the cases of heterogeneous boric acid dilution,

c) all loads appearing in the short and long run, which may jeopardise the integrity of the containment, in particular, the dropping of a heavy load, steam and hydrogen *explosion, interaction between the molten core and concrete loadbearing structures, and containment over-pressurization,*

d) loss of cooling during the storage of irradiated fuel elements, which may lead to damage to the fuel elements, and

e) loss of coolant with open containment, which may cause core dry out.

- For existing research reactors?

No.

- For new research reactors?

No.

The main difference between the use of the practical elimination concept for operating and for new NPPs is the extent to which it is used. For the new power reactors the use of the practical elimination concept is one of the cornerstones of the design and is widely used to ensure the expected level of nuclear safety of today's standards.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

The answer could be the safety improvements described in Q4.1.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

There are two ways the regulatory body accepts as justification of practical elimination, either the sequence in question has to have a lower occurrence frequency as 10^{-7}/ry or the occurrence has to be a physical impossibility. The fulfilment of the requirements shall be demonstrated by Level 2 probabilistic safety analyses.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Practically yes, but there are no exact requirements related to practical elimination in relation to SFPs.

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

The scope of PSR is clearly defined in the Hungarian regulatory framework. The respective requirements of PSR appear in various levels of legal instruments.

The Atomic Energy Act states: "The Licensee and the authority before the construction and commissioning, and taking into account the scientific development and the operational experiences throughout the lifetime in the form of a review and its report regularly prepared, shall analyse, evaluate and publish the results of: Safety of the facility, Fulfilling of the requirements, Risk level"

Additional requirement in the next level legal instrument in Governmental Decree 118/2011 Chapter V states the followings:

- *"The licensee shall prepare a periodic safety assessment every 10 year for all nuclear facilities and the results shall be presented to the nuclear safety authority in a Periodic Safety Assessment Report.*
- *PSR prepared one year before the authorities' own deadline; corrective action plan shall be prepared.*
- *Comparing operational risk of the facility with the Final Safety Assessment Report, national requirements, and best practices.*
- *Authority review finished by decision; modifications of operational licence possible.*
- *Scope: defined and justified, as wide as possible".*

The most detailed requirements regarding the PSR are included in Nuclear Safety Code. In addition to the above-mentioned requirements, the Nuclear Safety Code says:

"The scope of the review shall be defined and justified, but at the same time as wide as possible, taking into account the safety aspects of the facility. The minimal contents of PSR shall be as follows:

- *The design of the nuclear facility documented in the Final Safety Analysis;*
- *Site characteristics, resistance to external hazard factors;*
- *Decommissioning;*
- *The current condition of systems and system components;*
- *Equipment qualification;*
- *Ageing;*
- *Deterministic safety analysis;*
- *Probabilistic safety assessment;*
- *Analysis of hazard factors;*
- *Safety indicators of the nuclear facility;*
- *Evaluation and feedback of relevant technical and scientific results, and operational experience; Utilization of research results and the experience of other similar nuclear facilities;*
- *Organization, human factors, management system and safety culture;*
- *Procedures;*
- *Accident management;*
- *Nuclear emergency preparedness;*
- *Radiation protection of employees and the population and radiation exposure of the environment; and*
- *Experimental equipment in the case of research reactors".*

For supporting the review and assessment procedure the HAEA released specific guidelines:

A1.39. "Guideline to the implementation of Periodic Safety Review of the Paks Nuclear Power Plant".

1.51 "Guideline to the implementation of Periodic Safety Assessment of the Budapest Research Reactor".

The Licensee was actively involved in the process of the development of the guideline via consultancy meetings. The guideline summarizes the purpose of the PSAR and specifies the scope, the time lines, the legal requirements, the standards, the volumes of the report and the quality assurance requirements. The "safety factors" to be reviewed during the PSR, as detailed in the guide. With a specific chapter title, a requirement for content and mandatory attachments.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

The safety objectives of PSR is determined in the 1.73 Chapter of Nuclear Safety Code. The determined safety objective of PSR is in accordance with the IAEA SSG-25.

— The adequacy and effectiveness of the arrangements and the structures, systems and components (equipment) that are in place to ensure plant safety until the next PSR or, where appropriate, until the end of planned operation (that is, if the nuclear power plant will cease operation before the next PSR is due);

— The extent to which the plant conforms to current national and/or international safety standards and operating practices;

— Safety improvements and timescales for their implementation;

— The extent to which the safety documentation, including the licensing basis, remains valid.

The safety factors to be considered are fully in accordance with the IAEA SSG-25 and WENRA Safety Reference Level P2.2. In 2015 HAEA reviewed the Hungarian legal requirements compliancy with WENRA reference levels and found that the legislation are in line with WENRA standards, so no legislative amendment was required. The detailed safety factors are in the answer to question 10.1.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

In the section 10.1 it was mentioned HAEA released specific guidelines for supporting the review and assessment procedure of PSR. In the Hungarian practice the development of this guideline is the first step of PSR and the Licensee was actively involved in the process via *consultancy meetings. The guideline has been released before the Licensee begins his PSR process. Among others the guideline specifies the legal requirements and the standards. Under the development of guideline HAEA takes into account the most recent developments in international standards. The members of HAEA staff are actively involved in the development of international standards through the IAEA and WENRA, so they also have direct information about developments. For example:* the *IAEA Safety report on Periodic Safety Review for Research Reactors has two Hungarian contributors to drafting.*

The Hungarian regulations shall be periodically (at least every 5 years) reviewed as required by the Act on Atomic Energy. Although the Hungarian requirement system has been continuously developed for the recent years, HAEA started the periodic review of the actual requirements system in the beginning of 2015. The revised regulation was issued in 2018. An important aspect of regulation development was to take into account the recommendations of international organizations, especially the recommendations of the Western European Nuclear Regulators' Association on international good practice, the so-called reference levels, as well as the good practice of certain countries with developed nuclear technology.

Hungary is represented by the HAEA in the organization of WENRA. The WENRA reference levels were established for the first time in 2008, the review of the requirements was completed in 2013. The new reference levels have been incorporated by Hungary in the Nuclear Safety Code.

In addition to the continuous development of IAEA recommendations and WENRA reference levels, the update of the national regulations has been performed more frequently than every 5 years as required by law.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

The HAEA has elaborated Nuclear Safety Codes (NSC) which contain mandatory safety requirements for nuclear facilities. Under chapter 2.2.2. in the volume 2 of NSC, new requirements are included regarding to enhancing and promoting a strong safety culture.

Compliance with the nuclear safety requirements and provisions are mandatory for all those, who are under continuous regulatory supervision according to the Atomic Act. In addition to the nuclear safety requirements and provisions the requirements involve individual authority prescriptions, conditions and obligations, what the HAEA is authorized to determine as a nuclear safety authority in a resolution made for nuclear safety of a nuclear facility.

The HAEA developed a Regulatory Framework Web-portal with links between different levels of requirements and guides, among others guide No. 2.18. Assessing safety culture and utilizing results at nuclear facilities.

The HAEA implements its annual inspection plan including inspections on safety culture issues as well.

The licensees send their annual reports to the HAEA, which contain a separate chapter on safety culture and HAEA evaluates the reports.

Based on the findings of the self-assessment, the new requirements of the NSC and the inspection records the requirement of safety culture approach has been incorporated into newest version of the NPP's procedure No. FEL005.

According to the reporting criteria of the WANO guidelines "Operational Experience Program", a number of minor events are reported to the member NPPs that the licensee processes for its own development. The event reports are also received through IAEA's IRS system. A brief summary of all events and also the WANO case list are received by the specialists for information. The beneficial effects of external experiences are described in the Safety Indicator System - developed by the HAEA - BMR C/II/2/2. "Usage of exploited foreign event reports".

The requirement 1.7.4. „Occasional reports" in Volume 1 of the NSC describes which cases and within what time the licensee is required to report major errors, non-compliances, extraordinary events to the nuclear safety authority.

The paragraphs 4.4.1.1000.-2600. and the chapter 4.12. of NSC contain requirements on education and training relating to the nuclear safety and on-site preparedness of the licensee's staff.

HAEA has elaborated its procedure No. ME-0-0-69 on the survey and evaluation of its safety culture. Based on the procedure HAEA has performed the self-assessment of its safety culture level.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

Our answer contains the relevant points of the Directive in alphabetical order and then the related regulations which fulfil them. The main source of the information which will cover most of the answers is the Nuclear Safety Codes which are annexes of „Government Decree No. 118/2011 (VII. 11.) on the nuclear safety requirements of nuclear facilities and on related

regulatory activities" the second highest level of regulation.

Answers to the Question 11.2 according to the relevant points of the directive:

Relevant point:

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);

Answer:

Annex 2 of Government Decree No. 118/2011 (VII. 11.)

Nuclear Safety Code Volume 2: Management Systems of Nuclear Facilities

2.2 MANAGEMENT SYSTEM

2.2.2 Safety Culture

2.2.2.0100. The managements of the licensee organisation and the supplier organisations shall consistently and definitely expect and support the attitude required for a strong safety culture at all levels, and shall ensure that the employees recognise and understand the key considerations of safety culture. Among other things, they shall implement this in such a way that they do not support excessive self-confidence and encourage an open reporting culture and a questioning attitude, which prevent activities and conditions unfavourable from a safety point of view.

2.2.2.0200. The management system shall provide the means required for the systematic development and support of attitudes resulting in a strong safety culture. The suitability and efficiency of the means developing and supporting the safety culture shall be verified at regular intervals, in self-assessments and a review of the management system.

2.2.2.0300. The licensee shall ensure that suppliers and subcontractors also meet the requirements stated in Sections 2.2.2.0100 and 2.2.2.0200.

Relevant point:

(b) arrangements by the licence holder to register, evaluate and document internal and external safety significant operating experience;

Answer:

b) Annex 4 of Govt. Decree No. 118/2011 (VII. 11.)

Nuclear Safety Code Volume 4: Operation of Nuclear Power Plants

4.14. OPERATING EXPERIENCE

4.14.1. Collecting experience from nuclear power plants

4.14.1.0100. The licensee shall develop and implement a systematic programme for the regular and continuous collection, screening, analysis and documentation of operational data, experience and operational events of the nuclear facility throughout the commissioning, operating and decommissioning phases of the nuclear facility. Operating experience and operational events reported by other operators and relevant to the facility shall also be considered.

4.14.1.0200. The operating experience of the nuclear power plant and other operators shall be evaluated whether all hidden failures related to nuclear safety and potential precursor events *are identified and to detect any tendencies showing decrease of safety performance or safety margins.*

4.14.1.0300. During the analysis and evaluation of operating experience special attention shall be paid to investigate the safety related nonconformities and events experienced during operation including maintenance, repair, inspections and reviews–, to evaluate the severity of their actual and possible consequences and to define the necessary measures to avoid any similar nonconformities.

4.14.1.0400. The changes in external effects and site characteristics, especially the fairly quick changes in human activities and related parameters, such as the demographic distribution, built environment and industrial activities, shall be monitored throughout the lifetime of the nuclear facility and shall be regularly evaluated in order to prevent the increase of risks.

4.14.1.0500. The licensee shall appoint suitable personnel to execute programmes according to Section 4.14.1.0100, to distribute new information of nuclear safety importance and if possible for the development of action recommendations. The most significant observations and trends shall be reported to the top management of the licensee.

4.14.1.0600. The organization responsible for the evaluation of operating experience and investigation of events shall receive appropriate training and resources. Their work shall be supported by the management.

4.14.1.0700. The licensee shall ensure that the results are produced, the conclusions are deducted, the measures are implemented, good practices are contemplated and appropriate and timely corrective actions are executed to prevent recurring problems and developments that are unfavourable to nuclear safety.

4.14.1.0800. The licensee shall regulate the content, extent and methodology requirements of the collection, analysis and documentation of operational data and experience. Investigation methods shall contain analysis methods of human factors.

4.14.1.0900. Information shall be stored in a way that the appointed personnel can easily access, systematically search, filter and evaluate the information.

4.14.1.1000. The list of identified safety issues shall be continuously updated together with solution methods and planned measures.

4.14.1.1100. The list and deadlines of planned measures shall be continuously monitored by the management of the licensee. These measures shall be updated with current experience to the necessary extent.

4.14.1.1200. Information regarding operating experience shall be made available to the relevant personnel and shared with the competent national and international organizations.

4.14.1.1300. The licensee shall keep contact to the necessary and possible extent with the organizations that participated in the design and installation in order to provide operating experience feedback and request advice if necessary.

4.14.1.1400. Operating personnel shall report events of nuclear safety significance, reportable events and near miss situations related to the nuclear safety of the nuclear facility according to the relevant regulatory documents.

4.14.1.1500. The licensee shall provide investigation and analysis tasks required for event assessment and event report preparation.

4.14.1.1600. In case of events of nuclear safety importance a preliminary investigation shall be carried out without delay, but within 5 days, at the latest, to determine whether immediate actions are required.

4.14.1.1700. The scheduling of event investigation shall be in compliance with the significance of the event.

The investigation shall:

a) determine the sequence of events,

b) include the comparison of the event to other previous, similar, national and international events,

c) evaluate its effect on safety, the actual and potential consequences,

d) evaluate the activities of the personnel and management, the suitability of regulated processes and procedures,

e) define discrepancies,

f) include the definition of the direct, contributory and root cause, also *g) define corrective actions in order to restore nuclear safety, to prevent the recurrence of the event and where necessary to improve nuclear safety.*

4.14.2. Utilization of operational data and experience

4.14.2.0100. Such a process shall be developed, which ensures that the operating experience related to events at the nuclear facility and other nuclear facilities are utilized in the training programme of the employees.

4.14.2.0200. The determination, execution and follow-up of corrective actions that prevent the recurrence of the same and similar events shall be regulated.

4.14.2.0300. New data, scientific achievements, and reports of operating experience of other nuclear facilities shall be evaluated and utilized throughout the life-cycle of the nuclear facility.

4.14.2.0400. The collection of operating experience and analysis of safety indicators and trends shall be performed in a way that the resulting data can be utilized for the planning of in-service inspections, replacements and reconstructions of systems, structures and components important to nuclear safety while taking into consideration their designed and expected remaining lifetime.

4.14.2.0500. Operating experience shall be considered to better specify the input data for the probabilistic nuclear safety analysis.

4.14.2.0600. Safety indicators of the operation of the nuclear facility shall be regularly evaluated and if required corrective actions shall be defined.

4.14.2.0700. Operating experience shall be considered for the reviews of the operational documentation.

4.14.2.0800. The effectiveness of the operating experience feedback process shall be regularly reviewed based on performance criteria and documented within the self-assessment programme of the licensee or as an assigned independent review.

Relevant point:

(c) the obligation of the licence holder to report events with a potential impact on nuclear safety to the competent regulatory authority;

Answer:

c) Annex 1 to Govt. Decree No. 118/2011 (VII. 11.)

Nuclear Safety Code Volume 1: Nuclear safety authority procedures of nuclear facilities

1.7 REPORTING OBLIGATION OF THE LICENSEE

1.7.4 Event Reports

Event reports during the construction period of the nuclear facility

1.7.4.0100. The event reports of detection of significant failures and noncompliances during the design and construction works – including deviations in the management system that may result in non-compliances – shall be presented to the nuclear safety authority within 8 days of the detection. The event investigation report shall be submitted within 60 days following the detection to the nuclear safety authority.

1.7.4.0200. The content requirements of the investigation report:

a) the time and circumstances of the detection, the reporting person;

b) the presentation of measures implemented to prevent the use of unsuitable product, service, process, location, labelling before the execution of corrective actions;

c) description and safety evaluation of non-compliances;

d) immediate actions performed by the initiator or others in order to mitigate the effects of the non-compliance;

e) possibilities to improve the non-compliance;

f) verification that the required safety margins exist;

g) determination of design modifications necessary due to the non-compliance; also

h) determination and deadlines of the necessary corrective and preventive measures.

Event reports during the commissioning, operation and decommissioning of the nuclear facility

The scope of reportable events

1.7.4.0300. The licensee shall submit an event report for all reportable events that occurred at the nuclear facility to the nuclear safety authority. The scope of reportable events shall be defined *a) as part of the construction licensing procedure,*

b) as part of the commissioning procedure,

c) as part of the operation licensing procedure, and

d) as part of the final termination and decommissioning licensing procedure in a decision issued by the nuclear safety authority.

If necessary, the nuclear safety authority may change the scope of reportable events at the conclusion of the Periodic Safety Review or through procedure initiated ex officio.

1.7.4.0400. A guideline shall contain the recommendations of the detailed contents of event reports.

INES classification, coordination and notification obligations

1.7.4.0500. All events shall receive INES classification. The licensee shall propose the rating and forward it to the nuclear safety authority via fax. The final classification shall be determined by the nuclear safety authority.

1.7.4.0600. The nuclear safety authority shall be notified of the INES classification of a reportable event within 16 hours, at the latest after the occurrence of the event, or if the event was detected later than it occurred after the detection of the event.

1.7.4.0700. The International Atomic Energy Agency shall be notified within 24 hours from the occurrence or detection of the event of events rated to Level 1 or higher. The notification is the responsibility of the nuclear safety authority. The licensee shall provide the necessary information and the English version of the INES event form to the nuclear safety authority within 20 hours following the event.

1.7.4.0800. The public shall be notified of events of the Level 1 or higher within 24 hours. Of events rated 0 or below the public shall be informed regularly. The licensee is responsible for the notification after consultation with the authority. The licensee shall submit the text of the INES Level 1 or higher event report to the nuclear safety authority within 20 hours from the occurrence or detection of the event, before the public is notified.

The method of fulfilling reporting obligations

1.7.4.0900. The licensee shall fulfil its event reporting obligation according to the following:

a) a promptly reportable event shall be immediately reported by phone to the nuclear safety authority but not later than within 2 hours from its occurrence, or if the event was not detected at the time of occurrence within 2 hours from the detection,

b) an event that is not a promptly reportable event shall be reported by phone to the nuclear safety authority not later than within 14 hours from its occurrence or if the event was not detected at the time of occurrence within 14 hours from the detection,

c) INES classification shall be reported within 16 hours,

d) the event shall be reported in writing to the nuclear safety authority within 16 hours from the occurrence of the event,

e) the event investigation report shall be submitted to the nuclear safety authority within 45 days following the occurrence or detection of the event. 1.7.4.1000. The written report according to Section 1.7.4.0900 d) shall include the short description of the event, the developed operational conditions, the executed and planned measures as well as their expected success and probable effects, and the preliminary safety assessment of the event.

1.7.4.1100. The deadline of the event investigation report according to Section 1.7.4.0900 e) may be extended by the nuclear safety authority based on a substantiated application.

The nuclear safety authority investigation and assessment of reportable events

1.7.4.1200. The nuclear safety authority shall assess the reported events based on the information available at the time of notification and it shall make a decision to:

a) investigate and evaluate the event based on the investigation report of the licensee,

b) investigate and evaluate the event based on the information continuously provided by the licensee and if necessary to perform on-scene inspection, or

c) investigate and evaluate the event independently of the investigation of the

licensee in an on-scene nuclear safety authority investigation.

1.7.4.1300. During the on-scene investigation the nuclear safety authority may interview the involved personnel and their managers, may perform walk-downs, and request a reconstruction of the event sequence.

1.7.4.1400. The licensee shall ensure appropriate conditions and circumstances for the execution of the nuclear safety authority investigation. For this purpose it shall provide the available information related to the event and, if reasonably feasible and necessary, evidence shall be presented to the nuclear safety authority.

1.7.4.1500. The licensee together with the nuclear safety authority shall select events of which a report is to be prepared in the agreed form and content in Hungarian, English or both languages in order to promote information share among international nuclear safety authorities. The nuclear safety authority shall inform the licensee of the information received through international forums to ensure utilization of external experience.

Relevant point:

(d) arrangements for education and training, in accordance with Article 7.

Article 7: Member States shall ensure that the national framework requires all parties to make arrangements for the education and training for their staff having responsibilities related to the nuclear safety of nuclear installations so as to obtain, maintain and to further develop expertise and skills in nuclear safety and on-site emergency preparedness.

Answer:
d) Govt. Decree No. 55/2012

Section 3,4 and 5

furthermore

Annex 4 of Govt. Decree No. 118/2011 (VII. 11.)

Nuclear Safety Code Volume 4: Operation of nuclear power plants

4.4 THE ORGANISATION OF THE LICENSEE

4.4.1 Requirements for Employees

4.4.1.1000. In accordance with the comprehensive training policy of the licensee, the programme for the professional preparation of the operating personnel, verification of preparation, regular refresher trainings, and periodical checks of preparedness shall be recorded in writing and periodically reviewed.

4.4.1.1100. The training and refresher training programme, the managerial behaviour, leading by example, support and expectation shall ensure that the operating personnel at every level of the organization comprehend the primary importance of nuclear safety, and are able to properly fulfil their duty in the case of TA1-4 and TAK1-2 operating conditions, in accordance with written operating and emergency operating instructions and accident management guidelines.

4.4.1.1200. The training and refresher training programme shall include both theoretical and practical training, specifically on the simulator and at the work location.

4.4.1.1300. The applied simulator shall ensure the efficient practice of the use of normal operating and emergency operating instructions and the cooperation of operating personnel.

4.4.1.1400. The refresher training shall include operational experience and modifications.

4.4.1.1500. The annual refresher training of operating personnel who temporarily or permanently work in the control room shall include at least five days of simulator training.

4.4.1.1600. The maintenance and technical support personnel shall receive training for activities that are to be performed by them and which are critical to nuclear safety.

4.4.1.1700. In the training programme, special attention shall be paid to measures to be executed under TA3-4 and TAK1-2 operating conditions. The following shall be included in the training programme:

a) training for the inspections required after a service level earthquake and the action plan for a safety earthquake, as well as the periodical practice of the execution of the action plan, and

b) practice of transferring from emergency operating instructions to accident management guidelines and practice of severe accident management.

4.4.1.1800. For job positions important to safety

a) appropriate specific training programmes shall be developed based on the continuous survey of training needs;

b) the training organization shall possess the necessary resources, tools and facilities;

c) trainings shall be performed by instructors having appropriate knowledge, and the performance of the instructors shall be checked;

d) the efficiency of training shall be regularly measured; and

e) the suitability of operating personnel shall be continuously verified and the experience gained during inspections shall be considered during the composition of refresher trainings.

4.4.1.1900. A systematic approach shall be applied to training. The composition of a training programme shall include the following development phases for different job positions:

a) analysis: defining the training needs that are based on the knowledge requirements necessary to fulfil the specific job position;

b) planning: defining the training objectives that ensure the acquirement of

knowledge;

c) development: preparation of training materials necessary to fulfil the training objectives;

d) implementation: systematic education of training materials;

e) evaluation: the data process of all previous phases in order to improve and correct the training programme.

4.4.1.2000. The training documentation shall include the data relevant for the training programme and the performance of instructors. The efficiency of trainings shall be measured, and the management shall be regularly informed thereabout.

4.4.1.2100. The training programme shall include the management, operating personnel, trainers and instructors who shall also be aware and comprehend the authority requirements in order to fulfil regulations in a timely manner with appropriate measures.

4.4.1.2200. The licensee shall develop and continuously update an individual registry which contains the training, the results of knowledge assessment, and required exams for personnel employed in work positions important to safety.

4.4.1.2300. The personnel employed in work positions essential to safety shall have a valid license for that work position as defined in Section 1.8 of Annex 1. The licensee shall develop a procedure on how to obtain and renew a specified licence. Documented criteria shall be applied to determine whether the given employee has the knowledge and suitability to be granted that licence.

4.4.1.2400. Operating personnel shall possess basic knowledge on nuclear safety, radiation protection, fire protection, on-site nuclear emergency response and industrial safety.

Appendix 7: Answers to the questionnaire – Netherlands

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

Revision R2

September 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: ANVS

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

Answer: the ensurance of nuclear safety and its continuous improvement has already been established more than 30 years ago. After Tsjernobyl several major steps were taken to evaluate the safety of NPP's with a lot of impact: operational safety was checked by the first OSART mission and overall safety was checked by what can be called the first real periodic safety review. The OSART mission led to major upgrades in the organization and number of staff at the NPP Borssele. The PSR contained already the elements that are very important for the continuous improvement of safety, but also for the attempt to look for reasonable improvements in the existing plant compared to current regulations and more advanced designs. In 1991 this led to a so-called backfitting paper that contained these important parts of the improvement philosophy. Next to earlier versions of IAEA standards on PSR this document was a leading guidance for future PSRs. It also supported the development of the IAEA standard on PSR. The first PSRs led to a large safety modification programme in the 1990ties of the last century. Measures like PARs, filtered venting, SAMGs were introduced at this early stage already. Subsequent PSRs and stresstest led to smaller, but still meaningful improvement programmes, one of the latest being the realization of In-Vessel-Retention. This is a good example of the implementation of the safety philosophy.

Today larger and smaller IAEA Missions are an established practice for the NPP and RRs. Examples are OSART every ten years and IPSART, SALTO, ISCA as smaller in between ones for the NPP. For the larger RR (HFR) INSARR is every five years and the smaller ones ISCA and SCO (SALTO for RR). The smaller RR will have its second INSARR.

Today also the application of the PSR instrument has been widened to all nuclear installations.

Part of the philosophy is also to incorporate an update regulations based on the IAEA standards and other insights. As member of WENRA from the beginning the Safety Reference Levels and the Fukushima updates have been implemented. They are in many ways stronger than the IAEA standards.

In addition to this emphasis has always been on a well-functioning OEF system.

When EU decided to do the stresstest after Fukushima, in the Netherlands it was decided to do this for all nuclear installations.

It is important to note that of course most regulatory efforts have always been and will always be given to the existing NPP.

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- please provide concrete examples of more demanding objectives and/or demonstration.

Answer: Yes. The more demanding objective of Article 8a is implemented directly in legislation in the Netherlands, in article 6 of the 'Regeling nucleaire veiligheid kerninstallaties' (Ministerial Order on Nuclear Safety). Additionally, the Netherlands published in 2015 the guidance document 'Dutch Safety Requirements for Nuclear Reactors', to be applied to new nuclear reactors during licensing in the Netherlands and used as a reference for safety evaluations of existing reactors. This guidance is also to be applied with a graded approach to new research reactors and again as a reference for existing research reactors.

Relevance of differences: the Dutch Periodic Safety Review (PSR) approach for NPP includes a comparison of the design of the existing plant with modern designs. For example the first genuine PSR was in the early 90-ties of the last century and compared the NPP with the newest German design (Konvoi). This has resulted in a major safety improvement program. The idea behind the paragraph 2 of article 8a according to the Netherlands is to analyse if measures can be implemented to (partly) close the gap with a new plant for a reasonable cost versus the amount of safety improvement (measure of reaching the objective). The related recent WENRA paper describes this in a good way.

Examples of more demanding objectives for new installations are:

- *The Ministerial Order on Nuclear Safety states in article 6 that new installations (build after 14-08-2014) should prevent early and large releases. Older installations should use this requirement as a 'reference and assessment criteria' for the PSR.*

- *The DSR states that new nuclear reactors should have a core melt frequency < 10^{-6} per year.*

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

Answer: In 'besluit kerninstallaties, splijtstoffen en ertsen' (Decree on Nuclear Installations, Fissile materials and Ores), radiological acceptance criteria for 'design basis' and for DECs are defined. Design basis criteria are for the design basis events a maximum allowed effective dose outside the installation depending on the probability of the event:

	Maximum allowed effective dose	
Probability of event (per year)	*Age ≥16*	*Age under 16*
≥ 10^{-1}	*0.1 mSv*	*0.04 mSv*
≥ 10^{-2} & < 10^{-1}	*1 mSv*	*0.4 mSv*
≥ 10^{-4} & < 10^{-2}	*10 mSv*	*4 mSv*
< 10^{-4}	*100 mSv*	*40 mSv*

*For DEC, requirements exist for 'beyond design-basis' events (including DECs). First a maximum individual risk for a fictive person residing permanently and unprotected outside the facility of <10^{-6} per year, that that person would die from a beyond design-basis event, including stochastic effects over a long period of time (50 years). Second a maximum group risk, that the probability that at least 10*n persons die of deterministic effects due to a beyond design-basis event of no more than $10^{-5}*n^2$ per year (for n≥1).*

When showing compliance to the above dose-limits, it is not allowed to take credit for preventive measures 'evacuation' and 'sheltering'.

Additionally, specifically for the NPP, the 2014 WENRA reference levels for existing NPPs, and the IAEA NS-R-1 (safety of nuclear power plants: design) are attached to the license of the NPP as NVR (nuclear safety rule). These documents contain multiple articles related to design and DEC.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

Answer: Yes. In the previously mentioned 'Dutch Safety Requirements for Nuclear Reactors' for new NPP and RR requirements are included related to 'postulated core-melt accidents' (DiD level 4). For these accidents, no evacuation should be needed outside 3 km of the installation, and no sheltering outside 5 km of the installation. Additionally, probability of accidents with core melt should be <10^{-6} per year.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Answer: In the previously mentioned 'Dutch Safety Requirements for Nuclear Reactors', for new NPPs and RR it is required that external hazards up to a probability of 10^{-4} per year should be taken into account in the design. Credible combinations of hazards should be taken into account as well. These requirements are also used as reference for existing reactors

Additionally, as was mentioned before, the WENRA RL for existing reactors, attached to the license of the NPP, contain specific safety objectives related to natural hazards in Issue T.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

*Answer: See answer to 3.2 (group risk) and 3.3. 'Large' releases are first limited through the requirement of the maximum group risk in the Bkse, that the probability that at least 10*n* *persons die of deterministic effects due to a beyond design-basis event of no more than $10^{-5}*n^2$ per year (for n≥1). This requirement holds for all nuclear installations and demonstration of compliance is shown using level 1, 2 and 3 PSA.*

Additionally, in the previously mentioned 'Dutch Safety Requirements for Nuclear Reactors' requirements are included related to 'postulated core-melt accidents' (DiD level 4). For these accidents, no evacuation should be needed outside 3 km of the installation, and no sheltering or iodine prophylaxis outside 5 km of the installation. These requirements can then directly be translated to dose criteria using the intervention rules in The Netherlands for accidents with nuclear installations.

'Early' releases are defined in the Ministerial Order on Nuclear Safety, specifically article 6.1.c. Should for an event it be expected that off-site emergency measures are required (expected dose limits exceed the limits of for these measures), enough time should be available to execute those measures.

It has to be justified by the applicant that this is the case for the local circumstances.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Answer: Timeliness of implementation in the first place can be determined by license conditions. E.g. for the NPP after the PSR the measures shall be implemented within 5 years, unless this is not reasonable. In general after a PSR project the licensee is required to send an improvement plan with a time schedule, in agreement with the regulatory body. Measures shall be implemented as soon as reasonably achievable. This also happened after the stresstest. Through a mechanism of periodic progress reporting followed by supervision activities the regulator verifies if the licensee is acting according to the plan. Delays shall be justified to the regulator and maybe allowed.

Potential safety improvements are identified by the license holder. Whether possible improvements are reasonably practicable, all benefits and drawbacks are taken into account according to a risk matrix. This includes information from the PSA to evaluation safety gain (if *available), the monetary cost of the improvement, and possible risks for workers during implementation (e.g. expected dose for workers during implementation). The risk matrix that is is used by the NPP has been published and can be found via the following link:*

https://www.google.nl/url?sa=t&rct=j&q=&esrc=s&source=web&cd=1&ved=2ahUKEwjxn53fxrjhAhUKKVAKHSVnC40QFjAAegQIAxAC&url=https%3A%2F%2Fwww.autoriteitnvs.nl%2Fbinaries%2Fanvs%2Fdocumenten%2Fpublicatie%2F2016%2F04%2F06%2Fconceptueel-verbeterplan-kernenergiecentrale-borssele%2F7.1-conceptueel-verbeterplan-10eva13.pdf&usg=AOvVaw0RaBxx2AA1bD3LMsmG2Is5

Also the largest RR works with a risk matrix approach.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

Answer: the national approach contains the following elements

- *periodic safety reviews at least every 10 years*
- *operating experience feedback (licensees and ANVS both have their sources; use of IRS, IRSRR; membership of groups of licensees like German VGB and WANO; RR have their own relations; ANVS has initiated KWU Regulators Club; NPP has comparable group with AREVA*
- *IAEA-standards are amended into NVRs and can be attached to the license as condition; ANVS has written policy to evaluate and update regulations every 5 years; WENRA-RL are implemented*
- *IAEA Peer review missions to NPP and RR are (regularly) used like: OSART, INSARR, ISCA, SALTO; NPP is member of WANO that obliges to have WANO missions every 4 years.*
- *Participation in international R&D programmes*

On a high level, requirements on continuous improvement are embedded in the Ministerial Order on nuclear safety, specifically article 11, among them the requirement on 10-yearly periodic safety reviews. Continuous improvement and extensive 10-yearly periodic safety *reviews have been a staple of Dutch regulation for decades, with the first periodic safety review at the NPP finished beginning 1990's, leading to extensive safety improvements being implemented. See also the answer on question 2.*

Additional information is included in the 'Handreiking continu verbeteren van nucleaire Veiligheid' (guidelines on continuous improvement of nuclear safety, https://www.autoriteitnvs.nl/documenten/publicatie/2015/7/6/handreiking-continu-verbeteren-van-de-nucleaire-veiligheid). This includes extensive guidance on periodic safety reviews, use of operating experience feedback, IAEA peer review missions, etc. In addition a lot of information can be found in the National report on the Convention on Nuclear Safety from 2016: link https://www.autoriteitnvs.nl/binaries/anvs/documenten/rapporten/2017/04/25/convention-on-nuclear-safety-7th-review-meeting-%E2%80%93-2017/convention-on-nuclear-safety-7th-review-meeting-2017_Country+Review+Report+for+the+Netherlands.pdf

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Answer: amongst others the WENRA Safety Objectives for New Reactors (2010) and IAEA SSR-2/1 as well as Fukushima lessons learned were used to develop the Dutch guidance on the Safety of New Nuclear Power Plants (DSR) that has been published in 2015. The development was triggered by the new build plans (NPP and RR) that were published in 2009/2010. After Fukushima in 2011 only the new built plan of a RR (PALLAS) is still going on. The document contains an annex dealing with its application for research reactors, using a graded approach. The document is being used for the new RR (PALLAS) and to a certain extent for a modification of the HOR (Delft, 2 MW). The DSR shall also be used as a reference for the existing NPP Borssele during its upcoming periodic safety review (evaluation phase 2021-2023). In the regulatory framework NVRs, Dutch nuclear safety requirements and guides *based on the IAEA equivalent standards, exist for NPP Borssele. They are attached to its license as a condition. In 2017, after the national implementation of the EU-Directive on nuclear safety, a project started to consistently develop NVRs for all nuclear installations and modernize the NVRs for the NPP. The realization is planned in 2020 (before the PSR of the NPP). By then e.g. SSR-3 (version 2015), SSR-2/1 and SSR-2/2 in their actual version will have been adopted as NVRs for use in the Netherlands.*

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

Answer: the most important improvements to improve the safety of the NPP have been carried out during several PSRs (last evaluation 2011-2013) and the post-Fukushima stresstest (2011). Amongst others the WENRA safety objectives for new reactors and newer reactor designs like EPR were reference for these PSRs. The next PSR will be 2021-2023. This PSR will have amongst others the newest IAEA requirements as a reference. For details about implementation of NPP improvement measures we refer to the reports to the CNS and de latest update of the stress test improvement plan, both published on the ANVS website and/or ENSREG website. The NVR-NS-R-1 that is currently in the license of the NPP was adopted at the time that insights were available and partly included from the SSR 2/1 that was under development. With the implementation of the measures from the last PSR and stresstest (including In Vessel Retention) no further major improvements are expected to be necessary.

Research reactors HFR and HOR also undergo PSRs every 10-years, taking account of modern standards for RR (and sometimes for NPP). The last ones at HFR/HOR were in 2011/2012. On top of that they also underwent a post-Fukushima stresstest. HFR undergoes an IAEA full-scope INSARR mission every 5 years. The last mission was in 2016. The follow-up is planned in April 2019 (report: https://www.eerstekamer.nl/overig/20170131/iaea_rapport_insarr_missie_bij_de/document). In 2012 HFR created an integrated plan of improvements from INSARR, PSR and CSA which was agreed to be implemented in 2012-2016. Also HOR implemented its improvements in the same timeframe.

In the last years both research reactors have increased efforts in ageing management. The HOR completed the AMP in 2016. The HFR is implementing an Asset Integrity Plan, part of which is the AMP. The completion is expected within a few years. HFR and HOR have introduced PSA L1, L2 and L3 recently.

The following improvement measures are illustrative for these research reactors:

HFR (45MW)

- *Vacuum breakers and additional piping, to practically eliminate core damage when a LOCA occurs*
- *Improvement of pool injection valves to improve natural circulation*
- *Remote monitoring system*
- *Alternative shutdown system (Cadmium plate)*
- *Replacement of reactor hall crane*
- *Improved fire hazard analysis*
- *Revisiting seismic analysis, seismic measurements installed*
- *Increased leak detection, flood protection and remote operation of valves*
- *Improved and alternative ways of (mobile) water and electricity supply*
- *Diesel supply during flooding*
- *Improved safety classification and protection of classified SSC's*
- *Creation, extension and improvement of function recovery and accident management procedures*
- *Renewal of the Emergency Preparedness Plan and Organization, taking into account extreme external hazards*
- *Replacement of nuclear safety channels to improve reliability op operation*
- *Extension of secondary water outlet into the sea to improve protection against flooding of the site*

HOR (from PSR and stresstest)

- *Check drop of control rods at expected earthquake level*
- *Connect containment ventilation system to emergency power system*
- *Prevention of rain-accumulation roof control room*
- *Addition/modification/testing of procedures and organisation mainly in the area of Emergency Preparedness*
- *Increased and improved inspection of relevant SSC's*
- *Establishment of AMP according to IAEA SSG-10*
- *Establishment of IMS according to IAEA GS-R-3*
- *Analysis of possibility for a diverse and independent shutdown*

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

Answer: for the existing reactor NPP Borssele the requirements on PSA are included in the license conditions, where NVRs (modified adopted IAEA requirements) for siting, design and operation are included. This included also the NVRs on verification of safety and the NVR-SSG-3 and NVR-SSG-4 on level 1 and 2 PSA. In the regulatory framework of the Netherlands the effects to the people and environment have to be compared with dose and risk limits. Therefore already since the 1980-ties a PSA L1, L2 and L3 was developed. Later in the license a requirement was included that the PSA should be a living PSA and kept state-of-the-art. Also we have implemented the 2014 WENRA Reference Levels.

For new reactors the guidance document VOBK (Dutch Safety Requirements for new reactors, 2015) exists. This document requires a L1, 2 and 3 PSA for NPPs and RR.

The two existing research reactors have recently developed PSA L1, 2 and 3. Although this was not formally required it was the result of PSR and/or safety upgrade programmes.

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

Answer: a requirement on the level of detail per stage in case of a new reactor is not given in the VOBK. In the VOBK the scope etc….will be dealt with case by case.

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

Answer: see answer 5.1. Scope should be state of the art, all plant states. External events to be included, depending on the situation (to the extend practicable).

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Answer: includes spent fuel storage. No further specifics (see 5.1)

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Answer: Not applicable.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

Answer: PSA applications are: e.g. risk monitor, justification of modifications, optimization of the Technical Specifications.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Answer: for the existing NPP in the license it is required to update yearly.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed? See 5.1

- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?

  Answer:

  - *HFR: full scope PSA, covering all operating states, internal events, internal and external hazards and the reactor, the spent fuel pool, and experiments.*
  - *HOR: full scope PSA, covering all operating states, only core*
  - *New reactor PALLAS: full scope, all operating states, fuel storage, internal (fire) and external events*

- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?

  Answer:

  - *HFR: The use of PSA will be optimised decision making for e.g. maintenance, PSR etc..*
  - *HOR: The optimization of Technical Specification and maintenance*
  - *PALLAS (new RR initiative): for the license application, PSA is used to show compliance to risk requirements (core melt frequency, risk outside of installation). Additionally, it is a design aid. In the future, it is expected to be used in aiding draft of OLC's, maintenance, etc.*

- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

  Answer: No.

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

<u>*Answer:*</u>
According to the ANVS Licensing Policy, there is a set of non-binding Safety Guidelines (VOBK & DSR) for new reactors (both NPPs and RRs) that provides regulatory expectations related to technical preconditions, safe design, and operation. Here it can be found (DSR, Chap. 2) that postulated multiple failure events (DiD level 3.b) are considered as part of the DECs according to SSR-2.1.

For the existing reactor KCB (Safety Report - November 2015, p.5-5), DEC without fuel melt is associated at DiD level 4. Accidents within the design base are associated at DiD level 3.

DSR will be used as a reference for existing reactors during (future) PSRs or major modifications.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

<u>*Answer:*</u>
For new reactors in Annex 1 of the DSR, the postulated events are listed and classified according to their respective fundamental safety function: control of reactivity, cooling of the fuel assemblies, and/or confinement of radioactive material. These lists should be used as basis for the safety analysis. Deterministic and probabilistic methods and engineering judgement or combinations thereof are used. For new RR, this is used with a graded approach. The approach for grading is described in a separate part of the DSR.

For existing reactors the PIE-list is evaluated and updated if necessary during the PSRs.

Until now the existing NPP used current IAEA-standards and NVRs, but also information from new reactor projects, experience from the original Vendor, operating experience, WENRA Objectives for New Reactors, WENRA Reference Levels (2008 version). In future the SSR 2/1 rev.1 and SSG-25 will be used, as well as WENRA SRLs (2014 version), that also include the new Issue F (design extension conditions).

Existing Research Reactors use also the current IAEA Standards for Design and PSR. Sofar these were NS-R-4 and NS-G-2.12. In future PSR projects the SSR-3 and SSG-25. They also will be adopted as NVR. WENRA is developing now also SRLs for RR, an ad hoc group which NL is chairing.

ANVS verifies and checks methods used and results of PSR closely. In fact the PSR is from both licensee and regulator side a project with a lot of interactions. It is formally required that the licensee comes up with a detailed plan of the PSR, including the total scope of activities, the current regulations and other designs to be analysed, the whole process and interaction with the regulator, the intermediate documents that have to be sent to the regulator for review, the final report including potential improvements and the plan for implementations. This grossly follows the approaches have have been documented in IAEA documents. For instance ANVS requires a separate report for each Safety Factor, including the one on safety analyses. ANVS uses a TSO (GRS) to support the review and assessment, including of the PIE-list and the analyses themselves. For the NPP since 2009 GRS has, by order of ANVS, developed a Full Scope Analysis Simulator.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer: At the NPP the beyond design PIEs as analysed in the SAR:

- *Station Blackout*
- *Total loss of ultimate heatsink*
- *Total loss of intermediate cooling water system*
- *Possible loss of safety systems in the long term after a PIE.*

Besides DECs, PIEs are included for design basis accidents including the spent fuel pool (e.g., loss of coolant from the spent fuel pool, loss of cooling, load drop, etc.)

Furthermore, for new NPPs, the VOBK contains the events total loss of residual heat removal systems of spent fuel pool, and loss of coolant from spent fuel pool with decrease of filling level below minimum level required for cooling

For RR grading is applied

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?

  Answer: for new reactors for DEC single failure has to be assumed (n+1 redundancy), loss of offsite power, failure of most reactivity-effective control rod assembly, most unfavourable conditions, and similar rules as are common for analysis for design basis events.. For RR grading is applied.

  For existing NPP analysis has to done according to NVR-NS-R1 (design) and NVR-GS-R-4 (safety analysis). These deal with beyond-design and severe accidents. In future the approach has to comply with WENRA SRL version 2014 containing DEC A and B and also as far as reasonably achievable according to VOBK/DSR and SSR 2/1 rev.1.

- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?

  Answer: for new NPPs the DSR/VOBK talks about "additional safety features (SSC's) and accident procedures" (additional to "design provisions").

  Control postulated multiple failure events to limit radiological releases (limits are dose for the public <10 mSv), avert escalation to core melt conditions. Additionally, many specific requirements exist related to effectiveness of barriers, related to fuel storage, reactivity control, confinement, fuel cooling, safety margins, diverse ultimate heat sink, power supply for the SSCs, etc..

  For RR grading is applied (although not for radiological acceptance criteria).

  For existing NPP current WENRA RL (version 2014) applies. EOPs shall be developed. They shall primarily rely on qualified equipment (dedicated is not required).

- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

  Answer: Independent from DiD levels 1, 2 and 4. But they are usually the same systems or working together with systems on DiD level 3a (single failure). 3a and 3b together form DiD level 3, which has to be independent from other levels of DiD.

  For existing NPP this is not required.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

Answer: for new reactors, DEC without fuel melt is DiD level 3b, which is part of DiD level 3. see also the answer at 6.4.

For existing NPP: see answer 6.4

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer: Not applicable.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

Answer: No.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Answer: both for new and existing reactors, deterministic and probabilistic approach is used, together with engineering judgement. If evaluations make it necessary the scope changes during PSR.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer: Yes. For example for the NPP in the SAR:

Basic scenario is the total loss of heat removal combined with a total loss of safety systems with four different options:

1. *Filtered venting available after 48 h*
2. *Same as 1, but no containment isolation*
3. *Same as 3, but with using the containment spray system*
4. *Same as 2, but with containment failure after 48 h.*

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?

  Answer: The same as for DEC without fuel melt. Refer to 6.4. For DECs with fuel melt, realistic assumptions and parameters should be used. Uncertainties of the results shall be considered when assessing the fulfilment of regulatory limits.

- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?
  Answer: for new NPPs DSR/VOBK talks about "complementary safety features to mitigate core melt, Management of accidents with core melt (severe accidents)". These have to be graded for RR.

  For existing NPP: SAMGs, using primarily qualified equipment (no dedicated provisions).

- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

  Answer: for new reactors Yes, for existing reactors No

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Answer: see answer 7.4, second bullet. SSCs to mitigate DECs with fuel melt are on DiD level 4. Requirements for these SSCs are similar to those for DiD level 3b (active parts n+1, physical separation of redundant subsystems, back-up on-site electrical supply, automation if manual actuation not possible,)

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer: not applicable.

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Answer: For the existing NPP these requirements are included in the NVRs for design, the license and Technical Specifications (what has to be tested and frequency). WENRA SRL E7.5, E09.10-09.12, F4.8-4.14 and K3.13 are applicable.

For new reactors the VOBK (https://www.autoriteitnvs.nl/onderwerpen/handreiking-vobk/documenten/publicatie/2015/10/19/handreiking_vobk) gives information. The following circumstances shall be practically eliminated:

- *Core melt under high pressure and direct heating*
- *Accidents caused by rapid reactivity increase*
- *Steam explosions*
- *Hydrogen detonation*
- *Fuel melting in the fuel pool*
- *Core melt with a bypass of the containment leading to large early releases*

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

Answer: the requirements ask for controlling the pressure and temperature changes in the containment during design accidents, with a conservative approach. It is up to the licensee with its vendor to present this within the safety case.

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

Answer: Yes. Testing on a regular basis according the Technical Specifications, which are agreed with the regulatory body.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Answer: Yes: In the DSR it is stated: 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 level of confidence to be extremely unlikely to arise.
Additionally, for the existing NPP it is stated in the NVR NS-R-1 (attached to the license): "PIEs can be of internal or external origin. The nuclear power plant shall be so designed against PIEs that it can be demonstrated in a probabilistic safety assessment that the probability of a large release is not greater than 10^{-7}/reactoryear. Large releases are releases that could lead to doses outside the plant exceeding the acceptable limits for design basis accidents and are generally related to conditions where the integrity of the reactor core is no longer ensured (severe accidents, see § 5.31). They could eventually lead to plant external measures in the framework of emergency preparedness planning. Evidence shall be given that there is no sharp increase of risk just below the probability of 10-7/reactoryear."

Note that ANVS is also participating in the work of WENRA RHWG, which is working on the practical application of Practical Elimination.

Question 9.2: Do you have requirements related to practical elimination?

Answer: Yes. Accidents with core melt which would lead to early or large releases shall be practically eliminated. For new reactors, specific requirements are mentioned in the DSR: core melt sequences involving containment failure and bypassing shall be practically eliminated. Occurrence of rapidly propagating cracks and brittle fracture in reactor coolant pressure boundary shall be practically eliminated. For I&C (RPS) common cause failures on system-level in more than one redundancy shall be practically eliminated. Criticality event in the storage facilities of nuclear fuel shall be practically eliminated. These requirements are a reference for the existing NPP and RR.

*For the existing NPP, specific requirements related to this topic are incorporated in the license through NVR NS-R-1. E.g.: "2.5. "**Technical Safety Objective**: To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to* *ensure that the likelihood of accidents with serious radiological con-sequences is extremely low."*

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

Answer: For existing reactors the DSR requirements are is used as a reference e.g. for periodic safety evaluations. For the existing NPP, there also license requirements. For new reactors (both NPP and RR) it is used as a requirement. For references, answer 9.1 and, 9.2.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

Answer: for the operating NPP reactor, safety features implemented over the past decades include:

- *Added containment isolation valves to the volume control system at the containment penetration (closer than the original ones).*
- *Replacement of the existing main steam and feed water lines inside the containment and annular space (between the inner and outer containment) and partially in the turbine hall by qualified 'leak before break' piping; steam flow limiter at the containment penetration location and guard pipes around steam and feed water lines in the auxiliary building*
- *Catalytic hydrogen recombiners installed.*
- *Filtered containment venting installed.*
- *Added option to open hatches of the installation room from the control room. Without measure, insufficient mixing of atmosphere in installation room could possibly lead to local high hydrogen concentrations. By opening some of the explosion hatches, sufficient mixing of atmosphere is guaranteed.*
- *Added bunkered reserve suppletion systems (reserve injection system) for primary and secondary side, thus increasing redundancy, diversity and spatial separation.*
- *Connection of the bunkered primary reserve suppletion (injection) system to the pressurizer (spray) to make it easier to decrease pressure in the event of an SGTR*
- *Added bunkered reserve decay heat removal system and a reserve emergency cooling water system including deep-well groundwater pumps.*

- *Replacement of the primary power-operated relief valves (PORVs) on top of the pressuriser to improve the Bleed & Feed capability and to improve reliability in the event of ATWS situations (tandem principle). The number of PORVs has also been reduced, thereby reducing the LOCA frequency due to spurious PORV opening (although the reduction in the PORV LOCA frequency is due mainly to the revised staggered pressure set points for opening the valves).*
- *A system for the external cooling of the reactor pressure vessel was installed, to prevent core-melt from leading to failure of reactor pressure vessel. The system foresees in various possible sources for the cooling water.*
- *Design already assumed 10hr autarky. Improvement extends autarky to, in case of high temperature, automatically start room cooling systems to prevent overheating of the installations in the reserve supplementation building and reserve control room (to prevent possibly overheating of diesels or electronics) and start emergency cooling of spent fuel pool.*
- *Added connection for mobile pumps/water supply to directly inject water into the secondary side of steam generator*
- *Added connections for mobile pumps/water supply for direct injection of water into the primary system and into the spent fuel pool*

For existing RR, the most significant features implemented include (see also answer to question 4.3):

HFR (45MW): measures are to prevent possible core-melt scenario's, although for a RR these scenario's typically don't result into releases that would be categorized as 'large' or 'early' due to the low temperature and pressure, large water volume and (compared to an NPP) small power and source term.

- *Vacuum breakers and additional piping, to practically eliminate core damage when a LOCA occurs*
- *Improvement of pool injection valves to improve natural circulation*
- *Alternative shutdown system (Cadmium plate)*
- *Improved and alternative ways of (mobile) water and electricity supply*
- *Diesel supply during flooding*

HOR:

- *For the HOR RR, the stresstest showed given the small power and large volume of water in the pool, core melt is practically eliminated.*

For new RR reactors, design is not sufficiently evolved to give any details. Even then, the design as a whole prevents these type of sequences, not any single safety feature. For new NPP: Not applicable.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

Answer: Practical elimination can be shown based on two principles. Either physically impossible for the conditions to occur or if the conditions can be considered with a high level of confidence to be extremely unlikely to arise with a high level of confidence. This is contained in the DSR. But is also determined by the protection requirements for the environment, see answer to questions 3.2, 3.3, 9.1 and 9.2.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Answer: Yes.

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Answer: The PSRs are firstly required on a high level by the EU-nuclear safety directive, its implementation in the Ministerial Order on Nuclear Safety (article 11.3) and the license every ten years. The evaluation phase covers the last 10 year and after the timely implementation of the improvements the reactor should be as much on a state-of-the-art level as reasonably achievable.

The more detailed approach for the PSR for the existing NPP is defined in the NVR that is included in the license conditions and also in the reference document for the PSR: in principle the current IAEA standard for periodic safety review (SSG-25). This includes the Safety Factors as mentioned in WENRA SRL P2.2. In general compliance with the license conditions shall be verified and comparison with moderns standards, reactortypes and practices. For existing research reactors in practice the PSRs also are carried out according to the same safety standard, but with a graded approach. For the future PALLAS reactor the approach will be the same.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

Answer: see answer to question 10.1.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

Answer: before the start of the PSR project the licensee has to solicit agreement with the ANVS about the PSR plan, including the list of references (e.g. IAEA standards, modern reference plant) to be evaluated. This list of references should include the most recent documents and standards relevant for the installation.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Answer: the ntional approach consists of several elements:

- *Ministerial Order on Nuclear Safety implementing the Euratom Directive on Nuclear Safety.*
- *NVRs for the NPP (Operating Requirements and Guides; Requirements and Guides on Safety Management)*
- *IAEA safety missions, where safety culture is included and recently at the NPP (2014) and the largest RR (2017) the ISCA mission was applied with great success. The PALLAS organization has now some interest to have such a mission for the design stage.*
- *Regular inspections and audits on the safety culture issue (dedicated experts in ANVS available)*
- *Policy of regulator meetings at directors level for NPP and RR, at least once a year organisation and safety culture are addressed*

- *Also the ANVS has a safety culture programme on its own safety culture*

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

Answer: examples of requirements and arrangements are

- *Provisions in licenses, technical specifications and guidance about (official) reporting, documentation and analysis of relevant incidents, internal and external*
- *Reportable incidents are summarized on the ANVS website; information on major incidents is also shared through CNS-reports*
- *ANVS sends an annual report about reportable incidents to the Dutch parliament.*
- *Participation in IRS and IRSRR databases*
- *National INES coordinator is someone at ANVS, INES trainings have been carried out several times together with the nuclear sector. ANVS is member of NEA-WGOE*
- *NPP has incident evaluation group and exchanges information on WANO and VGB level*
- *Also ANVS has an internal group dealing with incidents*
- *ANVS stimulates the RR licensees to have regular exchanges with foreign RR*
- *ANVS receives all German GRS Incident information notices and uses advice from German safety committee (RSK)*
- *ANVS exchanges NPP information in the KWUREG club, WENRA and RR information in a small regional regulators group with Germany, Belgium, Austria.*

Appendix 8: Answers to the questionnaire – Slovak Republic

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

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Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: Urad jadroveho dozoru SR (UJD SR), Bratislava, Slovakia

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Safety Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/EURATOM is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.
- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 GENERAL SAFETY APPROACH (SAFETY "PHILOSOPHY")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/EURATOM would be welcome.

The general safety approach ("safety philosophy") adopted in national nuclear legal framework is expressed in introductory part of the Atomic Act No. 541/2004 Coll., as amended, §3, "Principles of peaceful use of nuclear energy". Particularly, in article 1) to article 5) and article 8) it is required that:

1) Nuclear energy may only be used for peaceful purposes and in compliance with the national strategies, international treaties, by which the Slovak Republic is bound, and in compliance with the legal acts of the European Union and legal acts of the European Atomic Energy Community; for the purposes of this Act, the European Atomic Energy Community is also deemed to be the European Union;
2) Use of nuclear energy for other than peaceful purposes is prohibited;
3) Use of nuclear energy must be justified by benefits outweighing the potential risks of such activities, in particular when compared with other ways of accomplishing the same purpose;
4) In using nuclear energy safety aspects must get preference over any other aspects of such activities. Approach to safety aspects shall be graded according to the type of nuclear installation, nuclear material inventory, radioactive waste and spent nuclear fuel and activities performed thereon;
5) The level of nuclear safety, reliability, safety and protection of health at work and security of technical facilities, protection of health from ionizing radiation, physical protection, emergency preparedness and fire protection, to be achieved when using nuclear energy so as to keep the life, health, the working environment or environment-related hazards as low as reasonably achievable according to the available knowledge, while the exposure limits must not be exceeded. Upon new significant information being obtained about the risk and consequences of use of nuclear energy, the above mentioned level must be reassessed and measures shall be taken as necessary to meet the conditions of this Act;
8) The legal, regulatory and organizational framework for nuclear safety shall be maintained and improved on the basis of operational experience, knowledge gained from safety

analyses of nuclear facilities in operation, technological developments and results of safety research, if available and usable.

The safety objectives are established in §23a, article 8) and 9) of the Atomic Act No. 541/2004 Coll., as amended, in full compliance with Article 8a, paragraphs 1 and 2, of Safety Directive 2014/87/Euratom. The safety objectives are valid for existing nuclear installations and planned nuclear installations (see answer in following Article 3). The safety objectives are transferred into the safety requirements stipulated in the national generally binding legal documents (e.g., acts, degrees, regulatory decisions). A concretisation is provided in guides. Safety requirements are generic and specific, qualitative and quantitative. Safety requirements are based on EU directives, requirements/recommendations of international organisations, WENRA reference levels, etc.

3 SAFETY OBJECTIVES FOR NEW REACTORS AND FOR EXISTING ONES

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- Please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;
- Please provide concrete examples of more demanding objectives and/or demonstration.

The safety objectives are established in §23a, article 8) and 9) of the Atomic Act No. 541/2004 Coll., as amended, in full compliance with Article 8a, paragraphs 1 and 2, of Safety Directive 2014/87/Euratom.

The nuclear legislation (the Atomic Act No. 541/2004 Coll. and other related implementing Decrees) doesn’t differentiate between existing and new reactors. However, the nuclear legislation for nuclear facilities with nuclear reactors in Slovakia is developed for the energetic PWR type reactors. There are no new types of reactor designs or research reactors in the country. It is supposed that achievement of existing safety objectives will be done for new reactors in larger extent, compared to existing ones. For example, it is expected that new designs will provide more extensive application of passive systems, higher level of protection against internal events and hazards, and will give lower values of probabilistic safety targets.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) – Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

The implementation of article 8b of the Safety Directive 2014/87/ Euratom is fully transposed in the Atomic Act No. 541/2004 Coll., as amended, in the UJD SR Decree No. 431/2011 Coll. on quality management system, as amended, and in the UJD SR Decree No: 430/2011 Coll. on nuclear safety requirements, as amended. Safety objectives associated with the different levels of defence-in-depth are stipulated in the Annex 3, Part B, Chapter I., C. of the UJD SR Decree No. 430/2011 Coll., as amended.

Requirements for acceptance criteria stipulated in generally binding legal documents are further developed and concretised in the regulatory guides, namely BNS I.11.1/2013 Requirements for deterministic safety analyses … and BNS I.6.2/2013 Requirements for description of reactor and its design basis. The legislation and guides contain radiological acceptance criteria and technical acceptance criteria. The radiological acceptance criteria and technical acceptance criteria are assigned to different plant states considered in the design such that frequent initiating events have only minor or no radiological consequences and that events that may result in severe consequences are of very low frequency. The criteria cover all plant operational regimes including full power, shutdown and open reactor. They are developed for nuclear power plants including spent fuel storage (pool). The values of acceptance criteria are set up conservatively in order to protect population and environment against radiation and release of radioactive materials from the nuclear facility/ activity. The criteria are qualitative and quantitative. Basic radiological criteria stipulated in legislation are quantitative. The limiting values for radiological risks are established for normal operation and abnormal operation and accident conditions including DBAs, DEC A (without core melt), and separately DEC B (with core melt). The technical criteria are specified for protection of fuel rod integrity, primary coolant pressure boundary, secondary coolant system, and containment. The acceptance criteria are derived from/based on requirements/recommendations of international organisations, design requirements, WENRA reference levels, etc.

The national legislation requires off-site emergency plans for protection of the environment/ population.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

See the answer to the question Q 3.1 - the national nuclear legislation doesn't differ between existing reactors and new reactors.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/EURATOM, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

The term “rare and severe external hazards” is not recognised by the NSD and requires specification.

Requirements for coping external hazards are stipulated in the UJD SR Decree No. 430/2011 Coll. on nuclear safety requirements, as amended, and in the regulatory guide BNS I.4.5/2018 Requirements for nuclear safety of nuclear facilities in relation to natural hazards. These requirements reflect relevant IAEA requirements and WENRA Reference Levels 2014, Issue T. Consideration of design basis external hazards, combinations of external hazards is requested.

The methodology used for the treatment of external hazards consist of evaluation of all external hazards relevant to the site including their combinations, assessment of frequencies of occurrence and corresponding magnitudes, analysis of the site/ and plant response to the external hazards, and protection against them. The characteristics of external hazards are periodically reviewed/ re-evaluated, at least in the frame of periodic safety review.

Question 3.5: Article 8a of the Safety Directive 2014/87/EURATOM requires “preventing accidents and, should an accident occur, mitigating its consequences and avoiding” “early radioactive releases that would require off-site emergency measures but with insufficient time to implement them” and “large radioactive releases that would require protective measures that could not be limited in area or time”. Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

The §23a, article 8 of the Atomic Act No. 541/2004 Coll., as amended, stipulates 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.

The license holders have implemented design provisions and organisational arrangements for the application of defence-in-depth and promotion and enhancement an effective nuclear safety culture. This includes but is not limited to the management of severe accidents and implementation of severe accident management guidelines.

The demonstration of avoidance of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release include deterministic considerations, and engineering aspects such as design, fabrication, testing and inspection of structures, systems and components and evaluation of operating experience, supplemented by probabilistic considerations, taking into account the uncertainties due to the limited knowledge of some physical phenomena.

The demonstration of avoidance of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release includes the following steps:

a) Identification of conditions that potentially endanger the integrity of the containment or allow bypassing of the containment, resulting in an early radioactive release or a large radioactive release;

b) Implementation of design and operational provisions in order to ‘practically eliminate’ the possibility of those conditions arising. The design of these provisions should include sufficient margins to cope with uncertainties;

c) Final confirmation of the adequacy of the provisions by deterministic safety analysis, complemented by probabilistic safety assessment and engineering judgement.

Question 3.6: Article 8a of the Safety Directive 2014/87/EURATOM requires “timely implementation of reasonably practicable safety improvements to existing nuclear installations”. Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

The generally binding legal documents and guides issued by UJD SR specify regulatory requirements and conditions and describe the approach for timely implementation of reasonably practicable safety improvements to existing nuclear installations. One of the approaches used is illustrated by the example of periodic safety review (PSR).

The license holder shall conduct the periodic safety review (PSR) of nuclear installation according to the requirements of the generally binding legal documents. Under the PSR, the license holder will process an overall nuclear safety assessment of the nuclear installation, taking into account all positive and negative findings and their cumulative impact on nuclear safety, and the results of the probabilistic safety assessment. The license holder will assess the safety relevance of the findings, and propose integrated measures to remove identified negative findings and safety improvements. The integrated plan is designed to take into account the links between the changes, the availability of procured services, the time of procurement, the management of changes, etc. The integrated plan will determine the timing, resources and priorities for making changes. The integrated plan and its justification is submitted to UJD SR for assessment as part of the periodic safety review carried out. Measures and dates of measure for implementation are subject of agreements with regulator. However, the license holder during PSR also performs preliminary assessment of safety relevant findings and informs UJD SR, when urgent corrective measures need to be adopted and implemented.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

The legislative requirements for continuous improvement of nuclear safety are stipulated in the Atomic act No. 541/2004 Coll., as amended. Particularly,

The §23 Article 2) f) states that license holder is obliged:

f) During the operation and during decommissioning of nuclear installation, to periodically evaluate, verify and where practicable, continuously, systematically and in a verifiable manner to increase the level of nuclear safety of nuclear installations and conduct periodical, comprehensive and systematic nuclear safety assessment of nuclear installations at least every ten years taking into account the state of the art in the field of nuclear safety assessment and to take measures to eliminate the deficiencies found and their recurrence in the future; this includes also verifying that accident prevention and mitigation measures have been put in place, including verification of the application of defence-in-depth principles.

The §23 Article 2) s) and t) states that license holder is obliged:

s) Create a system for evaluation and storing information relating to feedback from operational experiences so that the employees responsible for the feedback can easily search and evaluate such information at any time;
t) Regularly evaluate and document effectiveness of the introduced system of feedback to fulfil the goals pursuant to sub-article s) based on indicators and criteria specified by the authorization holder or by an independent natural person or a legal person.

The continuous improvement of nuclear safety is assured, above all, by periodic safety reviews performed in 10 years intervals. The requirements for Periodic Safety Review (PSR) are stipulated in the UJD SR Decree No. 33/2012 Coll., on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear installations, as amended. The regulatory requirements are concretized in the regulatory guide BNS I.7.4/2016 Complex periodic safety review of nuclear safety. The Decree and guide are fully based on IAEA guide SSG-25 on PSR and reflect relevant WENRA reference levels. The PSR aims to ensure that the current design basis is maintained and that nuclear safety is enhanced, taking into account aging, operational experience, the latest research results and the developments of international standards. The conducted PSRs result in the Action plan to remove the identified non-conformities and provide safety enhancements.

All the units operated in Slovakia were subject to several international missions, which carried out an independent assessment of safety. Between years 1991 and 2016, there were 35 IAEA missions (site assessment, design assessment, OSART mission, IPSART), 6 WANO missions, and RISKAUDIT and WENRA missions. The missions are invited by regulatory body or utility. Action plans are developed by the licensee to implement the findings from the missions. Their implementation is evaluated by follow up missions.

4 DESIGN AND OPERATIONAL ASPECTS OF THE DEFENCE-IN-DEPTH

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

UJD SR keeps the national legislative framework up-to-date to reflect changes in national and international conditions. UJD SR has transposed the recommendation of the IAEA SSR-2/1 (Rev. 1) and WENRA Reference Levels 2014 into the generally binding legal documents and UJD SR guides, namely the Atomic Act No. 541/2004 Coll., as amended, the UJD SR Decree No. 430/2011 Coll. on nuclear safety requirements, as amended, and other relevant Decrees. The §23a, article 6) and 11) of the Atomic Act No. 541/2004 Coll., as amended, and the Annex 3, Part B, Chapter I., C. of the UJD SR Decree No. 430/2011 Coll., as amended, stipulate requirements for the Defence-in-Depth (DiD), levels of DiD, physical barriers and principles concerned to function of different levels of DiD. The evaluation of adequacy of DiD in the design of nuclear facility and ensuring and maintaining an appropriate level of safety culture of license holder is subject of Periodic Safety Review. This includes reviewing the adequacy of incorporation of DiD in a design of nuclear installation, assessing the degree of independence of each level of DiD, systematic application of safety culture principles, etc.

The processes for continues improvement of safety and Periodic Safety Review led to the preparation and implementation of safety improvements of existing NPP on all levels of DiD. Description of SAM programs is provided in the National Report of Slovakia, Chapters 2.2.1 and 2.3.1. The Post-Fukushima National Action Plan (NAP) included provisions strengthening the level 4 of DiD and reinforced independence between levels 3 and 4 of DiD (illustrations of realised safety improvements are provided in answer to the next Question 4.2).

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

An overview of safety improvement programs is provided in the National Report of Slovakia, Chapters 2.2.1 and 2.3.1. The most noticeable safety improvements at existing nuclear power plants in Slovakia were realized under the activities on seismic upgrading and severe accident management prepared relatively long-time ago and Post-Fukushima National Action Plan (NAP), which summarized findings and conclusion of the stress tests as well as outcomes of peer reviews of stress tests organized by ENSREG in April 2013.

The modifications for severe accident management were for example the installation of passive autocatalytic hydrogen recombiners (PAR), depressurization line for primary circuits, alternate safety cooling and heat sink, strengthening of AC/DC power supplies by mobile devices and by bunkered/hardened diesel generators systems, molten core in-vessel retention system and strategy, I&C for severe accident management, etc. The NAP is now implemented.

The NAP is a live document, which has been regularly revised taking into account the latest technical knowledge and operational experience and reflecting actual status of fulfilment of safety measures.

5 PROBABILISTIC SAFETY ASSESSMENTS

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- PSA level 1
- PSA level 2
- PSA level 3.

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required/ provided at the different stages (licensing, operation…)?

The licensing procedure for the nuclear installation in Slovakia consists of five main phases, that is: siting of the nuclear installation, its construction, commissioning, operation, and decommissioning. Before granting an operating license the UJD SR performs inspections according to the approved schedule of program of individual phases of commissioning the nuclear installation (tests, fuel loading, physical start-up, energy start-up, trial operation).

In Slovakia, the PSA is considered as an integral part of the licensing process of NPP. In the design phase the use of PSA is not prescribed, however the PSA is one of possible solutions how to demonstrate fulfilment of requirements for the design of nuclear installations. The performance of PSA is required for phase of construction, commissioning and operation. In case of construction, the PSA is part of the preliminary safety report which provides evidence for the meeting of the legal requirements on nuclear safety. In other cases of licensing process, the PSA is strictly required for issuing a positive statement for an application of an operating license.

According to the Atomic Act No. 541/2004 Coll., as amended, the license holder prepares and submits to UJD SR in the administrative proceedings (Annex 1, Section C, Letter j) a specific PSA for full power reactor, for low power levels, and for shutdown reactor. The basic requirements on the PSA performance and its use are stated in the Atomic Act. The license holder is obliged to use a PSA aimed at identification, quantification, qualification, and evaluation of key indicators and aspects of nuclear safety and their mutual functioning in order to enhance the level of nuclear safety.

Details on the use, scope, content and method of preparation of PSA level 1 and level 2 are defined in Article 20 of the UJD SR Decree No. 58/2006 Coll., as amended. The application of PSA level 3 is not required by Slovak nuclear legislation.

The Annex 6 letter g) of the UJD SR Decree No. 431/2011 Coll., as amended, stipulates requirements for the quality of nuclear installations. The requirements include probabilistic safety targets and probabilistic safety criteria and their relationship to internationally accepted requirements.

UJD SR concretises probabilistic safety targets in the BNS I.4.2/2017. The same probabilistic safety targets for increasing the safety are specified for existing and new nuclear power plants.

Question 5.2: Do you have requirements/expectations regarding the scope of PSAs? For example, are external events included? Are PSA required/ expected for all plant operating states (normal operation, shutdown…)?

Details on the use, scope, content and method of preparation of PSA level 1 and level 2 are stipulated in Article 20 of the UJD SR Decree No. 58/2006 Coll., as amended. The requirements of generally binding legal documents are concretized in UJD SR guides – BNS I.4.2/2017 on Requirements for PSA and BNS I.12.3/2014 on PSA quality for PSA applications. PSA is developed for all operational modes, significant initiation events and risks, including internal fires and floods, considering extreme climatic conditions and earthquakes. PSA is conducted for reactor facility and spent fuel pool as well. PSA is based on a realistic modelling of nuclear device response. It uses the data relevant to the nuclear installation design and takes into account the permanent control room interventions within the scope of the operating procedures. PSA is developed according to the current proven methodology (e.g., IAEA SSG-3 and SSG-4), taking into account the available international experience.

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Yes, PSA shall cover both the reactor and the spent fuel storage with the same level of details. In order to increase the level of nuclear safety, the license holder shall use a PSA aimed at identifying, quantifying, qualifying and evaluating the core indicators and aspects of nuclear safety and their interaction, taking into account the parameters, extent of suitability and objective limitations of probability depending on the type of nuclear installation (§28 Article 4) of the Atomic Act No. 541/2004 Coll., as amended.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Interdependencies among various nuclear facilities, located at the same site, are taken into account in the PSA model. However, the multi-unit PSA is not legislatively required.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

As we have understood the question Q5.5, the list of PSA applications and conditions on the use of PSA are published in the UJD SR Decree No. 430/2011 Coll., as amended. The license holder uses PSA:

a) To support safety management and decision-making;
b) To identify the need for modifications to the plant and its procedures, including for severe accident management measures, in order to reduce the risk from the plant;
c) To assess the overall risk from the plant, to demonstrate that a balanced design has been achieved, and to provide confidence that there are no "cliff-edge effects";
d) To assess the adequacy of plant modifications, changes to operational limits and conditions and procedures and to assess the significance of operational events;

e) To develop and to validate of the safety significant training programs of the license holder, including training on a full scope representative simulator;
f) To verify that the main contributors to risks are included in the facility's maintenance, inspection and test programs.

When using a PSA, it is necessary to:
a) Define its purpose and scope of applicability in the permit holder's internal decision-making process;
b) Recognize and take into account the probabilistic assessment's limitations and ensure that it is suitable for a specific use;
c) Include systems components in the assessment, including their states and safety functions, that are important from the perspective of assessing changes to test intervals and the allowed period of down time of these systems and components;
d) Ensure that systems and components that were identified in the probabilistic assessment as important to safety are operable and that their importance is documented in the safety report.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Conditions for PSA update are stipulated in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended, and in the UJD SR Decree on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear installations No. 33/2012 Coll., as amended. PSA is regularly re-evaluated during periodic safety review (10-year interval) and always when:
a) An important change to the nuclear facility's design has been implemented;
b) An important change to operating procedures has been implemented;
c) A new significant risk has been found.

UJD SR recommends a 5-year interval for a periodic PSA update in the guide BNS I.4.2/2017 on Requirements for PSA.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed?
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider)?
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

The nuclear legislation (the Atomic Act No. 541/2004 Coll. and other related implementing Decrees doesn't differentiate between existing and new reactors. In addition, there are no research reactors in Slovakia.

6 DESIGN EXTENSION CONDITIONS WITHOUT FUEL MELT

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

The levels of DiD are specified in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended. There are three categories of accident conditions for which basic design requirements are specified:

a) Design basis accident conditions, where the proper functioning of the safety systems ensures that the relevant reference or exposure limits are not exceeded;

b) Design extension conditions without serious damage to nuclear fuel;

c) Design extension conditions with fuel melt (they may include severe accident conditions).

Taking into account the marking of DiD levels according WENRA, the DECs without fuel melt pertain to third level of DiD (3b). The results of safety analyses of Design Extension Conditions without fuel melt (DEC A) shall fulfil the same acceptance criteria, as analyses of Design Basis Accidents (DBA). In difference to the DBA, the DEC A (DiD, 3b) can be analysed with realistic approach. Modified technical acceptance criteria are allowed. Radiological criteria are the same for the DBA and the DEC A.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Initiating events, which lead to DECs, can arise in the event of multiple failures of an equipment or operator not considered in a design, by causing various plausible combinations of incidental events, including internal and external hazards. Events and their scenarios for the release of radioactive substances outside their physical barriers shall cover all operational modes of nuclear installation. They include both the reactor and the spent fuel pool.

The minimal set of the initiating events for DECs without fuel melt is determine in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended. They are concretized in the UJD SR guide on Requirements for deterministic safety analyses BNS I.11.1/2013.

In practice, the initiating events, which lead to DECs, are identified on the basis of plant layout and location, operating experiences, engineering judgment, combination of deterministic and probabilistic assessments and research findings. The applicant uses a recommendations of safety standards (national and international, e.g. IAEA) containing the recommended list of DECs and also uses the plant specific PSA study to verify selected and identify others DEC scenarios which are plant specific and which are not included in safety standards.

Regulatory body checks the list of scenarios used for identification enveloping scenarios of DECs without fuel melt for licensed plant.

During the PSR list of DECs without fuel melt shall be reviewed using both a deterministic and a probabilistic approach as well as engineering judgment to determine whether the selection of design extension conditions is still appropriate.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Yes, the list of DECs without fuel melt includes scenarios that might happen in the spent fuel pool. These scenarios are analysed to verify fulfilment of acceptance criteria, e.g. total loss of cooling systems in the spent fuel pool.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?
- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

What are the rules to analyse DEC without fuel melt?

The rules to analyse DEC without fuel melt are determined in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended. They are concretized in the UJD SR guide on Requirements for deterministic safety analyses BNS I.11.1/2013. Based on these rules, the methodology and basic attributes of accident analyses should be clearly specified (initial and boundary conditions, assumptions and conditions used in the analysis, functioning of SSCs, operator actions, analytical tools). Realistic approach may be used to analyse DECs.

Specific rules include: realistic approach can be used, a single failure of systems, structures and components may not be considered, systems and equipment used by operating personnel shall be operative and must be proven that they can to fulfil safety function even under the DECs conditions for a necessarily long time, an end state need to be defined, which should where possible be a safe state.

Do you require provisions dedicated to the mitigation of DECs without fuel melt?

The general requirements are set out in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended. Provisions dedicated to the limitation of the consequences of DECs with fuel melt are generally the same as for other states of NPPs: control of reactivity, removal of heat from the reactor core and from the spent fuel pool, confinement of radioactive material and regulation and limiting the amount of radioactive substances released into the environment. The implementation of safety approach shall ensure sufficient resources on sustainment of nuclear facility in operation, adequate response immediately after initiating event and relieve the management of nuclear facility during all of the postulated initiating events in design basis as well as in DECs.

Do you expect provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

Not directly, but in general national nuclear legislation requires:

Atomic Act No. 541/2004 Z. z., as amended:

§23a, Article 6

6) Defence-in-depth is a hierarchical system of multiple levels of different technical means and organizational measures designed to prevent deterioration of operational events and maintain the effectiveness of physical barriers located between nuclear materials, spent nuclear fuel or radioactive wastes and workers, the population and the environment during operating conditions and some barriers even under accident conditions.

ÚJD SR Decree 430/2011 Coll., as amended:

§3

8) Failure of classified equipment in any safety class shall not cause failure of the classified equipment in the lower order safety class. Ancillary systems and subsystems that assist classified equipment will be assigned to the appropriate safety class with respect to the inclusion of the related or superior system.

Annex 3, Part B., Chapter I., B.

1) The safety approach shall ensure sufficient resources on sustainment of nuclear facility in operation, adequate response immediately after initiating event and relieve the management of nuclear facility during all of the postulated initiating events in design basis as well as in DECs.

Annex 3, Part B., Chapter I., C.

2) In design of nuclear installation, defence-in-depth must be included so that the design must
   c) Provide multiple means of performing safety functions by ensuring the effectiveness of physical barriers by mitigating the consequences of failures

Annex 3, Part B., Chapter II., C.

5) The design must include the solution of reliable ultimate heat sink from selected installations during normal operation, abnormal operation, design basis accidents and accidents under design extension conditions without significant damage to nuclear fuel; in the case of an accident in design extension conditions with fuel melt, the project can also deal with heat sink in a different way than for other plant states. Ultimate heat sink means the removal of residual heat into the atmosphere or into water, or a combination thereof.
6) The reliability of systems contributing to the ultimate heat sink by transferring, supplying energy or media to ultimate heat sink systems must be achieved, for example, by selecting proven equipment and systems, backing them up, diversity, physical separation, interconnection, isolation.

Defense-in-depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, defense-in-depth ensures that no single human induced event, organizational shortcoming or technical failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability. The independent effectiveness of the different levels of defense is a necessary element of defense in depth.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards, etc.)?

The rules for the safety classification of SSCs according its safety function are specified in §3 and the Appendix 1 of the UJD SR Decree No. 430/2011 Coll., as amended. There are four safety classes defined: Safety classes BT1 (principally for primary circuit boundary), BT2 (principally for Safety Systems) and BT3 (principally for other SSCs important to safety), and BT4 (principally for SSCs assigned for prevention or limitation the consequences of malfunction of other SSCs, which are classified in safety class I to III (BT1 – BT3)). This classification serves for graded approach to the Quality Assurance of the “Classified Equipment”, and specification of SSC redundancy, electrical supply, qualification, etc.).

In general, the SSCs for limiting the consequences of DECs without fuel melt are classified in safety class BT2 and safety class BT3 (of Classified Equipment). They must be seismically qualified.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

The provisions in nuclear legislation in Slovakia do not make any difference between currently operated and newly build nuclear power plants. There are no new reactors currently built in Slovakia.

7 DESIGN EXTENSION CONDITIONS WITH FUEL MELT

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

The term "Severe Conditions" is used in Slovakia for the purpose of full transposition of Safety Directive 2014/87/EURATOM into the national legislation. The term "Severe Conditions" covers both type of conditions DEC A and DEC B of WENRA/IAEA terminology.

There are the following definitions of the terms "DEC with fuel melt" and "severe conditions" in the generally binding legal documents in Slovakia:

- **Severe conditions** are conditions that are more severe than conditions under design basis accidents; such conditions may be caused by multiple failures, such as complete failure of all trains in the safety system or extremely unlikely event.
- **Accident under design extension conditions** is an accident that is not a design basis accident, but has been considered during the nuclear facility's design process according to realistic methodology, and for which radioactive substance release will not exceed stipulated limits; these can be without serious damage to nuclear fuel (DEC A) and with nuclear fuel meltdown (DEC B). Design extension conditions may include severe accident conditions.
- **Severe accident** is a state of nuclear facility involving an event with fuel melting or with release of radioactive substances that requires the implementation of protective measures to protect the public.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

According to requirements of UJD SR Decree No. 430/2011 Coll., Annex 3, Part B., Chapter II., E, 5) DEC with fuel melt conditions events shall be postulated in design. List of DEC scenarios must be determined on the basis of operating experiences, research results, combination of probabilistic methods, deterministic methods and engineering judgment. The list shall be plant specific and relevant for the particular nuclear facility. Generic list of DEC scenarios is provided the UJD SR guide on Requirements for deterministic safety analyses BNS I.11.1/2013. Regulatory body checks the list of scenarios for licensed plant.

During the PSR the scope of DECs with fuel melt is reviewed using both a deterministic and a probabilistic approach as well as engineering judgment to determine whether the selection of DECs is still appropriate.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

DECs with fuel melt events that might happen on the spent fuel pool should (SFP) be developed according to the specificities of the design and the characteristics of the SFP. Framework sequences are:

- Loss of ability to cool nuclear fuel in SFP caused by loss of external power supply (operational and emergency sources) associated with partial or total loss of internal power supplies and loss of residual heat removal;
- Loss of SFP cooling system integrity associated with loss of coolant without availability of emergency cooling systems or with exceeding the capabilities of emergency cooling systems,
- Heavy load drop to spent fuel pool.

Question 7.4: What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?
- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

What are the rules to analyse DEC with fuel melt?

The rules for DEC analyses are determined in the UJD SR Degree No. 430/2011 Coll., as amended, and concretised in the UJD SR guide on Requirements for deterministic safety analyses BNS I.11.1/2013. Based on these rules, the methodology and basic attributes of accident analyses should be clearly specified (initial and boundary conditions, assumptions and conditions used in the analysis, functioning of SSCs, operator actions, analytical tools). Realistic approach may be used to analyse DEC.

The general rules include:

- Specification of methods, assumptions or conditions that may not be too conservative; taking into account of severe accident phenomena, where relevant;
- Initiating event and initial operational regime;
- Consideration of uncertainties and their impact on the results;
- Reflecting the results of the PSA level 1 and level 2;
- Definition of the end state that should be a safe state (conditions) and the duration of operation of systems, structures and components, if it is necessary;
- System failures and operator error;
- Taking into account the layout and location of the NPP, the characteristics of the equipment, the conditions associated with the scenarios and the feasibility of the planned accident management measures;
- Assessing potential in-site and out-site radiological consequences resulting from the DEC (for successful accident management measures);
- Demonstrating, where applicable, that adequate safety margins are available to avoid cliff-edge effects that would lead to unacceptable consequences, i.e. for DEC-B early or large release of radioactive substances.

Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?

Provisions dedicated to the limitation of the consequences of DECs with fuel melt are generally the same as for other states of nuclear facility: control of reactivity, removal of heat from the reactor core and from the spent fuel pool, confinement of radioactive material and control and limiting the amount of radioactive substances released into the environment. The implemented safety approach shall ensure sufficient resources on sustainment of NPPs in operation, adequate response immediately after initiating event and relieve the management of nuclear facility during all of the postulated initiating events in design basis as well as in DECs.

Specifically, the design of a NPP must provide different technical means and organizational measures to protect the integrity of the containment and limit the release of radioactive substances into the environment:

- Avoidance of high-pressure core melt scenarios;

- Suppression of reactivity in the melted corium;
- Retention of the corium within the hermetic zone of the containment;
- Long-term cooling of the corium with heat transfer to heat removal systems from the containment;
- Maintaining the capability of the containment system to contain the radioactive substances (e.g., isolation of the containment shall be possible in DEC, if an event leads to bypass of the containment, severe core damage shall be prevented; tightness of the containment shall not be expressively reduced on reasonably long time after DEC; pressure and temperature in the containment shall be managed; the threats due to combustible gases shall be managed; the containment shall be protected from overpressure).

Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

National nuclear legislation does not require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states, whereas they derive from the independence of levels of defence-in depth (see answer to Question 6.4).

To limit the consequences of DECs with fuel melt is possible to use some of the safety systems, as well as non-safety related systems, additional temporary systems to perform functions other than those originally considered, and under other than anticipated operating conditions to bring the nuclear installation under control or to mitigate the effects of selected events. For multi-units nuclear facilities with nuclear reactors is possible to use available support resources/ measures from other units, provided that the safe operation of these units is not compromised.

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

In general, SSCs for limiting of consequences for DECs with fuel melt are classified in safety class BT3 (of Classified Equipment).

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

There is no new reactors currently built in Slovakia. Nevertheless the list of DECs with fuel melt and provisions to limit their consequences shall be always plant specific and relevant for the particular nuclear facility so this will differ from case to case.

8 DESIGN OF THE CONTAINMENT

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

As defined by the IAEA safety glossary (rev. 2016), confinement means prevention or control of releases of radioactive material to the environment in operation or in accidents. Confinement is typically used to refer to the safety function of preventing the ‘escape’ of radioactive material. Basic requirements for assuring this function are formulated in the Atomic Act No. 541/2004 Coll., as amended.

The Atomic Act defines nuclear safety as: The technical condition and capability of a nuclear installation or a transport facility, as well as the ability of its operating staff, to prevent unauthorized release of radioactive substances or ionizing radiation into the working environment or the environment and the ability to prevent and mitigate the consequences of events in nuclear installations or in the transport of radioactive materials., and requires a high level of nuclear safety, e.g.:

- In the use of nuclear energy, the level of nuclear safety, reliability, safety and health at work and the safety of technical equipment, the protection of health against ionizing radiation, physical protection, emergency preparedness and fire protection must be attained so, that the risk to life, health, work or safety is as low as reasonably achievable, according to the knowledge available, without exceeding the exposure limits. In obtaining new significant information on the risk and consequences of using nuclear energy, this level must be reassessed and necessary measures taken to comply with the terms of this Act.;

- The license holder is obliged to ensure nuclear safety, physical protection, emergency preparedness, including verification;
- The license holder is obliged to identify appropriate emergency procedures and measures in the territory of the nuclear installation, including severe accidents management guidelines or similar instructions, for an effective response to incidents and accidents in order to prevent or mitigate the consequences; … .

(For details see https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/AA_541_2004_en/$FILE/At_Act_541_2004.pdf)

More detailed regulatory requirements related to containment are specified in Annex 3 of the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll., as amended, and conform to the respective WENRA safety reference levels. This legislative document specifies general principles of peaceful use of nuclear energy to achieve such a high level of nuclear safety, reliability, safety and occupational health and safety of technical devices, health protection against ionizing radiation, physical protection, emergency preparedness and fire protection, that a risk of threat to life, health hazards, or the environment respects the ALARA principle, as well as specify a clear general obligation of license holder to ensure nuclear safety, physical protection, emergency preparedness, including its verification, to apply the defence in depth principle, to ensure fundamental safety functions and to establish mandatory procedures for dealing with incidents and accidents, to prepare and carry out preventive as well as mitigation measures for coping with accidents and/or mitigate their consequences.

The Decree on nuclear safety requirements No. 430/2011 Coll., as amended, contains:

A part of general requirements related to basic safety principles, safety functions and characteristics, defence-in-depth, preventing the emergence and development of equipment failures, protection against external events, accidents considered in the design, safety and control systems, approach to tackling nuclear safety, safety functions and safety features, heat removal, protection against external events, etc. Besides other it is required, e.g. that:

Decree on nuclear safety requirements No. 430/2011 Coll. as amended, Appendix 3, Part B, Chapter I., B:

5) To ensure safety during commissioning, normal operation, abnormal operation, design basis accidents, and adequately during accidents in design extension conditions the nuclear facility shall fulfil these fundamental safety functions:
   a) Control of reactivity;
   b) Removal of heat;
   c) Confinement of radioactive materials within the physical barriers;
   d) Control and limiting the amount and type of radioactive substances released into the environment

A part of the Decree is related to specific requirements for NI with a nuclear reactor, i.e. to NPPs. This part is covering specific requirements for a primary circuit, pressure vessel and core, for systems of primary make-up and cleaning, core cooling systems, containment, safety systems, systems of power supply, safety analyses including severe accident analyses, acceptance criteria, fire protection, etc. The Decree covers also general and/or some specific requirements related to engineered systems for protecting the containment (systems for hydrogen management, management of radiological releases, management of containment pressure and temperature, avoiding high pressure melt ejection, etc.), for cooling the molten core (reliable, redundant and backup decay heat removal, reliable ultimate heat sink), etc. The design extension conditions and related requirements for them are explicitly mentioned in all relevant sections. There are explicit formulation like e.g.: in DEC the containment must be isolated, pressure and temperature in the containment shall be managed in DEC, containment degradation by molten fuel must be prevented as far as reasonably achievable, high pressure core melt scenarios must be prevented, the design must include analyses that verify the behaviour of nuclear facilities during design extension conditions, including severe accidents, so that the radioactive releases harmful to the population and the environment are minimized as far as reasonably practicable, acceptance criteria for protection of the containment shall be defined, acceptance criteria for protection of the primary and secondary circuits integrity including allowable pressure, temperature, pressure and temperature transients and internal stresses shall be defined, etc.

Requirements related to operating procedures including emergency operating procedures and severe accident guidelines (EOPs and SAMGs) are in the UJD SR Decree No. 430/2011 Coll., as amended, Appendix 4.

Besides a clear general obligation of license holder to have accident procedures (EOPs) and severe accident management guidelines (SAMGs), this Appendix defines also more specific requirements related to the scope, purpose, content, development, verification and validation, documentation, training, exercising, regular review and update of the EOPs and SAMGs.

Severe accident management (SAM) measures and SAMGs have been implemented in all Slovak NPPs. Various hardware modifications have been completed, to assure success of the accident management strategies for coping with severe accident (e.g. installation of passive autocatalytic recombiners (PAR), SAM valve on pressurizer to depressurise primary circuit, in-vessel melt retention concept by external cooling of the reactor pressure vessel, additional emergency sources of power and coolant supply, etc.).

(For details see https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/Regulation_430_2011.pdf/$FILE/Regulation_430_2011.pdf)

The requirements for specifying quantitative safety goals, covering radiation goals, PSA safety goals and safety criteria, methodology of PSA, requirements for specifying the acceptance criteria for keeping the integrity of the barriers during normal operation, shutdown states, for DBA, DEC and severe accidents, etc., are specified in the UJD SR Decree on quality management system No. 431/2011 Coll., as amended, Appendix 6 – Requirements for the quality of nuclear installations.

(For details see https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/Regulation_431_2011.pdf/$FILE/Regulation_431_2011.pdf)

A list of recommended acceptance criteria for safety analyses, related to the integrity of fuel, primary and secondary circuits, containment and reactor pressure vessel, can be found in the regulatory guide on requirements for deterministic safety analyses BNS I.11.1/2013 (available only in Slovak language).

(See https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/BNS.pdf/$FILE/BNS.pdf)

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

The design of the containment considers large spectrum of loads, i.e. permanent, long-term, short-term, extreme and their possible combinations. Thus, the design pressure and temperature are derived from the thermo-mechanical analyses for the loads caused by pressure, pressure differences and temperature generated by the design basis accident, i.e. by a double ended break of the main circulating pipe in primary circuit, and increased by sufficient margin.

Besides that, an experimental qualification of the containment was performed, checked and confirmed within various international projects and expertise (e.g.: Project PH 2.13/95 – Bubble Condenser Experimental Qualification, Stress Analysis Report Document BC-D-VU-EA-0008, Issue 2 January 1999; PHARE SK/HU/CZ/TS/08 (1998), PR/TS/17 (2003); "Answers to Remaining Questions on Bubbler Condenser" – Activity Report of the OECD NEA Bubbler-Condenser Steering Group, Nuclear Safety NEA/CSNI/R(2003)12, January 2003).

The requirements for design of the containment are specified in the UJD SR Decree on nuclear safety requirements No. 430/2011 Coll. as amended. Example of the requirements is provided below:

D. Containment building system

1) A nuclear facility must be equipped with a containment building system that, when postulated initiating events with leak of radioactive substances and ionizing radiation into the environment occur, it will limit these leaks so that they are lower than established limit leak values, if this function is not provided for by other means.
2) The containment building must be designed so that its required degree of tightness is maintained also during design basis accidents. Aside from this, the ability to mitigate the consequences of selected severe accidents and to limit the escape of radioactive substances into the environment must be taken into account.
3) Pressurized parts of the containment system must be designed with sufficient margin for the highest pressures, under-pressures and the highest temperatures that can occur during design basis accidents.
4) The containment system must consist of a full-pressure enclosure or an enclosure equipped with a pressure and temperature reduction system, of sealing facilities and ventilation and filtration systems that are dimensioned for all postulated initiating events

and that must ensure that permitted parameters are not exceeded even during design basis accidents.

5) Facilities inside the containment building must be designed so that they fulfil their function and so that their effect on other systems, assemblies and components is limited.
6) Insulation materials, sheathing and coatings of systems, assemblies and components inside the containment building must be designed so that their fulfilment of their safety functions is ensured and so that they resist the effects of their environment even during design basis accidents.
7) The containment building and systems, assemblies and components important for it to maintain its containment ability must be designed so that it is possible to
   a) Perform leak tests at design pressure following
      1. The installation of all penetrations and passages,
      2. Repairs;
   b) Prior to commissioning, prove its integrity through a pressure test using a test pressure higher than design pressure;
   c) During normal operation of the nuclear facility,
      1. Perform regular checks of individual assemblies and components of the containment building;
      2. Perform functional tests of individual containment building systems, assemblies and components;
      3. Perform regular leak tests of the containment building at design pressure or at lower pressures that permit extrapolation;
   d) Prevent a reduction of its tightness by flying fragments or pipe whipping.

8) Penetrations passing through the walls of the containment building must be designed so that
   a) Leak tests can be performed,
   b) Regular tests of their tightness can be performed at design pressure independent of leak tests of the containment,
   c) They are protected from the effects of dynamic forces,
   d) Their number is as low as possible,
   e) They all meet the same design requirements as the containment building itself.

…

16) A containment building equipped with a pressure and temperature reduction system must have important supporting systems, assemblies and components backed up to ensure their functionality even during a single failure.
17) It must be possible to isolate the containment building during accidents in design extension conditions. If the incident leads to a bypass of the containment building, its consequences must be mitigated.
18) The tightness of the containment building must not be reduced significantly for a reasonable time following a severe accident.
19) The pressure and temperature inside the containment building must be controlled during a severe accident.
20) The concentration of flammable gases must be controlled during a severe accident.
21) The containment building must be protected from internal overpressure during a severe accident.
22) An active zone meltdown scenario at high pressures must be prevented.
23) Damage to the containment building by molten fuel must be prevented to a reasonably achievable extent.

(For details see https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/Regulation_430_2011.pdf/$FILE/Regulation_430_2011.pdf)

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

Comprehensive series of regular tests, covering tests of leak tightness as well as tests of the strength, are performed regularly for containment and containment penetrations. The scope and intervals of individual tests are specified in the quality plans of NPPs and the plans are subject to approval by the regulator. Testing includes:

a) Local tightness tests of individual components and/or devices of containment boundary and hermetic penetrations, e.g. hermetic penetrations: electrical, piping, pulsed; door sealing, hermetic lining, hermetic hatches and/or manhole covers, etc. (performed during every shutdown for refuelling);
b) Integral tightness tests of hermetical rooms;
c) Control integral tests including leak detection (at the beginning of outage, using overpressure as well as vacuum);

d) Periodic integral test of containment tightness (1 or 2-year cycle);
e) Strength integral test of the containment (10-year cycle).

Several common methods are used for local leak tests: visual inspection, compressed air pressure test, vacuum test. A foaming solution, an ultrasonic detector, a colour imprint test, and the like are used to determine the location of the leaks. Modern methods include identification gas (helium) leak detection method and method of monitoring time dependence of pressure gradients under hermetic and non-hermetic lining. During overpressure integral leak tests, a pressure drop is created on the hermetic and non-hermetic lining due to pressure.

The regulatory requirements related to penetrations are specified in Annex 3 of Decree on nuclear safety requirements No. 430/2011 Coll. as amended, and conform to respective WENRA reference levels. Besides other it is required that:

Penetrations passing through the walls of the containment building must be designed so that:
a) Leak tests can be performed;
b) Regular tests of their seals can be performed at design pressure independently of leak tests of the containment;
c) They are protected from the effects of dynamic forces;
d) Their number is as low as reasonably possible;
e) They all meet the same design requirements as the containment.

Each line that penetrates the containment as part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of a design basis accident. These lines shall be fitted with at least two containment isolation valves arranged in series. Isolation valves shall be located as close to the containment as is practicable.

Each line that penetrates the containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve. This valve shall be outside the containment and located as close to the containment as practicable.

(For details see https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/Regulation_430_2011.pdf/$FILE/Regulation_430_2011.pdf)

9 PRACTICAL ELIMINATION

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

"Practical elimination of a situation (or event, scenario, condition, etc.)" is understood as physical impossibility for the situation, event, scenario, etc. to occur or that the situation (or event, scenario, condition, etc.) can be considered with a high degree of confidence to be extremely unlikely to arise.

As it is stated e.g. in the UJD SR guide on Requirements on deterministic safety analyses (BNS I.11.1/2013, available only in Slovak language): Initiating events and event scenarios derived therefrom, which are physically impossible or are considered to be extremely unlikely with a high degree of confidence, can be considered as practically eliminated and not considered in deterministic safety analyses. The probability of a practically eliminated event occurring should be less than 1×10-7/year (recommended value).

(See https://www.ujd.gov.sk/ujd/WebStore.nsf/viewKey/BNS.pdf/$FILE/BNS.pdf)

Question 9.2: Have you got requirements related to practical elimination?

An explicit request for practical elimination is related to new NPPs and it is formulated in the UJD SR guide BNS I.11.1/2013 (available only in Slovak language). This guide states that: For new nuclear installations, severe accidents involving melting of nuclear fuel (which could lead to large doses of irradiation or large releases of radioactive substances outside of the nuclear installation) must be practically eliminated. In the event that nuclear fuel melting accidents cannot be practically eliminated, such design provisions must be taken to ensure that only territorially-limited and time-limited protective measures are required for the population (no permanent relocation, evacuation only in the immediate vicinity of the nuclear installation, limited sheltering, no long-term restrictions on food consumption), with sufficient time to implement these protective measures.

Thus, the guide provides some more details on how to fulfil the requirement of the Atomic Act, according to which: Nuclear installation must be designed, sited, constructed, commissioned, operated and decommissioned so as to prevent accidents and mitigate their consequences, if they occur, as well as prevent

a) The early release of radioactive substances that would require external emergency measures, with insufficient time for their implementation, and
b) Large releases of radioactive substances that would require spatial and time-wise unlimited protective measures.

Some other recommendations related to practical elimination (based on IAEA and WENRA documents) are implemented in the drafted update of this regulatory guide (expected to be published this year).

Question 9.3a: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

As explained in answer to Question 9.2 above, application of concept of practical elimination is explicitly requested for new reactors. In case of existing reactors, this concept is used mainly for screening of initiating events and/or scenarios needed to be or not to be considered in further analysis.

Question 9.3b: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

As indicated in answer to Questions 4.2 and Question 8.1, various organizational and technical improvements were implemented for coping with severe accidents. Severe accident management (SAM) measures and SAMGs have been implemented in all Slovak NPPs. Various hardware modifications have been completed, to assure success of the accident management strategies for coping with severe accident (e.g. installation of passive autocatalytic recombiners (PAR), SAM valve on pressurizer to depressurise primary circuit, in-vessel melt retention concept by external cooling of the reactor pressure vessel, additional emergency sources of power and coolant supply, etc.).

As an example, safety features of these improvements for SAM valve on pressurizer to depressurise primary circuit can be described as follows:

Management of severe accidents (SA) requires avoid high-pressure core melt scenarios, which would challenge the containment integrity and could result in inacceptable radioactive releases. The depressurisation system will be used to depressurize the primary circuit into the containment. Thanks to this modification, high-pressure core melt scenarios can be considered as practically eliminated. The modification will contribute also to management of containment by-pass scenarios such as steam generator tube rupture and interfacing LOCA, by changing their character into medium LOCA accidents with releases into the containment. Thus, the loss of coolant and potential radioactive releases into environment will be minimised and gradually stopped. The contribution to risk from containment by-pass significantly decreases. Depressurization of the primary circuit will also prepare conditions for reactor cavity flooding and coolant injection into the primary circuit.

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

As indicated above for answer to Question 9.1, it is stated in the UJD SR guide Requirements on deterministic safety analyses (BNS I.11.1/2013, available only in Slovak language) that: Initiating events and event scenarios derived therefrom, which are physically impossible or are considered to be extremely unlikely with a high degree of confidence, can be considered as practically eliminated and not considered in deterministic safety analyses. The probability of a practically eliminated event occurring should be less than 1×10^{-7}/year (recommended value).

However, as drafted in the update of this regulatory guide (expected to be published this year) in identification of DEC with postulated severe fuel damage is stated that: A low estimated frequency of occurrence for an accident with core melting is not sufficient reason for failing to protect the containment against the conditions generated by such an accident. Core melt conditions should be postulated regardless of the provisions implemented in the design. To exclude containment failure, the analysis should demonstrate that very energetic phenomena that may result from an accident with core melting are prevented (i.e. the possibility of the conditions arising may be considered to have been 'practically eliminated').

The event sequences for which specific demonstration of their "practical elimination" is required can be classified as follows:

a) Events that could lead to prompt reactor core damage and consequent early containment failure, such as:
   - Failure of a large pressure-retaining component in the reactor coolant system;
   - Uncontrolled reactivity accidents;
b) Severe accident sequences that could lead to early containment failure, such as:
   - Highly energetic direct containment heating;
   - Large steam explosion;
   - Explosion of combustible gases, including hydrogen and carbon monoxide;
c) Severe accident sequences that could lead to late containment failure:
   - Basemat penetration or containment bypass during molten core concrete interaction;
   - Long term loss of containment heat removal;
   - Explosion of combustible gases, including hydrogen and carbon monoxide;
d) Severe accident with containment bypass;
e) Significant fuel degradation in a storage fuel pool and uncontrolled releases.

The demonstration of practical elimination of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release include deterministic considerations, and engineering aspects such as design, fabrication, testing and inspection of structures, systems and components and evaluation of operating experience, supplemented by probabilistic considerations, taking into account the uncertainties due to the limited knowledge of some physical phenomena.

Demonstration of 'practical elimination' of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release should include, where appropriate, the following steps:

a) Identification of conditions that potentially endanger the integrity of the containment or allow bypassing of the containment, resulting in an early radioactive release or a large radioactive release;

b) Implementation of design and operational provisions in order to practically eliminate the possibility of those conditions arising. The design of these provisions should include sufficient margins to cope with uncertainties;
c) Final confirmation of the adequacy of the provisions by deterministic safety analysis, complemented by probabilistic safety assessment and engineering judgement.

Although probabilistic targets can be set, demonstration of the practical elimination of conditions arising that could lead to an early radioactive release or a large radioactive release should not be based solely on low probability values. Such event sequences should be deterministically defined and their practical elimination should be demonstrated based on the performance of safety features making the event sequences extremely unlikely to arise.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Yes, because it is applicable to nuclear installations, what is a general term defined by the Atomic Act as follows: Nuclear installation shall mean a set of civil structures and the necessary technological equipment in a configuration specified by the design, intended for:

1. Generation of electric energy or for research in the field of nuclear energy, part of which is a nuclear reactor or nuclear reactors, which will use, are using or had been using controlled fission chain reaction;
2. Management of nuclear material – quantities greater than one effective kg except the areas for storage of containers and shields, in which the nuclear material is used as shielding material for radioactive sources, facilities for treatment of uranium ore and storage of uranium concentrate;
3. Spent nuclear fuel management;
4. Radioactive waste management; or
5. Uranium enrichment or production of nuclear fuel.

10 PERIODIC SAFETY REVIEW

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

The periodic safety review (PSR) is required by the Atomic Act No. 541/2004 Coll., as amended, particularly in §23 article 2. The scope of the PSR is defined in the UJD SR

Decree on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear installations No. 33/2012 Coll., as amended. Regulatory requirements are concretized in the UJD SR guide on Comprehensive periodic safety review, BNS I.7.4/2016. The PSR is stipulated to be conducted in 10-year interval.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account?

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

The safety objectives of PSR are stipulated in §2 Article 4), 5) and 6) of the UJD SR Decree on the regular, comprehensive and systematic evaluation of the nuclear safety of nuclear installations No. 33/2012 Coll., as amended. They include the following safety factors:

a) Plant design;

b) Actual condition of the plant;

c) Equipment qualification;

d) Ageing management;

e) Deterministic safety analysis;

f) Probabilistic safety assessment;

g) Hazard analysis;

h) Operating safety;

i) Use of experience from other plants and research findings;

j) Organization, administration, safety culture and management system;

k) Procedures;

l) Human factor;

m) Emergency planning;

n) Radiological impact on the environment;

o) Long-term operation.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

On the basis of the §2 Article 4 and 6 of the UJD SR Decree No. 33/2012 Coll. as amended, where is determined that:

4) Periodic safety review is focused on:
   a) A comparison of the nuclear safety status achieved on a nuclear installation with current nuclear safety requirements and good practice;
   b) An assessment of the cumulative effects of aging, the impact of changes made and considered on nuclear installations, operational experience and technological development on nuclear safety;
   c) Identification of justified safety improvements to the nuclear installation in order to maintain the required high level of nuclear safety or to increase it to a level of modern nuclear facilities in the world;

   …

6) License holder on the basis of PSR identifies and evaluates the safety significance of deviations from applicable current safety requirements and internationally recognised good practices taking into account operating experience, relevant research findings and the current state of technology.

11 SAFETY CULTURE AND OPERATIONAL EXPERIENCE

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Safety culture should be promoted across licensee's management system.

According to the §22 Article 2 u) of the Atomic Act No. 541/2004 Coll., as amended, the license holder shall inform its suppliers and subcontractors whose activities may affect nuclear safety with the requirements of the safety culture and control of their performance.

Requirements for safety culture are determined in §3, Article 7, 8 and §4 Article 11 of the UJD SR Decree No. 431/2011 Coll., as amended:

- An applicant for an authorization or license holder shall apply the requirements of the quality management system in a graded approach at all levels of the quality management system in accordance with the current state of the nuclear facility for the purpose of enhancing safety culture and allocating the necessary resources;
- An applicant for an authorization or a license holder shall implement measurable indicators or evaluable performance indicators of the process and safety culture assessment within the quality management system;
- The quality management system documentation must be regularly reviewed and updated by the license holder according to the actual status of the quality management system of the permit applicant or license holder and the status of the nuclear installation. Examination of the topicality of the quality management system documentation must be done at least every three years. If this review results in the need to update the dossier that the Authority does not approve or assess, the license holder shall revise it within three months of the review.

Requirements for the quality management system for safety culture are defined in UJD SR Decree No. 431/201 Coll., as amended, Annex 1. The quality management system of the applicant for licensee of the authorization shall include:

q) Human resources requirements, the procedure for recruiting, selecting and staffing positions with a direct impact on nuclear safety and having an impact on nuclear safety, qualification and retention of staff, with an emphasis on the ability to ensure a high level of safety culture;

u) Requirements relating to processes involving a safety culture;

am) Continually improving and improving the efficiency of its processes based on input from self-assessment, independent evaluation, management review, monitoring and measurement processes, with emphasis on nuclear safety, radiation protection and safety culture, including plans to provide adequate resources for these activities; and

p) Ensuring and maintaining an adequate level of safety culture.

The principles defined in INSAG-4 have been followed in implementation of Safety Culture concept in Slovak NPPs.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

According to the Atomic Act No. 541/2004 Coll., as amended:

§10

t) The license holder obligations are to ensure a systematic analysis of operational events and experiences, the development of international safety standards and new knowledge gained through research and development, and use them to improve the safety of their nuclear facility and their activities.

§23, Article 2 the license holder is obliged

q) To record, evaluate and document safety relevant in-service experience and experience of the operation of other comparable nuclear facilities for the purpose of identifying covert disruption to the achieved level of nuclear safety or potential precursors and possible trends to reduce nuclear safety or safety reserves,

r) Ensure the identification of the causes of operational events and the evaluation of operational experience, including appropriate qualifications of employees,

s) Develop a system for evaluating and storing feedback-related information from operational experience so that feedback staff can easily search and evaluate this information at any time.

Appendix 9: Answers to the questionnaire - Slovenia

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

QUESTIONNAIRE ON NATIONAL PRACTICAL APPROACHES

SLOVENIA

Prepared at

Slovenian Nuclear Safety Administration

18 December 2018

Table of contents

This questionnaire is addressed to the national nuclear safety authorities of the European Union Member States. Please fill in the following information prior to sending your answers:

Name: [redacted]

Organization: Slovenian Nuclear Safety Administration

Position: [redacted]

1 GOAL, SCOPE AND EXPECTED ANSWERS

The goal of the questionnaire (chapters 2-11 below) is to gather answers in order to highlight similarities and differences between the Member States approaches regarding the practical implementation of Articles 8a-8c of Directive 2014/87/Euratom. An analysis of answers will be performed to provide corresponding insights, with an overall objective of sharing the experience between Member States. A report will be established and the outcomes will be presented and discussed during a workshop held by the European Commission in 2019.

The questions aim at identifying practical approaches which contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom in practice within the Member States. Therefore, answers are expected regarding the technical aspects, approaches and methodologies applied in the Member States. The corresponding implementation by the utilities should be also emphasized. <u>The assessment of the legal implementation of Articles 8a-8c of Directive 2014/87/Euratom is not in the scope of this study</u>.

Some of the topics covered by the questionnaire have already been discussed in international fora. Therefore, information and answers pertaining to some questions may be available in publicly available documents within another context. If relevant, and if such information comprehensively addresses the question, to avoid duplicating work and to ensure consistency, it would be acceptable to make a reference to the relevant document, including where to find the answer in the document.

Since this questionnaire addresses practical and technical aspects, illustrating the answers with examples is encouraged.

The questionnaire concerns nuclear power plants (NPP) as well as research reactors (RR) exceeding 1 MW thermal power only. Eventually, even though the questions are relevant for both existing and new reactors, it may provide benefit to consider within the answers that:

- Article 8a mentions the objective to be considered for design, siting, construction and commissioning of nuclear installations, in addition to their operation and decommissioning. This objective fully applies to new installations and is to be used as a reference for existing ones. Thus, it is conceivable that expectations from Member States are generally higher for new reactors than for existing ones whatever the topic is. The questions have been formulated in order to capture the essence of these higher expectations, when relevant.

- Specificities of each RR cannot be embedded within general questions. These specificities are worthwhile to be mentioned in the answers.

The questionnaire is divided into 10 chapters, each covering a topic relevant to at least one of the three articles 8a, 8b and 8c from the Nuclear Safety Directive:

- Safety goals and objectives for new reactors and for existing ones: Article 8a
- Design and operational aspects of the defence-in-depth: Article 8b
- Probabilistic Safety Assessments: used in the demonstration that the objective from Article 8a is achieved
- Design extension conditions without fuel melt: Article 8b
- Design extension conditions with fuel melt: Article 8b
- Design of the containment: Article 8b
- Practical elimination: is one of the elements of the demonstration that the objective from Article 8a is achieved
- Periodic safety review: Article 8c
- Safety culture: Article 8b

2 General safety approach (safety "philosophy")

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

Slovenian legislation is in line with the IAEA safety standards, WENRA SRLs (2014) and WENRA Safety Objectives (which basically envelope the NSD Articles 8a – 8b). Continuous improvement is assured with the implementation of Periodic Safety Reviews (PSR).

3 Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors?

Existing reactor(s) have a few exemptions in the legislation compared to new ones, but it is still required that existing reactors implement improvements (through PSRs and possible other processes) and strive towards safety level of new reactors.

If so:

- please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;

  We understand as it is written above under 3.1.

- please provide concrete examples of more demanding objectives and/or demonstration.

  One example would be an aircraft crash design of the containment. This is required for new reactors, while for existing one, mitigation measures must be in place. This is an example when further improvements are not reasonably possible (you cannot build a new containment).

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

Design Extension Conditions must be considered for new and existing reactors.

It is required that early and large releases are practically eliminated. It is also required that only limited protective measures are needed for public (no permanent relocation, no need for evacuation in immediate vicinity of the plant, limited sheltering and long-term food restrictions) and that sufficient time must be available to implement these measures.

Currently these are very general requirements, but will be amended in future, most probably with a definition of large and early releases (in terms of TBq of iodine and/or caesium, and time limit of early releases), and probably with some probabilistic limit of such an event.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

No.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Our requirements regarding extreme natural hazards are in line with WENRA SRLs (Issue T).

Besides natural hazards the design bases shall also take into account human made external hazards including airplane crash and other nearby transportation, industrial activities and site area conditions that might lead to fires, explosions or other threats to the safety of the nuclear power plant.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

Currently the requirements regarding prevention of early and large releases are still very general. They will be amended in future, most probably with a definition of large and early releases (in terms of TBq of iodine and/or caesium, and time limit of early releases).

The avoidance of early and large releases shall be demonstrated by conservative deterministic analyses, supplemented by best estimate deterministic and probabilistic analyses for the DEC.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires “timely implementation of reasonably practicable safety improvements to existing nuclear installations”. Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

We only have a general requirement using terms “timely” and “reasonably” as well.

The Krško NPP is implementing a post-Fukushima Safety Upgrade Program (to be implemented by the end of 2021), which will increase the safety level of the plant closer to the new power plants’ safety level.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

Our approach is comprised of all the above: operating experience feedback is a basis for regular, prompt improvements. Benchmarks and peer review missions are also regular practices of getting inputs for possible improvements regarding plant operation and overall safety. PSRs as well are a regular practice, which takes review of all modifications done in last 10 years, as well as possible changes on site, it reviews developments in legislation, standards and best practices, and suggests the needed improvements, whether as physical changes to the plant or its operating procedures.

4 Design and operational aspects of the defence-in-depth

Question 4.1: Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Severe accident mitigation guidelines (SAMG) were developed for the Krško NPP following the strong encouragement from the SNSA. SAMGs were reviewed in 2001 by IAEA RAMP mission and in 2013 in the Krško NPP's Periodic Safety Review. SAMGs are constantly validated in emergency exercises through the use of the full scope plant simulator.

Regarding the DiD independence expectations, the legislation was updated in 2016 to take into consideration the WENRA Safety Objectives.

In the Krško NPP this approach is also followed, especially with the implementation of the Krško NPP's Safety Upgrade Program (the newly installed safety systems, instrumentation and power supplies all follow this approach).

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

Recent safety improvements in the Krško NPP are:

- Control of hydrogen inside containment: installation of passive autocatalytic recombiners – PARs (in 2013);
- Pressure control inside containment: installation of the passive containment filtered vent system – PCFVS (in 2013);
- Depressurization of RCS to ensure core cooling in the case pressurizer PORVs are unavailable due to severe accident: addition of two new MOVs in series in a new bypass line parallel to the current pressurizer PORVs line, for bleed-and-feed operation (in 2018);
- Operational control room available during DEC situations: installation of emergency control room in a bunkered building with independent power supply and instrumentation (in 2018).

5 Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

PSA level 1 to level 3 are required for new power plants. For the existing reactor (Krško NPP) and research reactors only level 1 and 2 are required. For the existing research reactor PSA is not required.

PSA forms a part of the Safety analysis report (SAR) and as such is required for the licensing from the beginning of construction (consent for construction) up to decommissioning of a nuclear facility.

- PSA level 1: for nuclear power plants and research reactors (except the existing research reactor)
- PSA level 2: for nuclear power plants and research reactors (except the existing research reactor)
- PSA level 3: for new nuclear power plants

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

There are no differences foreseen in the legislation regarding different licensing processes.

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

A PSA shall cover:

- all the relevant operational modes of the facility; in the case of a nuclear power plant, these operational modes include, in particular, modes ranging from refueling and operation at low power levels up to the full power operation;
- all the relevant initiating events and potential hazards, including fire, flooding, severe weather conditions and seismic events

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Yes, both, with the same level of details.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

Currently we do not have such a requirement, but it will probably be amended in future in such way.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

Legislation requires that PSA is used for:

- in working processes relevant to radiation or nuclear safety;
- to identify needs for modifications to the facility and written procedures for its operation, including the needs for severe accident management measures;
- in assessing risks involved in the facility operation (balance and compliance of the design bases with design principles, stability and predictability of the facility response to small deviations in the facility parameters, adequacy of facility modifications, justification of changes to operational limits and conditions, justification of changes of written procedures for the facility operation, safety significance of operational occurrences); ;
- In NPPs, the results of PSA shall be used also in:
    - the verification of the contents of the SSC maintenance, testing and inspection programmes; and
    - the development and validation of the professional training programme for the personnel.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

PSAs have to be updated regularly taking into account actual condition of the facility, changes to procedures, update of the component reliability and availability database, new standards, etc.

In practice, the main PSA model and analysis is updated once per fuel cycle (18 months), while supporting PSA analyses (external hazards for example) are updated when needed (based on new information, standards, etc.).

Question 5.7: Are PSAs required / developed for new and/or existing research reactors?

PSAs are required for new research reactors. For the existing one it is not required.

If so:

- What levels of PSA are performed? N/A
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)? N/A
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)? N/A
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)? N/A

6 Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

There are no differences between approaches for new or existing reactors.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

DEC requirements are required for NPPs. If a research reactor is of special type or higher thermal power, DEC requirements shall be applied in a graded approach. Currently in Slovenia there is one operating NPP (DEC required), and one research reactor, 250 kW power, for which DECs are not required.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

DEC without fuel melt (DEC A) are associated at level 3b of DiD in our regulation and practice. We make a following difference between DEC without fuel melt and "accidents within the design basis": DEC shall consider all events and combination of events that could lead to severe accident and of which probability cannot be proven as negligibly low with a high degree of confidence. They shall include events in all operational states, internal and external initiating events, as well as common cause failures. The DEC analysis shall rely on methods, assumptions or arguments which are justified, and should not be unduly conservative while conservativeness is required for safety analyses of accidents within the design basis.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

The applicant shall use a combination of deterministic and probabilistic assessments as well as engineering judgement to derive and justify as representative a list of design extension conditions. The approval of the list is done through a comparison with such lists in various guidelines and standards combined with engineering judgment (e.g. WENRA guidelines).

The applicant has to regularly and when relevant as a result of operating experience and significant new safety information, review design extension conditions using both a deterministic and a probabilistic approach as well as engineering judgement to determine whether the selection of design extension conditions is still appropriate.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Incidents taking place on the spent fuel pool are considered in the list of DECs without fuel melt. Example is loss of all design basis cooling systems.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?

  For the DEC A the analyses have to show that fundamental safety functions are fulfilled (control of reactivity, if it is lost, it shall be reestablished after a transient period; removal of heat from the reactor core and from the spent fuel; and confinement of radioactive material).

  The design extension conditions analysis shall rely on methods, assumptions or arguments which are justified, and should not be unduly conservative.

  The use of mobile equipment on-site can be taken into account, as well as support from off-site, with due consideration for the time required for it to be available.

- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?

  For DEC A (DECs without fuel melt) there shouldn't be any core melt (core damage limited as for design basis accidents).

- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

  Provisions of each level of DiD should be independent of other levels as reasonably achievable.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

Classification of SSCs is determined based on the analysis of their safety importance. In Krško NPP DEC systems are classified the same as the plant's safety systems.

Nuclear power plant shall have sufficient independent and diverse means (including necessary power supplies) to remove the residual heat from the core and the spent fuel. At least one of these means shall be effective after events involving external hazards more severe than design basis events.

DEC SSCs have to be adequately qualified to perform their relevant functions for the appropriate period of time under the required environmental conditions.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

There are no new reactors under construction in Slovenia.

7 Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.

There are no differences between approaches for new or existing reactors.

Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

DEC requirements are required for NPPs. If a research reactor is of special type or higher thermal power, DEC requirements shall be applied in a graded approach. Currently in Slovenia there is one operating NPP (DEC required), and one smaller research reactor, for which DECs are not required.

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

DEC with fuel melt (DEC B) is a part of "severe conditions".

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Answer is similar as for question 6.2 with an additional assumption of a core melt. For Krško NPP it is assumed that no operator actions are taken in the first 24 hours.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

The list of DECs with fuel melt does not include sequences that might happen on the spent fuel pool spent fuel storage, for which all possible measures have to be taken with the goal that a severe accident in such storage becomes extremely unlikely to occur. This is in line with WENRA SRLs.

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?

  The answer is the same as in the case of DEC without fuel melt (first bullet under question 6.4), except that a severe fuel damage must be assumed.

- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?

  In the case of DECs with fuel melt, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

  - allow sufficient time for protective actions (if any) in the vicinity of the plant;
  - avoid contamination of large areas in the long term.

- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

  Provisions of each level of DiD should be independent of other levels as reasonably achievable.

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

The answer is the same as for question 6.5.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

There are no new reactors under construction in Slovenia.

8 Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Isolation of the containment shall be possible in the design basis as well as design extension conditions. For those shutdown states where this cannot be achieved in due time, severe core damage shall be prevented. If an event leads to bypass of the containment, severe core damage shall be prevented.

Pressure and temperature in the containment shall be managed.

The threats due to combustible gases shall be managed.

The containment shall be protected from overpressure. If venting is to be used for managing the containment pressure, adequate filtration shall be provided.

High pressure core melt scenarios shall be prevented.

Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

In design extension conditions A, radioactive releases shall be minimised as far as reasonably practicable.

In design extension conditions B, any radioactive release into the environment shall be limited in time and magnitude as far as reasonably practicable to:

- allow sufficient time for protective actions (if any) in the vicinity of the plant; and
- avoid contamination of large areas in the long term.

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?

For the Krško NPP the containment is designed to withstand the temperatures and pressures associated with postulated loss-of-coolant accidents and main steam line breaks as well as collapse pressures induced by inadvertent operation of the interior spray system. In addition, the DEC B systems (and instrumentation) are designed for conditions (temperature, pressure, radiation) that are expected during severe accidents (core melt with reactor pressure vessel melt-through).

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

Leak tightness of containment penetrations is regularly tested.

9 Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Our legislation requires that:

"The design shall ensure that the core damage accidents with core melt, which would lead to early or large releases are practically eliminated. For core damage accidents, which cannot be practically eliminated practical solutions shall be available, which shall ensure that only limited protective measures are needed for public (no permanent relocation, no need for evacuation immediate vicinity of the plant, limited sheltering and long-term food restrictions) and that sufficient time is available to implement these measures."

Question 9.2: Do you have requirements related to practical elimination?

See answer under 9.1.

Question 9.3: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- For existing power reactors?
- For new power reactors?
- For existing research reactors?
- For new research reactors?

Slovenia has only one operating NPP, the Krško NPP. The concept of practical elimination is not considered explicitly in the improvements, but all of the new improvements (e.g. the Krško NPP's Safety Upgrade Program) strive towards the goal of practically eliminating severe accidents and early or large releases from the containment.

For research reactors practical elimination is not required.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

- Installation of passive autocatalytic recombiner (PARs) and containment filtered venting system (PCFVS);
- Additional flood protection of the nuclear island;
- Installation of pressurizer PORV bypass;
- Spent fuel pool (SFP) alternative cooling;
- Alternate cooling of RCS and containment;
- Installation of emergency control room (ECR);
- Installation of additional independent instrumentation (also qualified for severe accidents);
- ECR / Technical support center ventilation and habitability system;
- Upgrade of bunkered building 1 (BB1) electrical power supply;
- Installation of additional injection systems for reactor cooling system / containment and steam generators capable of assuring reactor cooling for at least 30 days (additional reservoirs of cooling water (also borated) capable of replenishing with water from underground wells) - the Bunkered Building 2 (BB2) project;
- Construction of dry spent fuel storage facility;

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

As written above, the requirement is that early and large releases are practically eliminated. But this is a very general requirement, thus this requirement will in future be amended, most probably with a definition of large and early releases (in terms of TBq of iodine and/or cesium, and time limit of early releases), and probably with some probabilistic limit of such an event.

The avoidance of early and large releases shall be demonstrated by conservative deterministic analyses, supplemented by best estimate deterministic and probabilistic analyses for the DEC.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?

Yes.

10 Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Yes. The regulatory framework is required by the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV, Off. Gaz. 76/17), and the scope of the PSR is defined in the Rules on operational safety of radiation and nuclear facilities (JV9, Off. Gaz. 81/16) and extensively described in the SNSA Practical Guidance PS 1.01: The content and scope of periodic safety review of radiation or nuclear facility Issue 1. These requirements and guidance were applied in the second PSR for the Krško NPP as well as for other nuclear facilities in Slovenia, namely a research reactor and a radioactive waste storage.

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

The WENRA SRL P2.2 were completely transferred into the Slovenian regulation JV9 and therefore, all of above safety factors are included in the PSR of a NPP. Three additional safety factors are included: Radioactive waste and spent nuclear fuel; Safeguards; Radiation protection. The safety objectives of a PSR are defined in the PSR program that is submitted by the operator and approved by the SNSA prior to the start of the safety review of all safety factors. When performing Periodic Safety Review the operator shall systematically verify overall impacts of ageing of the facility, effects of modifications of the facility, operational experiences, technical research and progress, changes at the site and other possible impacts on the radiation or nuclear safety of the facility. Operator shall use an up to date, systematic, and documented methodology, based on deterministic as well as probabilistic approaches to analyses and assessment of radiation and nuclear safety. The main safety objective is to confirm that the facility is at least as safe as intended during the design, and that can be operated safely until the next execution of periodic safety review (e.g. in 10 years). In some cases, additional objectives may be defined and included in the PSR, such as scoping and screening for aging management and for equipment qualification (in the first PSR of the Krško NPP).

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

First of all, these international standards are included as references into the PSR program and are then used as comparison for all of the safety factors. As mentioned above, the PSR program has to be approved by the SNSA. Additionally, if there are some other relevant safety standards found out later on during the review process, these standards are also used for the review and are included in the references of topical reports (for individual safety factors). The SNSA reviews draft reports for all of the safety factors and may insist on addition of such safety standards if this is relevant. The result of the review is that any noncompliance of the NPP with the safety standards is a PSR finding. The safety significance of these findings is determined later on, in the preparation of PSR summary report, when prioritization of PSR actions is determined. The prioritization is again reviewed by the SNSA and if needed, additional actions may be recommended to the operator. The PSR summary report, with list of all standards used for PSR, includes also the action plan. The PSR summary report has to be reviewed by independent authorized expert (a TSO) and then approved by the SNSA.

11 Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

In the article 93 of the “Ionising radiation protection and nuclear safety act” it is written that:

- The licensee of a radiation or nuclear facility must establish such an attitude/behavior in its organization, which results in a strong safety culture.
- The safety culture must be integrated into the management system.
- The licensee must check the adequacy and effectiveness of the safety and security culture through self-assessment and regular reviews of the management system.

More detailed requirements are written in the article 53 of “Rules on radiation and nuclear safety factors (JV5)”. The most important requirements are listed below.

The investor/operator of a radiation or nuclear facility shall with management system:

- provide management and personnel to promote activities to ensure the safety and contribute to the continuous improvement of safety culture;
- establish and support the desired and expected behavior and conduct, that promote a strong safety culture, whereby the desired and expected conduct and behavior also apply to subcontractors;
- ensure that individuals and groups safely and successfully carry out the tasks relating to the safety, taking into account the interactions between people, technology and organization;
- provide the means by which the organization is constantly developing, upgrading and improving its safety culture.

Individuals in the organization of the investor/operator of a radiation or nuclear facility shall support and promote a strong safety culture by strengthening:

- individual and collective commitment to safety;
- the acceptance of personal responsibility for safety;
- an organizational culture that supports and encourages trust, collaboration, consultation and communication;
- the reporting of any deficiencies in SSCs to avoid degradation of safety;
- the timely acknowledgement of problems, reporting back and actions taken;
- the ways in which the organization is constantly striving to develop and improve the safety and security culture;
- the allocation of responsibilities and powers of organizations and individuals for safety at all levels;
- reinforcing a learning and questioning attitude at all levels of the organization;
- prevention of discourage complacency;
- ensuring a common understanding of the key aspects of safety culture within the organization;

- awareness of the risks and dangers related to work and working environment, and to understand the potential consequences of these risks;
- conservative decision making in the implementation of all activities relating to safety.

The management of the operator of a radiation or nuclear facility shall:

- periodically carry out an independent evaluation and self-assessment of safety culture and its management;
- with the results of the evaluations acquaint employees and ensure continuous improvement and encourage open communication, collaboration, questioning, critical thinking and continuous learning of employees at all levels of the organization;
- shall ensure that its suppliers and subcontractors whose work can affect the safety of a radiation or nuclear facility, carry out their activities in accordance with the first and second paragraph of this article.

In 2012, the SNSA adopted the first guideline on assessing and supervision of SC in nuclear facilities. Further revisions were in 2014, 2017 and 2018. The guideline requires collecting observations_during the inspections, communications with the licensee, administration procedure process, reviewing of the licensee‘s self-assessment report, reviewing of the NPP‘s root cause analysis reports. All the gathered observations are analyzed and categorized into the safety culture's characteristics and attributes according to the IAEA SCART Guidelines. The results are compared with the results from the previous years. At the end, the report is written by the SNSA and send to the Krško NPP. Those results are, among other, discussed during the annual meeting among the SNSA and NPP management.

Furthermore, the SNSA performs the oversight of licensees' safety culture through:

- Permanent assessment of the licensees' safety culture;
- When performing periodic safety review - PSR (every ten years);
- During the inspections (safety culture) - inspections on SC are planned annually or every two years; So far three inspections were carried out related to the safety culture in Krško NPP (2014, 2016, 2017).

Three safety culture self-assessments were performed at Krško NEK so far. For the safety culture self-assessment three methods were used: observation, interview and questionnaire. At the end of the safety culture self-assessments the report is written. The results from the report are also discussed on the inspections dedicated to the safety culture at the Krško NPP.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

The SNSA follows foreign operating experiences on the regular basis. The information is gathered through the IAEA website, INES reports from the events, documentation gained at the international cooperation, … Some on the operational experiences are also dedicated to the safety culture and in others some key elements on safety culture could be identified. The employee at the SNSA monitors the foreign operational experiences and determines the colleague(s) to review the foreign operational experience and write a report. The important findings which could be also interesting for the Krško NPP are sent to the NPP. The results are discussed in the meeting and inspections during the SNSA and Krško staff.

In the article 111 of the »Act Amending the Ionising Radiation Protection and Nuclear Safety« it is required that the investor/operator of radiation or nuclear facility shall monitor its own and foreign operational experiences and use them in order to implement safety improvements.

The Krško NPP follows the external (foreign) and internal operational experiences as well as a corrective action program. They gather the internal information through audits, observations, internal reports (outage, annual, …), key performance indicators, … The external information is gathered through reports of the regulatory body (URSJV), Periodic safety review, INPO/WANO/NRC/IAEA documents and operational experience (SOER, …), …

Appendix 10: Answers to the questionnaire - Spain

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom. Questionnaire on Spanish practical approaches

The recent policy of Spanish government dictates the programed abandon of nuclear energy. Therefore, no new nuclear power plant is under consideration. Taking this into account, the attached analysis on the application of Directive 2014/87/Euratom is in exclusive to existing nuclear power plants.

2. GENERAL SAFETY APPROACH

Describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

Answer:

Safety and its continuous improvement is a pillar on the national framework and is explicitly mentioned as a mandate in high-level regulation. RD 1836/1999 approved Regulation on Nuclear and Radioactive Installations (RINR) art. 8.3 sets to the license holder the responsibility to look after the permanent increase of the safety. To do so, the licensees must analyze the best available techniques and practices and implement those in line with the requirements established by the regulatory body.

This mandate permeates into the safety culture via organization and management system on the license holder and the regulatory body in day-to-day bases and with high degree in the periodic safety review.

Maintenance of risk and potential damage as low as reasonably practicable is the actual driver of safety improvement.

RD 1400/2018 approved "Regulation on Nuclear Safety at Nuclear Installations." (RSN) develops in detail the safety approach implementation all along the life of the NPP: siting, design, construction, operation and consideration to dismantling.

An exhaustive analysis of application for continuous improvement on nuclear safety on each Spanish nuclear power plant, is contained in the 2019 Spanish report on the compliance to Vienna Convention on Nuclear Safety.

3. SAFETY OBJECTIVES FOR NEW REACTORS AND FOR EXISTING ONES

Q.3.1 Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:

- **please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;**

Please provide concrete examples of more demanding objectives and/or demonstration.

Answer:

As responded to the previous question, the new reactors are out of the scope of the future Spanish energy supply strategy. But, whatever the energy planning be, the RD 1400/2018 by which the Directive 2014/87/Euratom is transposed to the Spanish regulatory framework, clearly states, on the main facts of the transposition, that "the safety objective must be required to new facilities and considered as a referring for the existing ones". Moreover, the RD states that there are "…requirements addressed to the licensing holder, but also to the applicants of authorizations of any in-scope nuclear facility".

It is the aim of the Spanish regulatory body to require the existing NPPs to fulfill the safety objectives established by the Directive, as if they were new reactors.

Q.3.2 Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defense-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defense-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

Answer:

Normal operation and highest frequency events (1R/Y), must be accommodated by limitation and control systems and operators routine operations.

Anticipated abnormal occurrences and accidents within design bases, must be accommodated by protection system, engineered safety features and emergency procedures (not in the short term), including declaration of NPP internal emergency and general external emergency in due course.

Design extension conditions, class A (no severe core damage), must be accommodated by safety important systems (includes all SSC relevant to severe conditions), emergency procedures including declaration of siting internal emergency and general external emergency in due course.

Design extension conditions, class B (severe core damage). As for DEC-A plus successful activation of full external emergency with urgent (confinement, prophylaxis, evacuation) and long-term actions (temporal population displacement, food restrictions, etc.).

For all events under design bases, quantitative radiological acceptance criteria exist, linked to the frequency classification of the event.

For DEC-A and DEC-B, emergency program dictates radiological criteria for urgent and long term actuations. The area and time are embedded in the radiological criteria, local in nature.

Technical acceptance criteria for protection and engineered safety features related to design bases events are embedded under the concept of design bases event.

For of DEC-A and DEC-B structures, systems and components important to safety, reliability and capability constitutes a technical acceptance criteria.

Q.3.3 For new reactors, do the safety objectives associated with the different levels of defense-in-depth differ from those for existing reactors?

Answer:

No new nuclear reactors are planned in Spain, though, should it occur, the new facilities would be in the scope of the new RSN transposing EU Directive 2014/87, as it was indicated in Q.3.1.

Q.3.4 Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Answer:

Specific requirements regarding different external, natural and man-made but not malevolent, are included in the Spanish regulatory framework as following:

- RSN, in which **art. 14** requires scope and content of the "site evaluation", and **art. 15** requires to implement specific monitoring programs for site parameters.
- **IS-26, art 4, Site** (general criteria for site evaluation and external hazards).
- **IS-27** (Design criteria), **criterion 2** ("Natural hazards and combinations").

- **IS-37** (Design basis accidents analysis), **art. 8 & 13** (design basis events and events beyond design basis or extended design, even severe external hazards).

Q.3.5 Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms *early* and *large* release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

Answer:

In fact, the term "early" is directly linked to the feasibility for the implementation of urgent measures (RD 1546/2004 PLABEN, Annex II):

Confinement → Avoided effective dose in as much as two days > 10mSv
Prophylaxis → Avoided equivalent dose thyroid > 100 mGy
Evacuation → Avoided effective dose in as much as 7 days > 50 mSv

Area and permanent evacuation are triggered when the projected dose for the whole life is above 1 Sv or the cumulated dose in a month, one or two years after temporal evacuation, is above 10mSv.

Compliance to early and large release are verified through accident scenarios, and time implementation of protective actions are verified through emergency exercises.

Additionally, Spanish NPP were licensed before 14th August, 2014. Consequently, in the context of the European Stress Tests process after the Fukushima-1 NPP accident, a set of improvements to cope with Design Extension Conditions (type A and B) were identified as reasonably practicable safety improvements and, therefore, have been implemented in every Spanish NPP. These improvements are directly linked to the dynamics of the accidents ("early") and to the magnitude of the releases ("large").
These improvements deal with:

- The prevention of the accident. For example: enhancement on systems and portable equipment available on-site, and improvements related to the accident management, and
- The delay or mitigation of the magnitude of the releases. For example: Passive Autocatalytic Recombiners (PAR) for the management of combustible gases with the objective to prevent deflagration and detonation in the containment, protecting the confinement function; or Filtered Containment Venting System (FCVS), which provide protection to the containment while mitigating the release due to venting.

Q.3.6 Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is *timely* implemented and *reasonably practicable*. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

Answer:

Safety significance of the improvement is the main driver for the concepts of "timeliness" and "reasonably practicable". The marginal benefit of the technical or operational (good practice) improvement in terms of safety is relevant to evaluate both terms. A significant improvement on an issue with high relevance in terms of safety will be applied with no delay, whereas an identified practicable improvement with marginal safety benefit will be implemented as soon as reasonably practicable under the concept of continuous improvement or as part of a periodic plant update at the latest i.e: Periodic Safety Review.

Technical as well as non-technical information supports decision making in this matter. As an example and regarding the case of periodic safety review, the need for design safety improvement is evaluated as a result of the assessment of the licensing bases, requirements and national and international current practices. This assessment is extended to many different aspects: Safety important systems, structures and components; equipment qualification; ageing; deterministic and probabilistic safety analyses; internal and external risk analyses; organization, safety culture and management system; procedures; emergency planning; environmental impact; radiological protection.

Safety improvement to anyone of the issues mentioned above, makes use to different degree of technical information like: dose indicators, source terms, discharges, incidents and operational data, reliability data, systems capability,… In a similar way, use is made of non-technical information like: adequate maintenance and compliance to official documentation, adequacy of the multiple safety related programs in the plant to fulfill their objectives…

Additionally, Spanish NPP were licensed before 14th August, 2014. Consequently, in the context of the European Stress Tests process after the Fukushima-1 NPP accident, a set of improvements to cope with Design Extension Conditions (type A and B) were identified as reasonably practicable safety improvements and, therefore, have been implemented in every Spanish NPP. These improvements are directly linked to the dynamics of the accidents ("early") and to the magnitude of the releases ("large").
These improvements deal with:

- The prevention of the accident. For example: enhancement on systems and portable equipment available on-site, and improvements with regard to the accident management, and

- The delay or mitigation of the magnitude of the releases. For example: Passive Autocatalytic Recombiners (PAR) for the management of combustible gases with the objective to prevent deflagration and detonation in the containment, protecting the confinement function; or Filtered Containment Venting System (FCVS), which provide protection to the containment while mitigating the release due to venting.

All these improvements to design are already implemented in the Spanish NPP.

Q.3.7 What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

Answer:

NPP nuclear safety is driven by different means. On a permanent bases, the own operational experience of the NPP, as well as from nuclear industry and others NPPs, and the assessment of new technical regulation at least annually.

Independent assessment from international missions like IAEA review missions and WANO is also an important element to push for safety improvement. On a ten year basis the periodic safety review, as a full scope review, aims to plant safety improvement for the incoming period of time.

From the regulatory point of view, the analyses of regular supervision results, and inspection findings, as well as regular plant assessment are the main approach for continuous improvement of plant safety.

4. DESIGN AND OPERATIONAL ASPECTS OF DEFENCE-IN-DEPTH

Q.4.1 Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests

Answer:

Related to the design of the plant, the CSN Instruction IS-27 *General Design Criteria for NPP* was reviewed and issued the revision 1 in June 2017, which was prepared taking into account the upcoming of WENRA, as well as the Directive 2014/87/EU.

Specifically, this IS-27 includes the Criterion 2, addressing the protection against **natural phenomena**, which contemplates the selection of historical site severe natural phenomena, with margin enough to accommodate uncertainties and the potential for credible combinations of these phenomena with accident conditions. IS-27 also includes Criterion 3, devoted to **fire protection**, including the avoidance of fire spreading and confinement barriers. Additionally, Criterion 4 addresses **environmental and dynamic effects** on SSC during the whole life of the plant. Accordingly to the SSC safety classification, qualification and protection criteria are stated, taking into account flooding, whip effect and external events. The IS-27 provides for the possibility to preclude dynamic effects from postulated pipe breaks for which it has been demonstrated, and accepted by CSN, a low likelihood.

DiD is specially enhanced and reinforced by the new RSN, transposing EU Directive 2014/87. Specifically, the RSN distinguishes among DBA, BDBA, external and internal events, well indicated in the article 3. *Definitions* in order to provide for a clear and overall view of the different plant safety analysis.

At the level 3 of DiD the regulatory framework is well developed by IS-37, yet mentioned in the previous paragraph. At the level 4 of DiD, the requirements imposed by the RSN are fulfilled by both the safety evaluations that have been carried out since the initial and the subsequent licensing processes, and the implementation of the Fukushima National Action Plan (NAcP), upcoming from the incorporation to the plants licensing basis the mandatory CSN Technical Instructions to undertake the stress tests and to accomplish the requirements for both DEC and severe accidents. The Spanish NAcP has been followed up, with the only on-going action, related to the seismic characterization of sites, having recently finished the phase I, related to walkdown and data collection.

For the purposes of illustration concerning the level 3 of DiD, carried out before European stress tests, it can be cited, among the most relevant improvements:

- Reinforcement of the safe shutdown analysis, both in case of fire and other scenarios requiring to bring the plant to a safe condition. This resulted in many design modifications affecting the remote shutdown panels, electrical independence, procedures, approval of operator manual actions, etc.
- The PSAs in other operational modes (APSOM), which permitted to identify operational situations during outage in which the safety should be better managed by reviewing procedures or elaborating new strategies. Among them, it can be cited the optimization of the RHR operation in conditions of low RCS level.
- Improvement in the surveillance of critical safety functions during outages, considering the NUMARC 91-06. Development of strategies to cope with loss of cooling systems, water inventory and containment integrity.

Related the NAcP, the updated revision 2 of this document was submitted to ENSREG by the end of December 2017, according to the term agreed with this organization. No relevant changes have occurred since that date, unless the progress in the action related to the seismic characterization of sites, which phase I (data collection and elaboration of databases) have finished and phase II is ongoing.

Q.4.2 What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

Answer:

As mentioned in the previous question, Spain carried out the whole set of actions that the international community undertook under the scope of the post-Fukushima accident context. The associated NAcP was issued and reviewed, according to the statements set forth by the main regulatory international fora. Among these actions, many rules in the Spanish regulatory framework were reviewed, covering topics as, among others, the nuclear safety and engineering aspects (site selection and evaluation, assessment of extreme natural hazards, including their combined effects, management of severe accidents, station blackout, loss of heat sink, accumulation of explosive gases, the behaviour of nuclear fuel and the safety of spent fuel storage), focusing specially to the following main areas:

- 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 issuance of the new RSN is one of the most relevant enhancements of the Spanish regulatory framework, as it requires many actions focused on the continuous safety improvement. Among the most noticeable facts related to the implementation of those requirements, it can be cited:

- Fukushima NAcP, with the whole relevant measurements (analysis, design modifications, procedures, etc) 100% implemented, except the ongoing seismic characterization of sites.
- Reinforcement of safety culture by incorporating this acknowledgement under the scope of training processes and procedures.
- Elaboration and assessment of the first Topical Peer Review on ageing management and elaboration of action plans, according to the upcoming findings and results.
- Issuance of the CSN Safety Guide GS-1.10 Rev. 2 on Periodic Safety Review, according to IAEA SSG-25, EU Directive 2014/87 and the international operating experience. This guidance incorporates the methodology based on safety factors, considering aging effects and long term operation, as applicable to the renewal of licensing authorizations. The associated licensing process incorporates two milestones requiring CSN approval, both the PSR Basic Document and the PSR itself.

- Reinforcement of tools focused on the stakeholders participation and communication to the public to better implement the requirements on that.

5. PROBABILISTIC SAFETY ASSESSMENTS

Q.5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- **PSA level 1:**
- **PSA level 2:**
- **PSA level 3:**

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

Answer:

Spanish Regulation (Nuclear Safety Rule) requires a safety assessment for the design stage and the operating stage that must be complemented with a systematic risk assessment.

In the case of operating NPP, mandatory Instruction IS-25 "Criteria and requirements on the performance of probabilistic safety assessments and their applications for nuclear power plants" must be considered. It is specified that such assessment must be a Probabilistic Safety Analysis (PSA). The scope for the PSA must be level 1 and level 2 including internal events, fire an internal floods for all operating modes and covering both reactor and spent fuel storage.

Core damage frequency (CDF), large early release frequency (LERF) and large release frequency (LRF) must be estimated. PSA level 3 is not required.

NPP must to set out an operational experience data base to collect information about the main components in PSA model in order to be ready to get failure rates, unavailability´s data and frequencies for specific initiating events.

For the main results (CDF and LERF) sensibility and uncertainties assessment are required.

Q.5.2 Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

Answer:

Instruction IS-25 requires a Probabilistic Safety Analysis (PSA). The scope for the PSA must be level 1 and level 2 including internal events, fire an internal floods for all operating modes and covering both reactor and spent fuel storage.

Regarding external hazards (earthquake, external floods, etc.) have to be assessed but PSA is not required.

Q.5.3 Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

Answer:

Yes, as it said in Q5.2 spent fuel storage analysis is covered in the same detail level.

Q.5.4 More generally, do you expect PSAs covering multiple installations at a same site?

Answer:

PSA has to cover all reactors at the same site, also they must cover all the spent fuel pools at the site, but it is not expected a PSA for dry spent fuel storage at the same site.

Q.5.5 Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

The definition for Technical Specifications is deterministic and is based on design base analysis, however IS-25 as well as IS-32 (Technical Specifications for NPP) require to include Technical Specifications for systems and components with high risk impact.

On a voluntary basis, some licensees have applied for several regulatory PSA applications such as Risk informed-Inservice testing, risk inform-Inservice Inspections service and Transition to NFPA 805.

In addition, some design modifications in NPPs have been implemented in order to reduce risk in terms of CDF or LERF.

Q.5.6 When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

Answer:

Instruction IS-25 requires to update PSAs in order they represent the current state and the real operation of the NPP as much as possible. With that goal, after refueling outages, licensees must assessment the design modification implemented and changes in procedures in order to update PSA´s models in case that those plant modifications have risk impact. In any case, PSA level 1 must be updated to include new failures and unavailability of components.

Finally, PSA level 1 and level 2 for internal events for full operation, level 1 for shutdown and level 1 for fire and internal floods must be updated every five years. Others models than these must be updated every 10 years.

Q.5.7 Are PSAs required / developed for new and/or existing research reactors?

Answer:

There is not research reactors in Spain. No new nuclear reactors are planned in Spain, though, should it occur, the new facilities would be in the scope of the new RSN transposing EU Directive 2014/87, as it was indicated in Q.3.1. This RSN incorporates requirements for using PSAs as part of a systematic safety evaluation.

6. DESIGN EXTENSION CONDITIONS WITHOUT FUEL MELT

Q.6.1 At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice?

Answer:

According to IS-37, Design Extension Conditions have been defined as a "*set of measures that are part of the defense in depth of the installation and that aim to improve the safety of the central by reinforcing the capabilities of the plant to withstand situations more demanding than those considered in the design bases, as well as the reduction of radioactive emissions to the environment*".

DEC are categorized in the regulation (IS-36 and IS-37) as DEC-A and DEC-B, being DEC-A a Design Extension Condition without fuel melt.

Taking into account the previous information, DEC without fuel melt would be part of Level 4 of DiD "Control of BDB conditions, including prevention of accident progression and mitigation of severe accident consequences" by the use of complementary measures and accident management to ultimately maintain confinement function, as:

- Fuel melt is not foreseen within DEC-A, so no radiological consequences of significant releases of radioactive materials are assumed, and
- DEC implies more demanding situations than those considered in the design bases.

Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.

Answer:

In RSN, important to safety Structures, Systems and Components (SSC) are divided in: safety SSC (or safety-related) and relevant to safety SSC.
Safety SSC are those that have to fulfil their safety function in design postulated events. IS-27 sets out the general criteria applicable to safety-related SSC at NPP.
Relevant to safety SSC are, among others, those that have to fulfil their safety function in Beyond Design Basis Events (or Design Extension Conditions), a group of SSC. For these SSC and for DEC, besides the RSN requirements, the requirements included in IS-36, IS-37 and Evaluation criteria developed by CSN for Post-Fukushima modifications are applicable.

In accordance with the requirements of IS-37, plant's safety analysis report shall include a set of studies associated with the initiating events postulated within the design bases, as well as a study of the design extension.

As mentioned before, the Design Extension Conditions are classified in DEC-A and DEC-B depending on whether or not fuel melt has been reached, and both imply more demanding situations than those considered in the design bases so, a difference between DEC without fuel melt and "accidents within the design basis" has been established both in CSN regulations and plant's safety analysis.

Q.6.2 What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Answer:

The applicant has come up the list of DEC justified on the basis of deterministic and probabilistic arguments and of engineering judgement. The selection process shall take into account all those events or combinations of events that cannot be considered extremely unlikely with a high degree of confidence and that might give rise to accident conditions more severe than those considered in design basis accidents. This methodology is described in IS-37.

The licensees have included a new chapter in their Safety Reports for DEC analysis.

The CSN has revised the list proposed by the applicants mostly using engineering judgement.

According to GS-1.10, the safety deterministic analysis has to be revised. This revision, according the IS-37, must include DECs without fuel melt

Q.6.3 Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer:

Sequences that might happen on the spent fuel pool where analyzed during the European Stress Tests after the Fukushima Dai-ichi accident. These sequences included the loss of cooling and the loss of coolant in the spent fuel pool in the event of total and long-term loss of power and ultimate heat sink. They are referenced in the DEC analysis chapter that have been included in the Safety Reports of the NPP.

Q.6.4 What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- **What are the rules to analyse DEC without fuel melt?**

Answer:

In general, the requirements are those included in the CSN Instruction IS-37.

The methods and hypotheses used shall reflect the foreseeable conditions and evolution in an overall conservative manner, but without this conservatism detracting from the expected evolution of the facility.

The methodology and assessment shall take into consideration both the uncertainties and their impact, with special attention to those cases in which use is made of expert judgement.

- **Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?**

Answer:

Design extension analysis shall be used to define the design basis of the systems required to prevent the appearance of the conditions postulated in this analysis or, if they occur, to be able to control them and mitigate their consequences.

- **Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?**

Answer:

No

Q.6.5 What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

Answer:

The analysis has taken credit only for systems, structures and components used for mitigating DBAs.

Q.6.6 If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer:

No new nuclear reactors are planned in Spain, though, should it occur, the new facilities would be in the scope of the new RSN transposing EU Directive 2014/87, as it was indicated in Q.3.1.

7. DESIGN EXTENSION CONDITIONS WITH FUEL MELT

Q.7.1 Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

Answer:

Article 8b has been included in the recently issued Spanish RSN.

In accordance with the requirements of IS-37, plant's safety analysis report shall include a set of studies associated with the initiating events postulated within the design bases, as well as a study of the design extension.

As mentioned before, the Design Extension Conditions are classified in DEC-A and DEC-B depending on whether or not fuel melt has been reached, and both imply more demanding situations than those considered in the design bases so, a difference between DEC without fuel melt and "accidents within the design basis" has been established both in CSN regulations and plant's safety analysis.

Q.7.2 What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Answer:

DEC with fuel melt conditions were not considered in the original design during the commissioning of the Spanish NPP. However, in Spain these situations have been included in the so-called Design Extension Conditions.
DEC with fuel melt (or DEC-B) were analyzed during the European Stress Tests after the Fukushima accident. The rules of this analysis were stablished in the specification developed by ENSREG (document published by ENSREG in May 2011). This analysis was done by NPP operators and thoroughly evaluated by CSN. As a consequence of this process, CSN issued Technical Instructions (legally binding) associated to the Authorization of the facility requiring Spanish NPP to implement a number of enhancements and improvements related with severe accidents (along with other requirements related to external hazards and to the defense against loss of power and ultimate heat sink).
The licensees have included a new chapter in their Safety Reports for DEC analysis.

The scope of DEC is revised during PSR. As stated in GS-1.10, the PSR applies to important to safety SSC (DEC SSCs are a subset of the important to safety SSC); additionally, as stated in GS-1.10, the Severe Accident Management Guidelines are reviewed in PSR.

Q.7.3 Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.

Answer:

Sequences that might happen on the spent fuel pool where analyzed during the European Stress Tests after the Fukushima accident. These sequences include the loss of cooling and the loss of coolant in the spent fuel pool in the event of total and long-term loss of power and ultimate heat sink. They are referenced in the DEC analysis chapter that have been included in the Safety Reports of the NPP.
The strategies included in these studies make use of portable means for injecting and spraying water to the spent fuel pool. The objective is to cool the fuel in the pool and, for the case of spray, to minimize the release of fission products in case the uncovering of spent fuel happens.

Q.7.4 What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- **What are the rules to analyse DEC with fuel melt?**
- **Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?**

Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

Answer:

Spanish regulation related to DEC with fuel melt, or DEC-B: RSN and IS-36.

DEC with fuel melt were analyzed during the European Stress Tests after the Fukushima accident. The rules of this analysis were stablished in the specification developed by ENSREG (document published by ENSREG in May 2011). This analysis was done by NPP operators and thoroughly evaluated by CSN. As a consequence of this process, CSN issued Technical Instructions (legally binding) associated to the Authorization of the facility requiring Spanish NPP to implement a number of enhancements and improvements related with severe accidents (along with other requirements related to external hazards and to the defense against loss of power and ultimate heat sink).
In the Spanish regulation there is no explicit quantitative limitation in terms of radiological consequences associated to the analyses of DEC with fuel melt included.

For severe accidents (DEC with fuel melt), specific design modifications have been implemented in all the Spanish NPP. The main modifications implemented in each site were: Passive Autocatalytic Recombiners, Filtered Containment Venting Systems, Alternative Emergency Management Center, and one centralized External Support Center. For the structures, systems and components implemented with these design modifications, evaluation criteria were developed by CSN ("Evaluation Criteria to be considered in Post-Fukushima Design Modifications", December 18th, 2013; see answer to question Q.7.5).

Q.7.5 What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Answer:

CSN issued the following document: "Evaluation Criteria to be considered in Post-Fukushima Design Modifications", December 18th, 2013.
The following is taken out from this document.

✓ *In general, SSC, either mobile of fixed (e.g. PAR or FCVS), including the support systems allowing their operability to be maintained, should be capable of undertaking their specified functions under those conditions for which their actuation is foreseen. This means that they must have a high degree of availability and reliability, which is understood as having a sufficiently robust design.*

Thus, they should be qualified for severe accident conditions and able to withstand extreme external conditions. Redundancy was not required.

Regarding the electrical supply (and other possible support systems or capabilities): should ensure correct performance of their function autonomously, without resources other than those available at the facility itself, following events they are required to address, for at least 24 hours without off-site support and for at least the following 72 hours with the support only of light equipment brought in from off the site.

In relation with the safety qualification: as these SSC are for beyond design basis situations, it was not considered necessary to be tested against standard safety qualification procedures, but there should be a reasoned demonstration (analytical or based on experience of verifiable use or, when no other method is possible, expert judgment) showing with a high degree of confidence that the SSCs will be capable of withstanding such condition while maintaining subsequent operability.

Q.7.6 If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

Answer:

No new nuclear reactors are planned in Spain, though, should it occur, the new facilities would be in the scope of the new RSN transposing EU Directive 2014/87, as it was indicated in Q.3.1.

8. DESIGN OF THE CONTAINMENT

Q.8.1 What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

Answer:

The applicable regulation, (IS-27) defines the main characteristics of the barriers to avoid fission products releases under design bases accidents.

For beyond design bases, the analyses performed during the European Stress Tests after the Fukushima Dai-ichi accident covered the topic of maintaining the containment and penetrations integrity on these situations.

Q.8.2 How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario...)?

Answer:

The containment design pressures and temperatures were initially defined for the vendors for each nuclear unit to be above of the design bases accidents worst case results. These values have been analytically verified in all the Spanish NPP as part of different licensing processes (power uprates, Plant Safety Analysis revisions, etc.).

Q 8.3 What is the approach for containment penetrations? Is their leak tightness regularly tested?

Answer:

Currently, in Spain there are four nuclear plants of US design (Westinghouse and GE) and one of German design (PWR, KWU). The former are tested according to the 10CFR 50 Appendix J, including Integral Leak Rate Tests (ILRT, each 10 years) and Local Leak Rate Tests (LLRT, with a performance based frequency); for the German plant the applicable standard is KTA-3405 which requires the ILRT to be performed each 5 years. In both cases the ILRT tests all the potential leaks (through penetrations and others) affecting the containment specified integrity which is considered in the radiological safety analysis.

9. PRACTICAL ELIMINATION

Q.9.1 Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.

Royal Decree RD/1400/2018 approved RSN art 6. “Nuclear safety objective for nuclear facilities” requires to avoid early or large releases, because physically impossible or being extremely unlikely with high level of confidence. That is, there are two considerations. First is the physical elimination. Second is the very low probability of the event/sequence with large confidence. These conditions can also be expressed in terms of annual frequency for the case of very low with high confidence.

Q.9.2 Do you have requirements related to practical elimination?

Answer:

There are not specific requirement related to practical elimination. However the requirements to implement measures in order to reduce risk or related with DEC analysis aim to the practical elimination of sequences beyond design basis with a potential for large damage.

According to WENRA issue F “Design extension of existing reactors”, in the case of DEC A scenarios reasonably and practical provisions have been taken to prevent these types of scenarios. For scenarios with core damage (DEC B) and potential large and early energy releases, provisions were taken under the scope of the EU Stress Tests analysis.

Practical elimination of accidental sequences is treated in detail under IS-37 art.12 for DEC-A sequences. For the case of DEC-B, reasonably practicable safety improvements results from EU Stress Tests were systematically implemented.

Q.9.3 Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

- **For existing power reactors?**
- **For new power reactors?**
- **For existing research reactors?**
- **For new research reactors?**

Answer:

The answer is only focused on existing reactors as no new reactor is considered and no research reactor neither exists nor is planned.

Technical regulation in force allows for the use of PSA Level 1, 2, shut down, external (e.g.: seismic) and internal events (e.g.: flooding, fire) together with defense in depth and safety margins. Increase on core damage or large early release frequencies and the presence of truncation frequencies for specific events are important elements to consider in the assessment.

Q.9.4 Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

Answer:

Following Fukushima event a number of plant changes were implemented, among them:

- Installation of passive autocatalytic recombiners
- Alternative emergency management center
- Containment filtered venting system
- Direct injection to cavity
- Portable equipment (injection capability to RCS, spent fuel pool, Steam Generators (PWR), containment; diesel generators, etc.)
- Containment atmosphere sampling system.

Q.9.5 What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

Answer:

Practical elimination of accidental sequences is treated in detail under IS-37 art.12 for DEC-A sequences

"The accident analysis for the facility shall be complemented with a study of the extension of its design.
A. This article addresses the treatment of scenarios belonging to the DEC-A category design extension. DEC-B category scenarios are not part of the subject matter of this Instruction.
B. The facility design extension study shall have the following objectives:

1. Study of the performance of the plant, including interactions between groups in the case of sites with more than one group, or close to other sites, in response to specific accident scenarios whose hypotheses exceed those considered in the design basis of safety-related structures, systems and components.
2. Determination of the possibility of improving the design of existing structures, systems and components, incorporating new structures, systems and components or implementing procedures or other measures, such that these actions contribute to reasonably minimising the risk to the population and the environment of harmful exposures to ionising radiations, and assurance of the existence of a margin with respect to limit situations in which minor variations of parameters give rise to disproportionate changes to the consequences.
3. Identification of reasonable measures allowing severe damage to the fuel to be prevented, such that this be extremely unlikely with a high degree of confidence.

C. Selection of DEC-A category events.

1. The selection of events to be analysed shall be justified on the basis of deterministic and probabilistic arguments and of engineering judgement.

2. The selection process shall take into account all those events or combinations of events that cannot be considered extremely unlikely with a high degree of confidence and that might give rise to accident conditions more severe than those considered in design basis accidents.

This selection process shall contemplate the following:

- *Any plant operating condition,*
- *Its origin in on or off-site risks,*
- *Common cause failure modes,*
- *The presence of more than one group on the site,*
- *Events having an impact on the site groups and on interactions between them or on other nearby groups.*

Appendix III includes an illustrative list of events to be considered in the analysis of design extension, excluding those already incorporated in the design basis.

D. Consideration shall be given to the set of credible accident sequences beyond the design basis of the facility and with respect to which it is reasonably feasible to implement prevention or mitigation measures. For the selection of these scenarios, use shall be made of a combination of deterministic methods, probabilistic analysis and engineering judgement.

E. Safety assessment methodology and contents.

1. The methods and hypotheses used shall reflect the foreseeable conditions and evolution in an overall conservative manner, but without this conservatism detracting from the expected evolution of the facility.
2. The methodology and assessment shall take into consideration both the uncertainties and their impact, with special attention to those cases in which use is made of expert judgement.
3. Resources and possibilities for the prevention of damage to the fuel by improving the capacity of the facility to withstand more serious scenarios than those contemplated in the design basis shall be identified.
4. The possible radiological consequences shall be assessed when, given their magnitude, they may exceed the maximum acceptable included in the design basis of the facility.
5. Consideration shall be given in the study to the configuration, location and capacities of the equipment, the conditions expected in each scenario and the feasibility of the actions foreseen for management of the accident.
6. Consideration shall be given in the assessment to the information on the availability of systems obtained from probabilistic safety assessments, allowing the probability of the sequences for analysis to be estimated.
7. A final safe status and mission times shall be defined for the structures, systems and components of which demands are made.
8. The analyses of these accidents shall make use of tools qualified for this purpose.

F. Design extension analysis shall be used to define the design basis of the systems required to prevent the appearance of the conditions postulated in this analysis or, if they occur, to be able to control them and mitigate their consequences.

G. The result of this analysis shall be incorporated in the Safety Assessment as an appendix associated with accident analysis."

For the case of DEC-B, reasonably practicable safety improvements results from EU Stress Tests were systematically implemented. IS-36 on emergency operating procedures and the management of severe accidents at nuclear power plants specifically considers (art.6) the verification and validation process of emergency operating procedures and severe accident management guidelines. This V&V process will pursue that (Art.5):
...

5.2 The licensee of the nuclear power plant shall have available suitable means to protect the containment against the consequences of a selected set of beyond design basis accidents, such that:

• The capacity to isolate the containment is available. If containment isolation cannot be guaranteed, there shall be means available to allow for mitigation of the consequences of loss of this safety function.

• The capacity to maintain containment leaktightness for a reasonable time following occurrence of the accident and as a result of it is available.

• The capacity to control containment pressure and temperature is available.

• The capacity to control combustible gases is available.

• The containment is protected against overpressure conditions.

• The probability of high-pressure molten core ejection scenarios is prevented or minimised.

• Degradation of the containment as a result of molten core attack is prevented or mitigated to the extent possible.

When these means are based on equipment, systems and components already contemplated in the design, they shall be evaluated and, where necessary, modified for performance of the new function. If new equipment, systems or components are included for the performance of these functions, they may be designed using realistic criteria.

The selection of beyond design basis accidents shall be carried out considering a combination of deterministic and probabilistic analyses and engineering judgement. Consideration shall be given in these analyses to internal and external events.
...

Q.9.6 Is the practical elimination concept applicable also for fuel storage pools?

Answer:

According to Spanish regulation, the practical elimination concept is applicable to any nuclear facility under the scope of RSN approved by RD 1400/2018. This implies that spent nuclear fuel pools are also subject to the "practical elimination" requirement.

10. PERIODIC SAFETY REVIEW

Q.10.1 Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?

Answer:

High level Regulation on Nuclear Safety (RSN) art 13. "Periodic Safety Revision" sets the mandatory character of the PSR and scope

Article 13. Periodic safety review.

1. The licensee must systematically and periodically re-evaluate the nuclear safety of the facility at least once every ten years. The purpose of this periodic safety review is to confirm the nuclear safety of the facility and obtain an overall assessment of its performance during the period in question, via systematic analysis of all aspects of nuclear safety and radiation protection.
The periodic safety review must:
a) confirm that the facility continues to comply with its design bases, or establish the necessary corrective measures if in any way they are not met.
b) confirm the availability and validity of measures for the prevention of accidents and the mitigation of their consequences, and the application of the principle of defence in depth.
c) ensure that nuclear safety remains at a high level during the following period.

2. Based on the periodic safety review, the licensee must introduce nuclear safety improvements within the facility that are reasonably feasible in terms appropriate to their safety importance, taking as a reference the safety objective established for facilities in article 6 of these Regulations.
To do so, the best practices and the evolution of international standards for nuclear safety and radiation protection must be taken into consideration. It must also take into account those aspects related to ageing, operational experience and the results of the most recent research and advances in science and technology that are compatible with the existing design.

Q.10.2 What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- **Plant design**
- **Actual condition of structures, systems and components (SSCs) important to safety**
- **Equipment qualification**
- **Ageing**
- **Deterministic safety analysis**
- **Probabilistic safety assessment**
- **Hazard analysis**
- **Safety performance**
- **Use of experience from other plants and research findings**
- **Organization, management system and safety culture**
- **Procedures**
- **Human factors**
- **Emergency planning**
- **Radiological impact on the environment**

Spanish, Nuclear Safety Commission issued, nuclear Safety Guide 1.10 (SG-01.10) on “Periodic Safety Review for Nuclear Power Plants”. It develops in detail the objectives as well as safety reference levels and methodology to comply with the high level regulation mentioned in Q.10.1

Objectives:

1. To substantiate the adequacy and effectiveness of the arrangements and the Structures, Systems and Components (SSC) that are in place to ensure plant safety until the next RPS or where appropriate final commercial operation.
2. To verify the extent to which the plant conforms to current national and international regulation and good practices.
3. To identify the need of actuations to solve any deviation from the licensing bases
4. To elaborate an action plan to increase or maintain a high safety level until the next RPS or final commercial operation.
5. To identify the improvements on operating official documentation, including the licensing basis, until the next RPS or where appropriate final commercial operation.

All safety factors mentioned on WENRA Safety Reference Level P2.2 are covered literally and explicitly in SG-01.10, whose development was based on IAEA SSG-25 "Periodic Safety Review for Nuclear Power plants". Two additional safety factors: "Radiological protection of people and workers", and "improvement arrangements" are included on SG-01.10.

Q.10.3 How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

Answer:

Continuous assessment of nuclear power plant safety explicitly includes, among others, operational experience and applicability of new regulation. This is verified in detail on an annual basis (CSN safety guide 1.7). This analysis includes new developments on applicable national regulation, as well as those regulations specific from the country where the technology of the NPP was originated (USA and Germany). However, this continuous assessment is complemented on the PRS to evaluate all regulatory innovations and amendments, included technical standards, having in mind the new period of operation. The ten-year periodic safety review benefits from the annual basis assessment, but insists on checking the degree of compliance with the newest applicable national and international regulation.

Just in the context of PSR, the analysis of current regulation comprises the regulation issued by international organisations, in particular codes and standards from the IAEA as well as national regulation and regulation in force on the nation origin of the technology.

Safety factors (see Q.10.2) and potentially derived safety improvements are reviewed from the point of view of compliance with regulation and good practices. A fundamental constituent of this review is to check the compliance degree with newly applicable requirements, standards and codes.

The title holder will elaborate a list with rules, technical standards and practices applicable to the PSR, with special mention to selection criteria, cutoff date and revision procedure.

Mandatory clauses must in any case be revised to verify compliance.

11. SAFETY CULTURE AND OPERATIONAL EXPERIENCE.

Q.11.1 Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

Answer:

Safety culture promotion and assessment in the Spanish nuclear installations has been part of the CSN regulatory work since early '90 decade, when the CSN specialists worked with nuclear power plants (NPPs) Human and Organizational Factors (HOF) specialists in order to reach a common understanding of the objectives, projects and methodologies to be considered in those programs.

Around 1999-2000 the Spanish nuclear regulatory body, CSN, requested the licensees to implement Evaluation and Improvement Programs on HOF and on Safety Culture (SC), the HOF Program becoming a framework for all the projects and initiatives in the field, including SC and the SC Program. Subsequently, the CSN stablished a biennial inspection to the HOF Programs, being therefore the SC Programs within the scope of these inspections. HOF inspections are part of the CSN NPPs integrated oversight process (known as SISC). The goal of the inspections is to ensure that the licensees have the capability (mandate, processes, resources…) to assess and improve the organization from the HOF point of view. The focus on the inspection of SC Programs is not to assess or determine the state of SC in the plants, but to ensure that each Licensee has in place an appropriate SC Program, with self and independent assessments and action plans adequate to deal with their results.

Around 2010, the CSN developed and introduced a third approach to SC oversight in the SNPPS, as a part of the SISC oversight process, which was called the Crosscutting Components Process (based on the substantive crosscutting issues process in the USNRC ROP). This approach uses the inspection results as the inputs for the SC follow up (as an internal alert threshold, that allows the CSN to take actions when necessary), and includes in the classification of the NPPs based on performance results the consideration that declining performance is linked to organization and cultural issues (triggering therefore regulatory actions).

The CSN has included requirements regarding SC in its regulation:

- o RD 1400/2018 approved "Nuclear Safety in Nuclear Facilities" (art. 7): requires the licensees to:

  - Provide technical, economical and human resources (with adequate qualification) and an appropriate organizational structure to support nuclear safety and emergency response
  - Stablish a Nuclear safety policy that promotes continuous improvement
  - Development an integrated management system, that considers safety an overriding priority, and supports safety and its continuous improvement
  - Include the safety culture improvement and promotion in the management system

- o CSN IS-19 "Integrated Management Systems": require the licensees to promote and improve a healthy SC, including periodic SC assessment (art. 4.2, 5.5.2, 8.2).

- o CSN IS-26 "Basic Nuclear Safety Requirements": require the licensees to stablish a framework for a healthy SC and an integrated management system (art. 3.5, 3.6).

- o CSN GS-1.10 "Periodical Safety Reviews, PSR": require Evaluation and Improvement Programs on Human and Organizational Factors (HOF) and on Safety Culture (SC) (art. 4.7).

Q11.2 Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

Answer:

Approved by RD 1400/2018, "Nuclear Safety in Nuclear Facilities" (RSN) requires the licensees to stablish and maintain a systematic operating experience program (art 32), that includes both internal and external events, to identify, select and implement lessons learned relevant to safety, and report significant events to the regulator.

Operating experience is also considered to be one of the pillars of the continuous improvement that the nuclear safety policy must promote (as required by RSN art. 7.2.b) and one of the aspects to consider in the periodical safety reviews (PSR) (as required in RSN art. 13).

The specific regulatory requirements related to the events reporting criteria are included in CSN IS-10. The CSN holds monthly meetings to review the nuclear installations events reported and their root cause analysis quality, conclusions and proposed corrective actions.

Additionally, the nuclear installations Operating Experience Programs regulatory oversight is part of the CSN inspection process. The acceptance criteria for those programs are described in the inspection procedure, PA-IV-118. Also, operating experience is considered in the integrated oversight system as one of the criteria used to select the components to be included in design basis inspections.

ANNEX

LIST OF REGULATIONS REFERENCED IN THE ANSWERS:

1. RSN: Regulation on Nuclear Safety at Nuclear Installations. Approved by Royal Decree 1400/2018. 23rd November 2018. This Regulation transposes the Euratom Directive 2014/87 to the Spanish regulation.
2. RINR: Regulation on Nuclear and Radioactive Installations. Approved by Royal Decree 1836/1999. Last modified 26th March 2015.
3. PLABEN: Nuclear Emergency Basic Plan. Approved by Royal Decree 1546/2004, Last modified 17th September 2011.
4. IS-10: CSN Instruction IS-10 Establishing the criteria for reporting events on NPP to the CSN 30th July 2014
5. IS-19: CSN Instruction IS-19 Requirements of the management system for nuclear facilities. 22nd October 2008
6. IS-25: CSN Instruction IS-25 Criteria and requirements on the performance of probabilistic safety assessments and their applications for nuclear power plants. 9th June 2010
7. IS-26: CSN Instruction IS-26 on Basic Nuclear Safety Requirements Applicable to Nuclear Installations. 16th June 2010.
8. IS-27: CSN Instruction IS-27 revision 1 on General NPP Design Criteria. 14th June 2018.
9. IS-32: CSN Instruction IS-32, on Plant Technical Specifications of nuclear power plants. 16 November 2011
10. IS-36: CSN Instruction IS-36 on Emergency Operating Procedures and the Management of Severe Accident at NPP. January 21st, 2015.
11. IS-37: CSN Instruction IS-37 on the Analysis of Design Basis Accidents at NPP. January 21st, 2015.
12. Evaluation criteria developed by CSN for Post-Fukushima modifications: “Evaluation Criteria to be considered in Post-Fukushima Design Modifications”. December 18th, 2013.
13. GS-1.10: CSN Safety Guide GS 1.10 revision 2 on Periodic Safety Review at NPP. 30th May, 2017.
14. GS-1.07: CSN Safety Guide GS-1.07 Revision 2 Information to CSN from NPP in operation 9th October 2003
15. 10CFR50 Appendix J: Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
16. KTA-3405: Leakage Test of the Reactor Containment Vessel. 2015-11.

Appendix 11: Answers to the questionnaire - Sweden

Strålsäkerhetsmyndigheten
Swedish Radiation Safety Authority

Promemoria

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Diarienr: SSM2018-6257
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Fastställd: [redacted]

Answer to Questionnaire on Implementation of Articles 8a-8c of Directive 2014/87/Euratom

The present document provides the answers to the questionnaire on national practical approaches. The questionnaire has been issued within the framework of the European Commission project for Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom (EC Contract No.ENER/17/NUCL /S12.769200)

1. General information on the nuclear activities, legislation and regulations In Sweden

Nuclear facilities

At present, in December 2018, there are eight nuclear power reactors in operation in Sweden. The Swedish nuclear sector also includes a fuel factory, two waste storage facilities and one waste treatment facility. There is no longer any research reactor in operation in Sweden.

In July 2012, Vattenfall AB submitted an application for permission to replace one or two old nuclear power reactors with new reactors. However, late 2014 Vattenfall informed that all on-going work related to plans for new build of nuclear power plants have been put on hold and there is currently no intention to resume the project. Instead, Vattenfall AB has taken a principal decision on stepwise phase-out of two of the oldest reactors (Ringhals 1 and Ringhals 2) with following decommissioning process. During 2016 and 2017, OKG Aktiebolag has shut down the two oldest power reactors, Oskarshamn 1 and Oskarshamn 2. From 2021, it will thus be six remaining nuclear power reactors in operation, and they will according to the licensee’s present plans gradually enter into the long-term operation (LTO).

According to licence conditions decided after the European stress tests any Swedish nuclear power reactor operating after 2020 must have an additional and fully independent system in place for cooling of the reactor core. The introduction of an independent core-cooling function (ICC) strengthens the reactor’s ability to prevent core damage for extreme events previously not included in the design basis. The independent core cooling function protects the plant against the events leading to the extended loss of normal auxiliary core cooling function.

National regulatory framework

The following five acts constitute the basic nuclear safety and radiation protection legislation of Sweden:

- The Nuclear Activities Act (1984:3),
- The Radiation Protection Act (2018:396),
- The Environmental Code (1998:808),
- The Act on the Financing of Management of Residual Products from Nuclear Activities (2006:647), and
- The Nuclear Liability Act (1968:45).

Operation of a nuclear facility can only be conducted in accordance with a licence issued under the Nuclear Activities Act as well as with a licence issued under the Environmental Code. The Nuclear Activities Act is mainly concerned with issues of safety and security, while the Environmental Code regulates general aspects of the environment and the possible impacts of “environmentally hazardous activities”, to which nuclear activities are defined to belong.

The Nuclear Activities Act was amended I August 2017 with major provisions in the Council Directive 2014/87/Euratom (NSD).

The objective of the Radiation Protection Act is to protect people and the environment from the harmful effects of radiation. The Act applies to radiation protection in general and, in this context, it provides provisions regarding worker’s protection, radioactive waste management, and the protection of the general public and the environment.

A new Radiation Protection Act was decided in June 2018 and which include transposition of major provisions of the Council Directive 2013/59/Euratom (BSS).

The Swedish Radiation Safety Authority (SSM) is currently revising its new Code of Statutes related to nuclear activities and radiation protection. Experience has demonstrated the need to clarify and broaden the regulations in order to create more predictability for the licensees and to improve the regulatory support. Another reason for this revision is that the Swedish Government has ordered the Authority to develop regulations for potential application in case of construction of new nuclear power plants. The major and thorough review of Codes and Statutes began in 2013. The first parts of the new Code of Statutes was decided and entered into force in in June 2018 and in mid-2020 new regulations on radiation safety (ie, nuclear safety, security and radiation protection) at nuclear power reactors will enter into force.

The structure adopted for the new Code of Statutes means that the radiation safety of nuclear installations will be regulated partly for different stages of a plant's life and partly for main types of specific radiation safety aspects. The regulation will also be made on the "three levels"

Level 1. common to all activities involving ionizing radiation,
Level 2. on plant level for nuclear installations, and
Level 3. more specific radiation safety aspects.

In the meantime, until the full new Code of Statutes has been decided, necessary changes of existing SSM regulations have been made for transposition of those provisions of the Directive 2014/87/Euratom and the Council Directive 2013/59/Euratom which not are regulated by the amended Nuclear Activity Act, the new Radiation Protection Act and related government ordinances.

For further information on legislation and regulations, see Chapter 7 of Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30).

2. General safety approach (safety "philosophy") in Sweden

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety "philosophy") adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

The basic and long-applied general safety philosophy for existing nuclear power plants is based on that measures to increase the level of safety at Swedish nuclear power facilities shall gradually be taken in accordance with new knowledge and experience. New knowledge and experience have emerged from lessons learned from incidents and accidents, from research, from safety analyses and from new reactor designs. International accidents/ incidents such as the TMI nuclear power plant accident in 1979 and the severe accident at the Fukushima Dai-ichi nuclear power plant in 2011 as well as domestic incidents such as the 'strainer event' in Barsebäck unit 2 in 1992 and the electric power system event at Forsmark unit 1 in 2006, have had a major influence on these measures.

This philosophy is also the basis for the provisions of section 10 in the Nuclear Activities Act, which require that anyone who has a license for nuclear activities are responsible for the safety of the facility and its operation and shall continuously and systematically evaluate, verify and, as far as possible and reasonably, improve the safety with regard to

- the circumstances under which the activities are carried out,
- how equipment and facilities are affected by operation and age,
- experiences from operations and similar activities, and
- developments in science and technology.

This philosophy also includes that at licensing of new nuclear facilities the provisions of the Environmental Code regarding the use of best available technology apply.
In line with this basic philosophy, the regulator decided at the end of 2004 on new regulations that led to a comprehensive program for safety improvements of all Swedish nuclear power reactors. These regulations (previously SKIFS 2004:2, currently SSMFS 2008:17) and the general advice on their interpretations entered into force on 1 January 2005 with transitional provisions. These transitional provisions providing the basis for the regulator's decision concerning reactor-specific modernisation programmes, including a timetable for implementation of these programmes. The measures encompassed by the regulatory decisions for each reactor included further improvement of

- physical and functional separation,
- diversification of safety functions,
- accident management measures,
- robustness to local dynamic effects from pipe breaks, and
- resistance to external events

The most resource intensive measures were associated with improving the physical and functional separation and diversification. The modernisation programmes were essentially implemented in 2015.

Further information on the application of this basic safety philosophy can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including, among other things, the compilation in Appendix 2 with information on implemented safety improvements, as well as in previous national reports.

3. Safety objectives for new reactors and for existing ones

Question 3.1: Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors?

As stated above, there are currently no plans to construct new nuclear power plants in Sweden. However, SSMs review of the Code of Statutes includes to develop requirements that apply to both existing and new nuclear power plants. Most of the requirements will be the same for existing and new reactors. Some differences are considered. One important difference concerns radiological acceptance criteria, which most likely will be more stringent for new nuclear power reactors than for existing ones. What values to apply to new nuclear reactors are now under investigation.

Question 3.2: Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?

The basic safety requirements are contained in the Nuclear Activities Act (1984: 3), which was changed in August 2017 when a number of key provisions of the NSD were introduced into the Act. This included the Article 8a, paragraphs (a) and (b).

However, it should be noted in this context that license conditions, decided by the Swedish Government, for all Swedish nuclear power plants are in place since the mid to late 1980s which requires the reactors shall be capable of withstanding a core melt accident without any casualties or ground contamination of significance to the population. In the license conditions it was stated that these requirements can be considered to have been met if a release is limited to a maximum of 0.1 % of the reactor core content of caesium-134 and caesium-137 in a reactor core of 1800 MW thermal power, provided that other nuclides of significance are limited to the same extent as caesium. This resulted in an extensive back fitting for all Swedish nuclear power reactors including the introduction of

- filtered containment venting through an inert Multi Venturi Scrubber System (MVSS) with a decontamination factor of at least 500,
- unfiltered pressure relief in BWRs in the case of large LOCA and degraded pressure suppression function to protect the containment from early over pressurization,
- independent containment spray,
- all mitigating systems designed to withstand an earthquake, and
- a comprehensive set of SAM procedures and guidelines.

It was assumed during back-fitting design that the environmental protection requirements can be met if containment integrity is maintained during accident sequences (core melt scenarios) and that the releases and leakage from the containment can be controlled and limited.

Several potential threats to containment integrity occur during a core melt process. In brief, these can be categorized into the following groups: pressure loads due to gas and steam generation, temperature loads due to the high temperature of the molten core, impulse loads due to hydrogen combustion and the interaction between the molten core and water, concrete removal due to contact between the corium and concrete as well as high temperatures and aggressive materials.

These license conditions, along with other SSM decisions and regulations, also mean that the following acceptance criteria apply to different event categories, which also relates to the different levels in the defence-in depth of existing nuclear power plants. The event categories are ranging from H1 (normal operations) to H5 (events with extensive releases of radioactive substances). Acceptance criteria is given in terms of target values for the maximum dose to workers and general public, and for H5, as already mentioned above, maximum releases to the environment.

The table also shows the relationship to defence-in -depth levels and associated plant condition categories according to WENRA.

Event categories (H1-H5) according to Swedish regulations	Design criteria (mSv) for existing reactors (with respect to the general public)	Defence-in depth- level (DiD)	Associated plant condition categories (according to WENRA)
H1	0,1*	1	NO
H2	1	2/3	AOO
H3	10	3	AOO
H4A	100	3	DBA
H4B	100	3	DEC A
H5	**	4	DEC B

*) Per year and for all nuclear facilities at the site
**) No dose criterion is specified. Instead, here is a maximum permissible emission of Cesium-13, which corresponds to 100 TBq.

Further information on safety objectives and implementation of defence in depth in existing Swedish nuclear power plants can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B18.1. Further information on mitigation systems implemented after the TMI accident can also be found in section 1.2 in Sweden's national report to the Convention on nuclear safety 2012 extra ordinary meeting (Ds 2012:18) and the Swedish National Report on European stress test (December 2011, 11-1471), including section 1.8.

Question 3.3: For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?

See the answer to question 3.1.

Question 3.4: Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.

Resilience to failures and other internal and external events, including natural phenomena and human induced situations and activities, is regulated in Section 14 of SSMFS 2008:17. According to these requirements a nuclear power reactor shall withstand natural phenomena and other events that arise outside or inside the nuclear power reactor and which can lead to a radiological accident.

SSM's expectations, based on the general requirements of the regulations SSMFS 2008: 1, SSMFS 2008:12 and SSMFS 2008: 17, are that the licensees shall

1. identify assumed events and conditions,
2. categorize and divide these into event categories,
3. analyse and assess how events and conditions can develop over a certain period of time, and

4. analyse and assess the resulting radiation protection impacts against acceptance criteria, in terms of doses to workers, general public and releases to the environment.

The identification according to paragraph 1 shall include internal and external hazards, events and conditions, as well as hazards, events and conditions caused by malicious acts that are included in the design basis threat (DBT).

More detailed information about design basis regarding external events of different severity and analysis methodology used can be found in Swedish National Report on European stress test (December 2011, report no. 11-1471), and subsequent reports (National Action Plan 2012 and updated National Action Plans December 2014 and December 2017) with information on actions taken as a result of observed shortcomings. Further information is also available in in Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including sections B17.1-B17.3 and B18.1.

Question 3.5: Article 8a of the Safety Directive 2014/87/Euratom requires “preventing accidents and, should an accident occur, mitigating its consequences and avoiding” “early radioactive releases that would require off-site emergency measures but with insufficient time to implement them” and “large radioactive releases that would require protective measures that could not be limited in area or time”. Please, describe how the terms early and large release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.

See the answer to question 3.2.

Question 3.6: Article 8a of the Safety Directive 2014/87/Euratom requires “timely implementation of reasonably practicable safety improvements to existing nuclear installations”. Please, discuss your national approach for the decision making whether a certain measure is timely implemented and reasonably practicable. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.

This type of assessments and considerations of what is possible and reasonable safety improvements in individual cases includes assessing the need to take action and if a safety improvements is proportionate to the extent of the measures that are possible to implement. Also time and cost aspects may need to be considered. In such considerations, it is important to assess the impact for the nuclear power reactor´s overall safety.

It is consequently case-by-case assessments and considerations that determine the outcome. Major safety improvements that requiring comprehensive measures may take time to implement. This was for example the case in the mid-1980s when the systems for filtered pressure relief of reactor containments were introduced and is also the case now that the independent core cooling system (ICC) is being introduced. On the other hand, events have occurred that have shown the presence of designs with previously unknown deficiencies that requires immediate actions and safety improvements. This was the case after the ‘strainer event’ in Barsebäck unit 2 in 1992 and the electric power system event at Forsmark unit 1 in 2006.

The WENRA Guidance on “Timely Implementation of Reasonably Practicable Safety Improvements to Existing Nuclear Power Plants” along with other internal documents based on the basic and long-applied general safety philosophy (see the answer to question 2 above), guide SSMs assessments and consideration on issues regarding timely introduction of possible and reasonable safety improvements.

Additional information on the work with safety improvements in Swedish nuclear power plants can be found in Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B.18.

Question 3.7: What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?

As stated above in the section on safety philosophy the Nuclear Activities Act requires that licensees continuously and systematically shall evaluate, verify and, as far as possible and reasonably, improve the safety with regard to

- the circumstances under which the activities are carried out,
- how equipment and facilities are affected by operation and age,
- experiences from operations and similar activities, and
- developments in science and technology

SSM's regulations SSMFS 2008: 1 contain related provisions requiring that the continuous and systematic evaluation and verification under the Act also shall include new applicable rules and standards for design, and operation as well as design preconditions that have been changed after commissioning of the plant. The regulations also require that an established safety program shall be provided for the safety improvement measures, both technical and organizational, prompted by the continuous and systematic assessment and verification. The safety program shall be evaluated and updated annually.

All licensees have safety programmes in place as required by SSM regulations SSMFS 2008:1. The programmes are part of the management system documentation. They contain priorities and time schedules for technical, organisational and administrative measures to be implemented as a result of safety analyses, audits, safety culture surveys and other evaluations done at the plant.

SSM conducts oversight of the licensees work with their safety programs as well as their operating experience analysis and feedback programme.

According to the Nuclear Activities Act, an overall assessment of a nuclear safety, security and radiation protection shall be conducted at least every ten years (Periodic Safety Reviews, PSR) in addition to the continuous and systematic evaluation and verification. These overall assessments shall aim at ensuring compliance with the current design basis and identifies further safety improvements. According to the bill on which the PSR provisions of the Nuclear Activities Act are based, the aim is to ensure that older nuclear power reactors as far as possible and reasonably achieve a comparable level of safety as new nuclear power reactors.

For more information on PSR, see the answer to questions in section 10, and further information on operational experience can be found in the answers to the questions in section 11.

More detailed information on safety programmes and operating experience analysis and feedback programme can be found in Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including sections B10.2.2 and B19.2.8.

4. Design and operational aspects of the defence-in-depth

<u>Question 4.1:</u> Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.

Measures for enhanced defense-in-depth and a reinforced independence between levels in the defense-in-depth have been done over a long period of time, including the actions taking after the TMI accident. Measures for further enhance the defense-in-depth by improving the degree of separation and diversification was also a result of the regulations that was decided in 2004.

As stated above in the introduction SSM have decided on licence conditions after the European stress tests which means that nuclear power reactors operating in Sweden after 2020 must have an additional and fully independent system for cooling of the reactor core (ICC) in place. The introduction of an independent core-cooling function becomes a significant reinforcement of the nuclear power reactors defense-in-depth and strengthens the reactor's ability to prevent core damage for extreme events previously not included in the design basis. The independent core cooling function protects the plant against the events leading to the extended loss of normal auxiliary core cooling function.

Construction of ICC is currently underway at the six nuclear reactors that licensees plan to operate after 2020.

Further information on implementation of defence in depth in existing Swedish nuclear power plants can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B18.1. Further information on mitigation systems implemented after the TMI accident can be found in Sweden's national report to the Convention on nuclear safety 2012 extra ordinary meeting (Ds 2012:18). See also the Swedish National Report on European stress test (December 2011, 11-1471), including section 1.8, and subsequent reports (National Action Plan 2012 and updated National Action Plans December 2014 and December 2017) with information on actions taken as a result of observed shortcomings.

Question 4.2: What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).

As stated above in the answer to question 2, new knowledge and experience that have emerged from lessons learned from incidents and accidents, from research, from safety analyses and from new reactor designs have over the years led to safety improvements in Swedish nuclear power plant. Among the more important safety improvements are measures following the TMI accident that led to new license conditions for all Swedish nuclear power plants concerning measures to mitigating and avoiding early and large radioactive releases.

As also stated above in the answer to question 2 the regulator decided in 2004 on new regulations that led to a comprehensive program for safety improvements of all Swedish nuclear power reactors and which were essentially implemented in 2015. The most resource intensive measures were associated with improving the physical and functional separation and diversification.

Additional information on the work with safety improvements in Swedish nuclear power plants can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B.18 and Appendix 2 in which important safety improvements of the Swedish nuclear power plants during the period 1995-2015 are listed.

5. Probabilistic Safety Assessments

Note: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.

Question 5.1: Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:

- PSA level 1:
- PSA level 2:
- PSA level 3:

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?
According to SSMFS 2008:1, all existing Swedish reactors have to be analysed with probabilistic methods to supplement the deterministic safety analyses. All nuclear power reactors have complete level-1 and level-2 PSA studies including all operating modes and all relevant internal and external hazards for the sites. Work has been performed to further develop the studies and to finalize studies for low power operation, area events (internal events) and external hazards.

The basic PSA studies are expected to be updated every year taking into account the past year's plant modifications which have an impact on the PSA-result. In principle the licensees are moving towards applying a so-called "Living PSA" approach. PSA results are also used routinely by the licensees to support decisions concerning significant modification of the designs, modification of operations, documentation and assessment of events.

The numerical PSA figures are not regarded as a definitive and exact value of the actual risk level. There are no requirements related to numerical PSA results, although the licensees have internally developed such safety objectives. The studies are required to be sufficiently detailed, comprehensive and realistic to be able to identify weaknesses in the designs and to be used to assess plant modifications, modifications of technical specifications and procedures as well as assessment of the risk significance of events.

Further information on the use of probabilistic safety assessment in existing Swedish nuclear power plants can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B14.2.3. See also the Swedish National Report on European stress test (December 2011, 11-1471), including section 1.10.

Question 5.2: Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?

As stated in the answer above to question 5.1, the reactors PSAs are expected to include all plant operating states and those internal and external events that may be assumed to occur.

Question 5.3: Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?

The requirements of SSMFS 2008: 1 apply to the entire nuclear power reactor including, inter alia, spent fuel pools. The PSAs that have been developed and are used based on these requirements focus on the stages when fuel management takes place, and fuel is moved from the reactor pressure vessel to the pools and vice versa.

In this context, it should also be noted that in Sweden there is a central interim storage of spent fuel (called Clab) and where the fuel is stored pending encapsulation and disposal in rock formations. This also means that the amount of spent nuclear fuel stored in the reactor's spent fuel pools is limited.

Question 5.4: More generally, do you expect PSAs covering multiple installations at a same site?

PSAs covering multiple installations at a same site is an expectation, but is not yet fully developed. Research and implementation is in progress. To date, the focus has been on functions and systems that are shared between different nuclear power reactors.

Question 5.5: Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.

As stated above in the answer to question 5.1 the licensees are moving towards the concept of "Living PSA" which means that the PSAs are continuously used in enhancing and understanding plant safety status. A PSA is updated or modified when necessary to reflect any physical (resulting from plant modifications, etc.), operational (resulting from enhanced procedures, etc.) or organizational changes. It is documented in such a way that each aspect of the model can be directly related to existing plant information, plant documentation or the analysts' assumptions in the absence of such information.

PSA is used to justify parts of technical specifications (TS) and to identify needs for safety improvements. For example the need for improved separation In the Swedish nuclear power reactors was identified through PSA analyses in many cases.

In the Swedish PWRs have PSAs also been successfully used together with probabilistic fracture mechanics methods to further develop applied program of in-service inspections.

Question 5.6: When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.

See the answers to question 5.1 and 5.5.

Question 5.7: Are PSAs required / developed for new and/or existing research reactors? If so:

- What levels of PSA are performed?
- What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?
- How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?
- Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?

See the answers above for existing reactors.

As stated in the answer to question 3.1, there are currently no plans to construct new nuclear power plants in Sweden. However, SSMs review of the Code of Statutes includes to develop requirements that apply to both existing and new nuclear power plants. Most of the requirements will be the same for existing and new reactors. Some differences are considered. One such difference under consideration is to require PSA level 3 in connection with the licensing of any new nuclear reactor.

6. Design extension conditions without fuel melt

Note 1: For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.
Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.

Question 6.1: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.
As shown in the table in the answer above on question 3.2, DEC without fuel melt (DEC A) is referred to event category H4B which are associated to defence-in-depth level 3.

According to section 22 in the regulation SSMFS 2008:17 realistic analysis assumptions and acceptance criteria may be applied when analysing events that have not been taken into account in the reactor design. This means that independent single failure need not to be fully applied in the analyses and uncertainties may be handled less strict compared with DBA analyses. Furthermore, diversified systems such as automatic supply of borated water and independent core cooling systems (ICC) may be credited.

Question 6.2: What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?

Starting points for the licensees work with event identification are as follows:

The application of requirements according to section 10 in SSMFS 2008: 17 means that events with exceedance frequency of 10E-4 per annum plus common cause failures (CCF) have to be considered. Furthermore, according to the ICC decision Loss of Ultimate Heat Sink (LUHS) for 72 hours and Extended Loss of AC Power (ELAP) for 72 hours are a design basis as well as external events with exceedance frequency of 10E-6 per annum.

Explicit requirements for PSR to include a renewed inventory of events that may lead to DECs without fuel melt are not included in the current regulations, but it is SSM's expectation that this will be done. This will therefore be clarified in the forthcoming regulations that now are being prepared.

Question 6.3: Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.
SSM's current regulations do not impose any explicit requirements on identifying DECs without fuel melt that might happen in the spent fuel pools. However, general provisions according to chapter 4, section 1 in SSMFS 2008:1 requires that the capacity of a facility‟s barriers and defence in depth system to prevent radiological accidents and mitigate the consequences in the event of an accident shall be analysed using deterministic methods before the facility is constructed, or modified and taken into operation. This provision also requires that analyses shall subsequently be kept up to date and that the analyses shall be based on a systematic inventory of events, event sequences and conditions which can lead to a radiological accident.

Assessments and analyses have been done in the context of the stress tests. The integrity and robustness of the spent fuel pools during prolonged extreme situations have been further evaluated and reassessed. The assessments have defined technical and administrative measures to be addressed, e.g. regarding strengthening of the instrumentation and of the water supply to the fuel pools. Further information can be found in 2017 Status Report – Swedish national Action Plan. Response to ENSREG’s request within the European Stress Tests, December 2017.

Question 6.4: What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:

- What are the rules to analyse DEC without fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?
- Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?

See the answer to question 6.1 regarding the requirements for DEC A analysis and the table in the answer to question 3.2 regarding the design criteria that shall be shown to be fulfilled.

According to the regulations SSMFS 2008: 17 and SSM's expectations, systems, structures and components belonging to DiD3 should be independent of systems, structures and components belonging to DiD2 and DiD4 as far as is possible and reasonable. The dedicated systems should be diverse (i.e. independent and separated) from ordinary systems where CCF may occur.

Question 6.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?

According to the practice applied in Sweden for classification, diversified systems, structures and components belong to different safety classes. Parts within the containment belong to safety class 2, while parts outside of the reactor containment may be assigned to safety class 4. However, they must be environmentally qualified regardless of its classification.

The ICC systems that currently being are built as a result of new license conditions shall have independent power supply and redundant valves and pumps regardless of its classification.

Question 6.6: If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?

N/A

7. Design extension conditions with fuel melt

Note 1: For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.
Note 2: The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..

Question 7.1: Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.

In current regulations, SSMFS 2008: 17, the expression “highly improbable events” is used and with the abbreviation H5. This event class is defined as follows:
Events that are not expected to occur; if the event should nevertheless occur, it can result in major core damage. These events are the basis of the nuclear power reactor’s mitigating systems for severe accidents.

See also the table in the answer above on question 3.2.

Question 7.2: What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?

Two postulated events (special events) have been chosen as design basis events for the severe accident mitigating systems as described in the answer to question 3.2:

1. Loss of all AC power and steam-driven emergency core cooling systems for 24 hours (BWRs and PWRs). This is the main design basis event covering events where the core is damaged and measures to mitigate external release from the containment are required. It consists of a loss of all core cooling including loss of all ordinary and alternate back-up AC power supply systems. The loss of core cooling will cause core uncover and subsequent core melt. Since containment cooling is also lost, a pressure build-up will occur in the containment. At a certain pressure value, the filtered containment venting system will be activated in order to protect the containment against overpressure.
2. Large LOCA in combination with degraded pressure suppression function (for only BWRs). This is the design basis event with respect to early containment overpressurization in the BWRs. The large LOCA causes a rapid pressure build-up in the containment but it does not affect the emergency core cooling or the electricity supply. The maximum amount of radioactive material available for release in this case will thus be equivalent to the content of the primary water during normal operation as specified by the technical specifications.

The European stress test identified a need to re-assess time aspects for the design basis events according to 1 above, and to evaluate the use of the containment filtered venting system during prolonged severe accident conditions (more than 24 hours).
No other circumstances have emerged so far that gives reason to reconsider the validity of these two bounding or enveloping scenarios as design base events for the severe accident mitigating systems.

Further information and motivation of these design basis events for the severe accident mitigating systems can be found in Sweden’s national report to the Convention on nuclear safety 2012 extra ordinary meeting (Ds 2012:18) and the Swedish National Report on European stress test (December 2011, 11-1471), including section 1.8, and subsequent reports (National Action Plan 2012 and updated National Action Plans December 2014 and December 2017) with information on actions taken as a result of observed shortcomings.

Question 7.3: Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.
SSM's current regulations do not impose any explicit requirements on identifying DECs with fuel melt that might happen in the spent fuel pools. However, general provisions exist in chapter 4, section 1 SSMFS 2008:1. See the answer above on question 6.3.

Question 7.4: What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- What are the rules to analyse DEC with fuel melt?
- Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?

- Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?

As stated above realistic analysis assumptions and acceptance criteria may be applied when analysing events that have not been taken into account in the original reactor design. This means that independent single failure need not to be fully applied in the analyses and uncertainties may be handled less strict compared with DBA analyses. See also the answer to question 3.2 regarding the provisions for DECs with fuel melt and the design criteria for these systems that shall be shown to be fulfilled.

Question 7.5: What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?

Structures and components of the severe accident mitigating systems introduced as stated in answer to question 3.2 belong to different safety classes. Parts within the containment belong to safety class 2, parts between insulation valves and MVSS building belong to safety class 3 and building with MVSS equipment belongs to safety class 4. The systems are designed and constructed with requirements of redundant insulation valves and power supply from two different power lines, which are independent of ordinary auxiliary power system.

Question 7.6: If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?

N/A

8. Design of the containment

Question 8.1: What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?

According to Section 3 in SSMFS 2008:17 a nuclear power reactor shall be designed so that the safety functions of reactivity control, protection of the primary system integrity, emergency core cooling, residual heat removal and the containment function can be maintained to the extent needed depending on the operational state during all events up to and including the event class improbable events (H4). This event class corresponds to DBA.

In the case of boiling water reactors, the containment function refers to its leaktightness function and pressure suppression function. For pressurized water reactors, containment function refers to the leaktightness function.

According to Section 5 in SSMFS 2008:17 a reactor containment shall in addition to requirements according to Section 3 be designed taking into account phenomena and loads that can occur in connection with events in the event class highly improbable events (H5) to the extent needed in order to limit the release of radioactive substances to the environment. The H5 event class corresponds to DEC B.

To meet the requirement in Section 5, a safety evaluation should according to SSMs general advice in SSMFS 2008:17 be performed of events and phenomena which may be of importance for containment integrity in highly improbable events (H5). Examples of such events and phenomena which can result in the need to take measures include high pressure melt-through of the reactor pressure vessel, steam explosion, recriticality, hydrogen fire and containment underpressure.

Relevant provisions in the license conditions decided after the TMI accident also apply. Se the answer to question 3.2.

Question 8.2: How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?
Containments in Swedish nuclear power reactors were originally designed to maintain its leaktightness function, and pressure suppression function in BWR´s, in the event of an LB-LOCA. This resulted in design pressures of the containments, depending on reactor type and size, ranging between 5 and 6 bar(a). In connection with action taken in the mid to late 1980s concerning measures to mitigating and avoiding early and large radioactive releases, and also later based on results from severe accident research, verifying analyses have been made to clarify the containments capabilities to withstand higher loads. In these analyses, with the modern methods, the containments were verified for significantly higher pressure, 8-9 bar(a), than was analysed at the time of design.

Question 8.3: What is the approach for containment penetrations? Is their leak tightness regularly tested?

The containments leak tightness including penetrations, locks and isolation valves are verified periodically by pressure tests. Test methodology, frequency and scope, which are governed by the respective plants technical specification (TS), are based on the US NRC regulatory framework, option A of Appendix J of 10CFR50. This means that the tests for the different containment components take place at predetermined time intervals. The reactor containment (test type A) are normally tested three times evenly divided into ten years. Penetrations (test type B) and isolation valves (test type C) are tested in conjunction with opening or at least once every two years in accordance with the basic requirement. Locks (test type B) are tested at 6 months intervals or in conjunction with each opening.

If necessary, further tests and examinations are carried out, for example by visual inspection and non-destructive testing. This has been the case when pressure tests showed signs of damage or other defects.

9. Practical elimination

Question 9.1: Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.
The term practical elimination is not used for existing nuclear reactors in Sweden. The environmental protection requirements for the reactors follows from the answer to question 3.2 above.

Question 9.2: Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):

The environmental protection requirements for existing nuclear reactors in Sweden follows from the answer to question 3.2 above. See the answer to question 3.1 regarding any new nuclear reactors in Sweden.

Question 9.3: Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.

The environmental protection requirements for existing nuclear reactors in Sweden and action taken to meet these requirements follows from the answer to question 3.2 above. In addition to these measures, each reactor that will be in operation beyond 2020 shall according to decided license conditions have an additional and fully independent system for cooling of the reactor core in place (ICC).

Question 9.4: What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?

As stated above the term practical elimination is not used for existing nuclear reactors in Sweden. The environmental protection requirements for the reactors follows from the answer to question 3.2 above.

Question 9.5: Is the practical elimination concept applicable also for fuel storage pools?
As stated above the term practical elimination is not used for existing nuclear reactors in Sweden. The environmental protection requirements for the reactors follows from the answer to question 3.2 above.

10. Periodic safety review

Question 10.1: Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?
Periodic safety reviews (PSR) started in Sweden in the early 1980s as a result of the Three Mile Island accident. The requirements regarding the reviews have developed over the years and are now comparable to those recommended in the IAEA safety standards.

As stated above in section 2 on safety philosophy and in the answers to question 3.7, the Nuclear Activities Act requires the licensees to both continuously and systematically evaluate, verify and, as far as possible and reasonably, improve the safety, and to conduct an overall assessment at least every ten years through a PSR. SSM regulations SSMFS 2008: 1 contain more detailed provisions and guidance on PSR including which areas to be covered. See further in the answer to question 10.2

Question 10.2: What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:

- Plant design
- Actual condition of structures, systems and components (SSCs) important to safety
- Equipment qualification
- Ageing
- Deterministic safety analysis
- Probabilistic safety assessment
- Hazard analysis
- Safety performance
- Use of experience from other plants and research findings
- Organization, management system and safety culture
- Procedures
- Human factors
- Emergency planning
- Radiological impact on the environment

According to the Nuclear Activities Act PSRs shall aim at ensuring compliance with applicable provisions in legislations and regulations as well as the current design basis, and identify needs of further improvements of safety, security and radiation protection. According to the government bill on which the PSR provisions of the Nuclear Activities Act are based, the aim is to ensure that older nuclear power reactors as far as possible and reasonably achieve a comparable level of safety as new nuclear power reactors. The PSRs shall also verify the prerequisites for safe operation until the next PSR. The analyses, assessments and proposed measures as are results of the RSR shall be submitted to SSM.

According to current rules in SSMFS 2008: 1, a PSR should include analyses and assessment of 17 defined areas as well as integrated and overall assessments. Analyses and assessments should be made of how systems, structures, components and activities in each area meet both regulatory requirements and internal requirements at the time of the analysis and if the technical and other solutions applied continue to be able to prevent such possible deficiencies in barriers and the in the defence-in-depth that could lead can lead to a radiological accident situation. In addition, a systematic analysis should be performed of how systems, structures, components and activities in each area meet relevant new safety standards and practices. Action needs arising from these analyses should be listed and their safety significance evaluated using deterministic and, where appropriate, probabilistic methods, or where this is not possible or reasonable by expert assessment with specified criteria.

How these 17 areas relate to the 14 areas (Safety Factors) specified in WENRA SRL Issue P.2.2 and in IAEA SSG 25 are shown in the table below. In the ongoing review of the SSMs Code of Statutes, an adaptation will be made to areas of WENRA SRL and IAEA SSG 25, but with the difference, that Swedish PSR will continue to cover physical protection/security aspects as well as nuclear non-proliferation, exports control and transport safety.

Safety Factor in WENRA SRL and IAEA SSG-25		Areas according to SSMFS 2008:1	
No.		No.	
1	Plant design	1 5 16	Design and construction of facilities, including modifications Core and fuel issues and criticality issues On-site radiation protection
2	Actual condition of structures, systems and components (SSCs) important to safety	1 5 13 16	Design and construction of facilities, including modifications Core and fuel issues and criticality issues Archiving, handling of plant documentation On-site radiation protection
3	Equipment qualification	1 7	Design and construction of facilities, including modifications Maintenance, including materials and inspection issues with special consideration of degradation due to ageing
4	Ageing	7	Maintenance, including materials and inspection issues with special consideration of degradation due to ageing
5	Deterministic safety analysis	11	Safety analyses and safety analysis report
6	Probabilistic safety assessment	11	Safety analyses and safety analysis report
7	Hazard analysis	11	Safety analyses and safety analysis report
8	Safety performance	4	Operations, including handling of deficiencies in barriers and the defence-in-depth

		8 12 16	Primary and independent safety review, including the quality of notifications to SSM Safety programme On-site radiation protection
9	Use of experience from other plants and research findings	9 12	Investigation of events, experience feedback and external reporting Safety programme
10	Organization, management system and safety culture	2 16	Organisation, management and control of the nuclear activity On-site radiation protection
11	Procedures	2 16	Organisation, management and control of the nuclear activity On-site radiation protection
12	Human factors	2	Organisation, management and control of the nuclear activity
13	Emergency planning	6	Emergency preparedness
14	Radiological impact on the environment	17	Radiation protection of general public and the environment
		10	Physical protection/Security
		13	Archiving, handling of plant documentation
		14	Management of nuclear material and radioactive waste
		15	Nuclear non-proliferation, exports control and transport safety
		16	On-site radiation protection

Additional information on PSR of nuclear power plants can be found in Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B14.3.2.

Question 10.3: How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?

As stated in the answer above to question 10.2, the requirements of SSMFS 2008: 1 mean that for each of the 17 areas, to the extent applicable, should be included to systematic analyse and assess how systems, structures, components and activities meet relevant new safety standards and practices, and take action when necessary to the extent possible and reasonable. This is also one purpose of the PSR provisions in the Nuclear Activities Act.

Additional information on PSR of nuclear power plants taking into account new safety standards can be found in Sweden’s seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including section B14.1.1 and B14.3.2.

11. Safety culture and operational experience

Question 11.1: Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?

SSM's regulations SSMFS 2018: 1 provides that the management system should support and promote a culture so that issues relevant to radiation safety receive the attention and priority that their importance requires. The provision refers to both Safety Culture and Security Culture. With this provision, SSM emphasize thus the simultaneous and balanced considerations that need to be addressed for nuclear safety, nuclear security and radiation protection.

Regulations and policies that provided due priority to safety can be understood as normal safety policies and safety strategies but also safety management provisions and tools to manage a nuclear power plant in such a way that safety is prioritised and a good safety culture is created and maintained. A good safety culture that gives safety issues the attention warranted by their significance, is also a prerequisite for a robust implementation of a management system.

Maintaining a strong safety and security culture in the operation of nuclear plants is considered vital by both SSM and the Swedish utilities, and is emphasised in the policies of the different plants and in their strategic plans. Management at all levels, including the managing directors, is involved in activities to enhance the safety and security culture and to stress the responsibility of all personnel to work actively in maintaining and developing the safety and security culture standard.

SSM follows the licensees work with safety and security culture issues mainly through its regular inspections. The role of SSM in this context is to ensure that the licensees have proactive safety management. SSM expects the licensees to create and maintain a strong safety culture. It is essential that the licensees react in a timely manner to indications of deficiencies in their safety culture. If such deficiencies are not corrected, the ability of the operating organisation to handle difficult situations and maintain safety will deteriorate.

More detailed information on the Swedish licensees work with safety culture and SSMs oversight of this work can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including sections B10.2.1. B10.2.5 and B10.3.

Question 11.2: Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.

The objective of the operating experience analysis and feedback programme in Sweden is to learn from experience, from the licensees one's own plant and from other plants, and to prevent recurrences of events, particularly events that might affect plant safety. The operating experience process consists of a wide variety of activities within plants organisations as well as externally.

SSM imposes strict requirements, in SSMFS 2008:1 and SSMFS 2018:1, for both incident event reporting and systematic investigations and analyses of events. The event sequence must be fully clarified, including circumstances that might have prevented or stopped the sequence, causes and root causes are to be identified, and the consequences clarified and the measures defined to prevent recurrence. MTO analysis is used when root causes and in-depth analysis are deemed relevant. MTO analysis is an established methodology executed by a team of trained investigators available at all plants.

More detailed information on the Swedish licensees work with operating experience analysis and feedback programme and SSMs oversight of this work can be found in Sweden's seventh national report under the Convention on Nuclear Safety (Ds 2016:30), including sections B19.2.7 - B19.2.12.

Appendix 12: Answers to the questionnaire - United Kingdom

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Our Reference: 2018/408027
Unique Number:

Your Reference:
Unique Number:

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Germany

Date: 11 January 2019

Dear Dr Nünighoff

Questionnaire on national practical approaches with respect to Articles 8a-8c of Directive 2014/87/Euratom

The Office for Nuclear Regulation (ONR) is committed to support continuous improvement of international regulatory standards, in collaboration with other regulatory authorities and international bodies. Through this work we aim to support achievement of consistent, high standards of safety and security across the globe.

ONR's support to development of international relevant good practice in respect to timely implementation of safety improvements to nuclear power plants and implementation of the Article 8(a) to 8(c) of the Nuclear Safety Directive is prioritised and directed towards WENRA's Reactor Harmonisation Group (RHWG); of which ONR is an active member. We therefore fully support on-going dialogue between WENRA and the European Commission with a view to achieving alignment between each organisation's work in this area.

Annex A provides our response to your questionnaire on "practical approaches with respect to the Articles 8a-8c". In our response we have referred to an extensive range of publications available via our website which explain our regulatory requirements for nuclear power plants (both existing facilities and new plant) and the provision of safety analysis for such facilities. I encourage your team to review these references to understand the UK approach and how this helps ensure UK's full implementation of Article 8a – 8c. In particular, I would like to highlight the following key references which provide examples of practical application:

- The Generic Design Assessment of the UK Advanced Boiling Water Reactor (UK ABWR) took place in the period following the accident at Fukushima, undertaken against ONR's Safety Assessment Principles (SAPs), IAEA SSR-2/1 rev 1 and WENRA RHWG's "Safety of new NPP designs". ONR's GDA UK ABWR Step 4 reports on Fault Studies (which includes DEC-A), Severe Accidents (which includes DEC-B and Practical Elimination) and PSA (which considers Level 1, 2 and 3 PSA) are available for review from our website:

http://www.onr.org.uk/new-reactors/uk-abwr/reports.htm

- The UK National Report to the 7th Convention on Nuclear Safety explains the UK's approach in respect to practical elimination of large and early releases.

The document also explains how ONR is applying this approach at the Hinkley Point C nuclear power plant currently under construction: https://assets.publishing.service.gov.uk/government/uploads/system/uploads/attachment_data/file/585921/BEIS_Document_Template_CNS_7TH.pdf

Should you require any further details in relation to our response, please do not hesitate to contact [redacted] [redacted]

Yours sincerely

[redacted]

[redacted]

Enclosures:

Annex A - ONR's Response to ETSON Questionnaire on national practical approaches with respect to the Articles 8a-8c of Directive 2014/87/Euratom

Distribution

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Annex A – ONR’s Response to ETSON Questionnaire on national practical approaches with respect to the Articles 8a-8c of Directive 2014/87/Euratom

In this Annex, ETSON’s questions are presented in italics. ONR’s response is given in blue text.

1 GOAL, SCOPE AND EXPECTED ANSWERS

Section 1 of the ETSON provides an overarching introduction and context.

No response is required.

2 General safety approach (safety “philosophy”)

Prior to answering the more detailed questions in the following chapters, describe the general safety approach (safety “philosophy”) adopted in your national framework to ensure safety and its continuous improvement. Emphasis on how this philosophy may contribute to complying with Articles 8a-8c of Directive 2014/87/Euratom would be welcome.

The Energy Act 2013 created the Office for Nuclear Regulation (ONR) as the safety and security regulator for nuclear sites in Great Britain. The UK operates a predominantly goal-setting approach to nuclear safety regulation and is moving towards an aligned approach for security regulation across the civil nuclear sector. This means that ONR sets its regulatory expectations, and requires dutyholders to determine how best to achieve them and justify their chosen approach. This enables dutyholders to be innovative and flexible in how they achieve the high standards of nuclear safety and security required by implementing arrangements that meet their particular circumstances. It also strengthens accountability and encourages the adoption of relevant good practice and continuous improvement.

The Health and Safety at Work Act 1974, which requires dutyholders to ensure the health and safety of employees and members of the public So Far As Is Reasonably Practicable (SFAIRP), provides the legal basis for our goal-setting approach. Ensuring health and safety SFAIRP is the legal basis behind reducing risks As Low As Reasonably Practicable (ALARP).

In judging whether risks have been reduced ALARP, control measures are assessed against Relevant Good Practice (RGP). Sources of RGP include: Approved Codes of Practice (ACOPs); ONR Safety Assessment Principles (SAPs, Reference 1); ONR Technical Assessment Guides (TAGs); ONR Technical Inspection Guides (TIGs); and publications of the British Standards Institute (BSI), International Atomic Energy Agency (IAEA), and Western European Nuclear Regulators’ Association (WENRA). Another important source of relevant good practice in the nuclear industry is what other facilities (including non-nuclear, for example the major hazards industry) have done. (Reference 2)

The Nuclear Installations Act 1965 requires the ONR to attach conditions to a nuclear license. A standard set of 36 license conditions has become established (Reference 3). Within the license conditions are a set of primary powers (Direct, Approve, Notify, Specify, Agree, and Consent). Additional derived powers may be arranged with a licensee if this is a convenient working arrangement. The primary and derived powers are used to specify permissioning milestones, such that the licensee requires permission from ONR to start, continue or cease key activities.

Regulations made under the above Acts of Parliament provide specific and targeted legislation. Example regulations include the Radiation Emergency Preparedness and Public Information Regulations 2001 (REPPIR); Provision and Use of Work Equipment Regulations 1998 (PUWER); Lifting Operations Lifting Equipment Regulations 1998 (LOLER); The Management of Health and Safety at Work Regulations 1999 (MHSWR); Construction Design and Management Regulations 2015 (CDM); The Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009; and the Nuclear Industry Security Regulations 2003 (NISR). The Ionising Radiation Regulations 2017 define specific requirements for work with radioactive materials.

Figure 1 illustrates the key elements of ONR regulatory decision making. This incorporates relevant law and policy. The legislative framework is informed by European Commission (EC) Directives and international conventions. The ONR Enforcement Policy Statement complies with the UK Regulators' Code and the regulatory principles required under the Legislative and Regulatory Reform Act 2006.

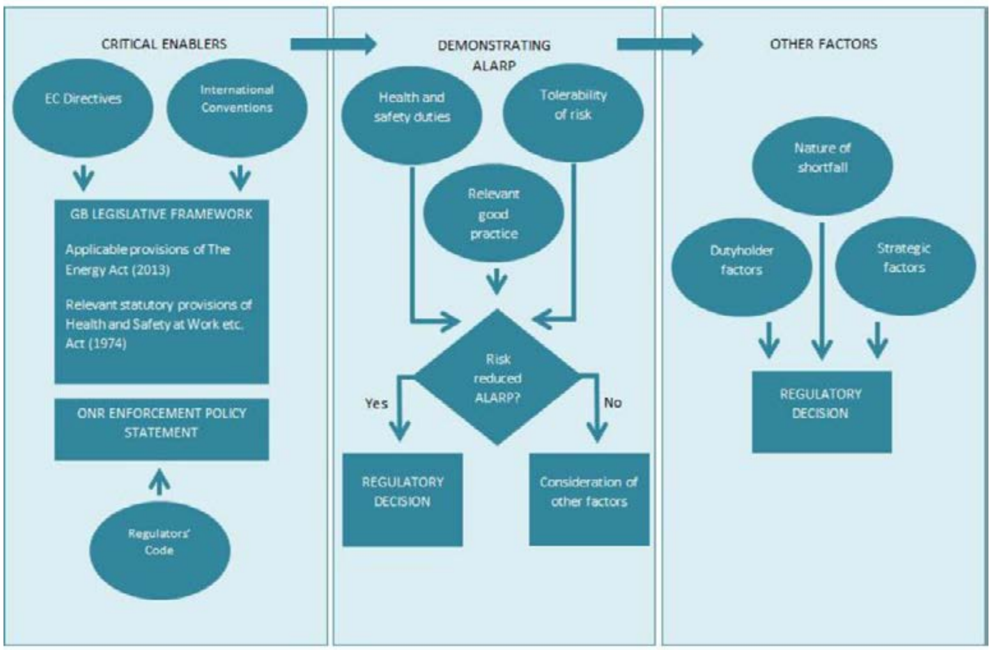

Figure 1 Overview of ONR Regulatory Decision Making

The UK has a mature nuclear industry with established terminology. Where equivalent concepts are expressed in different vocabulary within the international forum, then equivalence is often highlighted in the appropriate ONR guidance, for example in the ONR ALARP Technical Assessment Guide (Reference 5). Furthermore, Reference 5 identifies that the IAEA Safety Standards and the WENRA Safety Reference Levels / Safety Objectives should be considered to be UK RGP. Therefore, although international terminology is not always adopted, the requirements of the IAEA and WENRA are always encompassed within the UK regulatory framework. This includes consideration of design extension conditions and a demonstration of practical elimination.

License condition 23 requires dutyholders to produce an adequate safety case to demonstrate the safety of operation and to identify the conditions and limits necessary in the interests of safety (Reference 3). The starting point for demonstrating that risks are ALARP and safety is adequate is that the normal requirements of good practice in engineering, operation and safety management are met. This is a fundamental expectation for safety cases. The demonstration should also set out how risk assessments have been used to identify any weaknesses in the proposed facility design and operation, identify where improvements were considered and show that safety is not unduly reliant on a small set of particular safety features. (Reference 1)

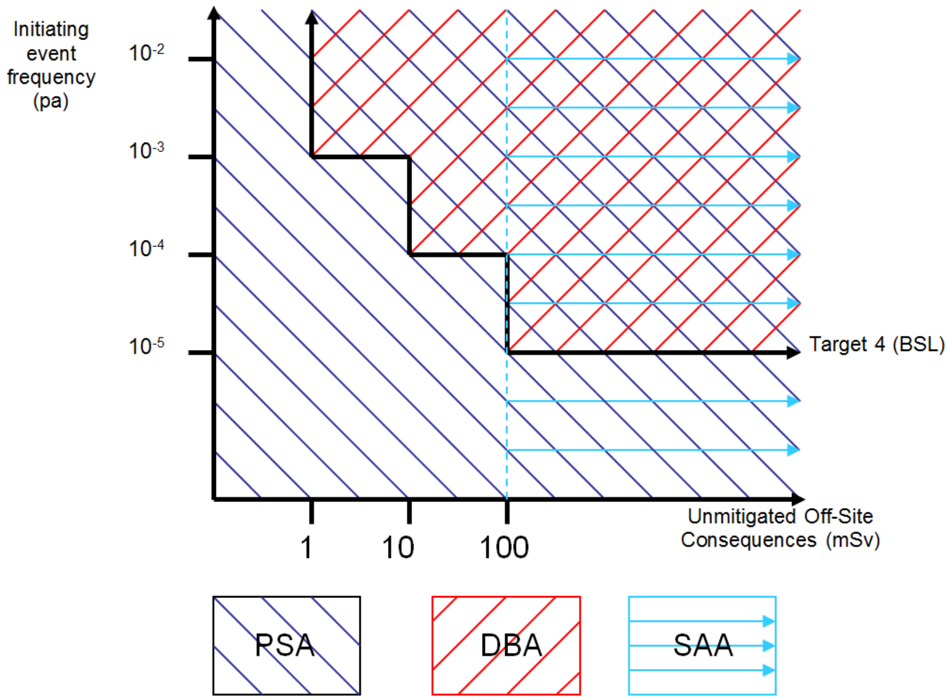

Figure 2 Full and Multiple Coverage of All Potential Scenarios in the UK Approach to Fault Analysis

The ONR SAPs for fault analysis are summarised graphically in Figure 2. All potential scenarios which may lead to any person receiving a significant dose of radiation are covered by at least one form of fault analysis; and all areas of concern (based on dose and frequency) are covered by multiple forms of analysis. Design Basis Analysis (DBA) is required to demonstrate that a physical barrier to the escape of radioactive material remains intact; that there is no release of radioactivity; and that no person receives a significant dose (as quantified in numerical target 4). Thus, all events in this region have robust, engineered solutions to prevent a significant off-site release. Beyond Design Basis Analysis (BDBA) is conducted to inform the Probabilistic Safety Analysis (PSA) and this is judged against numerical targets 5 to 9 in order to limit off-site dose to a tolerable magnitude and frequency (Reference 1). Rigorous application of DBA and PSA ensures that the predicted risks from fault sequences leading to significant radiological consequences are very low. Despite this, Severe Accident Analysis (SAA) is conducted to help plan for potential severe accidents and to assist with identifying what further plant, equipment and human actions are required, particularly with respect to containment and the mitigation of consequences. Thus, ONR Inspectors assess safety case submissions (as described in Article 8c) against Relevant Good Practice (including the SAPs and established international guidance) and thereby ensure that the 8a safety objective has been implemented in accordance with Article 8b.

3 Safety objectives for new reactors and for existing ones

Question 3.1: *Do you require more demanding safety objectives (and/or a more demanding demonstration that the objectives are fulfilled) for new reactors than for existing reactors? If so:*

- *please describe your understanding of paragraph 2 of Article 8a of NSD regarding the relevance of differences between new and existing reactors from a technical point of view;*
- *please provide concrete examples of more demanding objectives and/or demonstration.*

Article 8a requires the avoidance of large and early releases for new reactors (defined as a reactor with a construction license granted after 14th August 2014). For existing reactors Article 8a further requires the timely implementation of reasonably practicable safety improvements to avoid early and large releases.

ONR's regulatory expectations (safety objectives) are defined in our Safety Assessment Principles (SAPs), which can be found at http://www.onr.org.uk/saps/index.htm (Reference 1). The SAPS apply to assessments of safety at existing or proposed new nuclear facilities and while we expect new facilities to full satisfy the intent of the SAPs, facilities designed and constructed to earlier standards may not satisfy the SAPs to the same extent. In this case, the SAPs are used as a basis to judge whether risks have been reduced As Low As Reasonably Practicable (ALARP) in accordance with the Health and Safety at Work Act 1974, Sections 2 and 3.

The UK operates a second generation Pressurised Water Reactor (PWR). The UK is constructing two new PWRs at Hinkley Point C. Hinkley Point C is based on the third generation EPR design with associated reduction in core damage frequency. In the extremely unlikely event of a core melt, Hinkley Point C provides enhanced provisions for mitigating the consequences of severe accidents. The corium will collect in a core catcher inside a double containment structure which is protected by a spray cooling system. The molten core will therefore be stabilised and cooled, preventing and mitigating a potential severe accident (Reference 6). Hence, Hinkley Point C represents a concrete example of more demanding safety standards for new reactors, in comparison to existing reactors.

ONR apply Generic Design Assessment (GDA) to new reactors intended for construction and operation in Great Britain. In 2017 the ONR completed the GDA of the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Reference 7 is the ONR fault studies assessment and Reference 8 is the ONR severe accident assessment. ONR define IAEA Safety Standards and WENRA publications as UK RGP in the ALARP TAG (Reference 5). Therefore, the UK ABWR was assessed against IAEA SSR-2/1 (Reference 9), IAEA SSG-2 (Reference 10), and the WENRA report regarding the safety of new nuclear power plant designs (Reference 11). Thus design extension conditions and practical elimination were directly considered during the assessment.

Question 3.2: *Considering Article 8b of NSD for achievement of safety objectives, for existing reactors, what are the safety objectives associated with the different levels of defence-in-depth, notably Design extension conditions (DECs) - Are there radiological acceptance criteria (general, qualitative or quantitative) and technical acceptance criteria (general, qualitative or quantitative) defined for the different levels of defence-in-depth (e.g. limitations in area and time of population protective actions)? Is there a need for protective actions for the population?*

The ONR SAPs set an expectation for five layers of defence in depth:

1. Prevention of abnormal operation and failures by design.

2. Prevention and control of abnormal operation and detection of failures.
3. Control of faults within the design basis to protect against escalation to an accident.
4. Control of severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents.
5. Mitigation of radiological consequences of significant releases of radioactive material.

The ONR SAPs define Design Extension Conditions as accident conditions that are not considered for DBA, but are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.

Engineering reactor core safety assessment principle 1 requires that the design and operation of the reactor ensure the fundamental safety functions (control of reactivity, cooling, and confinement of radioactive material). This principle covers normal operation, refuelling, testing, shutdown, and design basis fault conditions including re-criticality following an event. Analysis of design basis sequences should use appropriate tools and techniques, and be performed on a conservative basis.

Safety Assessment Principle FA.15 relates to Severe Accident Analysis (SAA) and states that faults beyond the design basis that have the potential to lead to a severe accident should be analysed. SAA is usually (though not exclusively) performed on a best-estimate basis and its starting point is the degraded plant state following an event, rather than the event itself. The main aims of SAA are to help plan for potential severe accidents and to assist with identifying what further plant, equipment and human actions are required beyond what has been identified through DBA and PSA.

BDBA aims to confirm the absence of cliff edge effects on the edge of the design basis, and in this regard is similar to DEC A analysis (design extension conditions without severe fuel damage). SAA involves a degraded plant state and the mitigation of accident consequences, and in these regards SAA is similar to DEC B analysis (design extension conditions with severe fuel damage).

IAEA’s draft guidance on deterministic safety analysis (SSG-2) identifies three broad categories of DEC A events:

- Anticipated operational occurrences or frequent design basis accidents combined with multiple failures (e.g. common cause failures in redundant trains) that prevent the safety systems from performing their intended function to control the postulated initiating event.
- Initiating events that could lead to situations beyond the capability of safety systems that are designed for design basis accidents.
- Credible postulated initiating events involving multiple failures causing the loss of a safety system while this system is used to fulfil its function as part of normal operation. This applies to those designs that use, for example, the same system for heat removal both in accident conditions and during shutdown.

With regard to the first category, it is long standing UK safety case practice to demonstrate diverse safety provision for frequent faults as part of the design basis (i.e. conservatively). In many cases, additional levels of defence in depth are also credited, in which case analysis to support PSA success criteria will provide additional demonstrations of effectiveness.

With regards to the second category, it is long standing UK safety case practice to demonstrate that there are no cliff-edges in the design basis analysis, including the severity of the initiating event. Additionally, PSA considers events more severe than those considered in the design basis safety case, and post-Fukushima stress tests considered the margins to failure following extreme hazards.

With regard to the third category, it is long standing UK safety case practice to consider failures of support systems, operational systems and common cause failures within both the DBA and PSA.

In the most recent assessments of new reactors designs (UK ABWR and UK HPR1000), there is an expectation that the robust UK 'traditional' approach is supplemented by an explicit deterministic treatment of DEC A events consistent with SSR-2/1 (Reference 9).

There is a long standing requirement in ONR's SAPs for severe accident analysis to be performed for events or sequences associated with a significant release of radiation or an unintended relocation of a substantial quantity of radioactive material. This corresponds to DEC B analysis. For new reactors (for example HPC, UK ABWR, UK HPR1000), there is a more explicit expectation that large or early releases will be practically eliminated, consistent with IAEA SSR-2/1 (Reference 9).

In terms of radiological acceptance criteria the SAPs contain 9 numerical targets. Dose limits are defined for individual scenarios, and collective frequency limits are specified for dose bands. Dose bands are related to expected off-site actions including restrictions on foodstuffs, sheltering, issue of stable iodine, and evacuation of nearby population.

Article 8b requires organisational structures for on-site emergency preparedness and response to be in place. License condition 11 states that the licensee shall make and implement adequate arrangements for dealing with any accident or emergency arising on the site and their effects. Fundamental Safety Assessment Principle FP.7 states that arrangements must be made for emergency preparedness and response in the case of nuclear and radiological incidents.

Question 3.3: *For new reactors, do the safety objectives associated with the different levels of defence-in-depth differ from those for existing reactors?*

Tolerability of risk has been translated into 9 numerical targets in the ONR SAPs. These are in the form of Basic Safety Levels (BSLs) and Basic Safety Objectives (BSOs). The BSLs represent the boundary between unacceptable risk and tolerable risk. The BSOs represent the boundary between broadly acceptable risk and tolerable risk. It is ONR policy that the level of risk from a new facility or activity should at least meet our BSLs. However, meeting our BSLs is a minimum expectation and for new facilities we expect the application of relevant good practice to result in a level of risk which can be demonstrated to meet our BSOs, in addition to being demonstrated ALARP. (Reference 2)

Ageing of facilities could, in theory, result in BSLs being exceeded for existing reactors. In these cases, with the exception of the BSLs that are legal limits, it may be reasonable for operation to continue if it has been shown that no reasonably practicable options are available to reduce risks further in the short term; and a clear long term plan to manage and reduce risks as soon as reasonably practicable is in place. Any such facility would attract considerable regulatory attention to secure timely implementation of safety improvements.

Question 3.4: *Regarding Article 8b of the Safety Directive 2014/87/Euratom, do you have specific requirements for the different external hazards (including rare and severe hazards, and their combinations)? You may illustrate with the most relevant elements of the safety demonstration both for design basis external hazards and for rare and severe external hazards.*

The EHA set of safety assessment principles (EHA.1 to EHA.17) pertain to internal and external hazards. EHA.1 requires an effective process to identify and characterise all external and internal hazards that could affect the safety of the facility. The identification process should include reasonably foreseeable combinations of independently occurring hazards, causally-related hazards and consequential events resulting from a common initiating event. Further principles apply to specific external hazards, for example: EHA.8 aircraft crash, EHA.9 earthquakes, EHA.10 electromagnetic interference, EHA.11 weather conditions, EHA.12 flooding. EHA.18 requires beyond design basis events initiated by internal and external hazards to be analysed via the application of an appropriate combination of engineering, deterministic and probabilistic assessments.

The output of the fault analysis of events initiated by hazards, in combination with the output of fault analysis initiated by plant failures (including human error), is compared with the numerical targets defined in the ONR SAPs (Reference 1).

Question 3.5: *Article 8a of the Safety Directive 2014/87/Euratom requires "preventing accidents and, should an accident occur, mitigating its consequences and avoiding" "early radioactive releases that would require off-site emergency measures but with insufficient time to implement them" and "large radioactive releases that would require protective measures that could not be limited in area or time". Please, describe how the terms early and large release are applied in your country, and discuss your national approach to demonstrate that early and large release are avoided according to Article 8a.*

The need to practically eliminate large or early releases is now established in UK and international guidance, but the means to demonstrate it remains an area of international development. ONR is actively involved in both IAEA and WENRA working groups to develop such guidance. ONR has assessed and accepted arguments on practical elimination for both the UK EPR and UK ABWR (Reference 12). The UK's National Report to 7th Convention on Nuclear Safety should be reviewed for the EPR (HPC) design (Reference 22) and ONR's Step 4 SAA Step 4 report should be reviewed for the UK ABWR design (Reference 8).

Question 3.6: *Article 8a of the Safety Directive 2014/87/Euratom requires "timely implementation of reasonably practicable safety improvements to existing nuclear installations". Please, discuss your national approach for the decision making whether a certain measure is timely implemented and reasonably practicable. Please, specify which type of information (technical or non-technical) is required and/or used for such an assessment.*

ONR's TAG on periodic safety reviews (Reference 14) states the process should identify any shortfalls against modern standards and good practices, with a programme to implement all reasonably practicable improvements in the facility and its operations, including the documented safety case, to ensure that risks to the public and workers are ALARP. The licensee is required to develop and then execute this programme in a timely manner. The intent should be to implement all improvements a year after the licensee's review has been submitted to ONR. However, in cases where this is not reasonably practicable, the improvements must be completed in a timely manner within a three year period after the submission to ONR.

Reasonably practicable is a well-established principle within UK law. An important discovery in the interpretation of So Far As Is Reasonably (SFAIRP) occurred in the UK civil case Edwards v. The National Coal Board (1949). This case established that SFAIRP is the description of the computation which must be made in which the quantum of risk is placed in one scale and the sacrifice (whether in money, time or trouble) involved in the measures necessary to avert the risk is placed in the other. Only if it can be shown that there is a gross disproportion between them, the risk being insignificant in relation to the sacrifice, can the person upon whom a duty is laid demonstrate that they had taken all reasonably practicable steps (Reference 18). The nuclear industry commonly uses the terms As Low As Reasonably Practicable (ALARP) or As Low As Reasonably Achievable (ALARA), both of which are synonymous to SFAIRP and subject to the same test.

Guidance has developed over time which describes what is considered SFAIRP in particular circumstances. This guidance is known as Relevant Good Practice (RGP). In line with ONR's enforcement policy, ONR expect RGP to be followed (Reference 15). RGP includes ACOPs, SAPs, TAGs, TIGs; and publications of the British Standards Institute (BSI), IAEA, and WENRA (including References 16 and 17). Reference 17 provides specific guidance on timely implementation of ALARP improvements. Another important source of relevant good practice in the nuclear industry is what other facilities have done. As international standards are updated and technical improvements on sites are implemented, the expectations associated with RGP increase over time.

Question 3.7: *What is your national approach regarding the continuous improvement of nuclear safety (periodic safety reviews, evolution of safety objectives, operating experience feedback, benchmarks, peer review missions, etc.)?*

The Health and Safety at Work Act 1974 defines in section 2 that it shall be the duty of every employer to ensure, so far as is reasonably practicable, the health, safety and welfare at work of all employees. Section 3 defines that it shall be the duty of every employer to conduct undertakings in such a way as to ensure, so far as is reasonably practicable, that members of the public are not exposed to risks to their health and safety. These two sections are the primary legal basis for reducing risks ALARP in the UK nuclear industry.

ONR assessment of ALARP is conducted against Relevant Good Practice (RGP). IAEA Safety Standards, WENRA Reference Levels for existing reactors, and WENRA Safety Objectives for new reactors are defined as UK RGP in the ALARP TAG (Reference 5). In this way the UK regulatory framework automatically recognises new IAEA Safety Standards, WENRA Reference Levels and WENRA Safety Objectives on publication. The UK engages extensively with international forums in order to ensure that IAEA Safety Standards, WENRA Reference Levels and WENRA Safety Objectives are suitable for the UK nuclear industry in advance of publication. This ensures that the UK regulatory framework is always at the forefront of modern nuclear safety standards.

ONR subject their principles and guidance to periodic review. The ONR SAPs are reviewed against internationally endorsed standards every five years, and the TAGs and TIGs are similarly reviewed every three years. The ONR SAPs identify in paragraph 36 that the principle of continuous improvement is central to achieving sustained high standards of nuclear safety.

License condition 15 (periodic review) states that the licensee shall make and implement adequate arrangements for the periodic and systematic review and reassessment of safety cases. What is accepted as relevant good practice is subject to change over time. This is due to technological innovation and improved knowledge of the hazard following operational experience. For existing facilities undergoing Periodic Safety Review (PSR), the facility should be compared with the benchmark of modern standards. When considering compliance and the reasonable practicability of improvements, the dutyholder should take account of aspects such as the age of the facility, its future lifetime, future operations and the degree and importance of any shortfall. The dutyholder must implement measures to the point where the sacrifice of any additional measures (in terms of money, time or trouble) would be grossly disproportionate to the further risk reduction that would be achieved (the safety benefit). Licensee Periodic Safety Reviews are assessed by the national regulator, ONR.

4 Design and operational aspects of the defence-in-depth

Question 4.1: *Please describe how you updated your safety approach and/or its technical application, notably after WENRA published its Safety Objectives for new reactors and IAEA published SSR-2/1 (Rev. 1). More specifically, emphasize how you implemented the expectations for an enhanced defence-in-depth and a reinforced independence between levels 3 and 4 of DiD, in compliance with Article 8b of the Nuclear Safety Directive. You may illustrate with elements from your National Action Plan, or with updates that have been carried out before the European stress tests.*

The Severe Accident Analysis (SAA) Technical Assessment Guide (TAG) was updated in September 2017 (Reference 4). The TAG highlights the importance of multiple independent barriers to fault progression in relation to severe accidents and the defence in depth principle (SAP EKP.3). EKP.3 states that nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression. The TAG identifies that SAA is focussed on Levels 4 and 5 of the defence in depth hierarchy and Level 3 safety measures should normally be assumed to be unavailable. The SAA should then determine the additional safety measures required to regain control of the accident.

ONR updated its Safety Assessment Principles in 2014 in order to incorporate learning from the 2011 accident at Fukushima. During this revision, the SAPs were benchmarked against IAEA and WENRA RGP and any necessary amendments incorporated.

Question 4.2: *What are the most noticeable safety improvements at your existing plants that have been carried out by the applicants/licensees in your country regarding the expectation of continuous improvement of safety? You may illustrate with the safety improvements associated with regulatory updates, whether they have preceded or followed the publication of WENRA safety objectives for new reactors and IAEA SSR-2/1 (Rev. 1).*

Key safety improvements are detailed in ‘Japanese Earthquake and Tsunami: Update on UK National Action Plan’, December 2017 (Reference 19). Notable examples include: Passive Autocatalytic Re-combiners; Continuous Emergency Monitoring System (CEMS); upgrading the resistance of buildings to natural hazards; portable generators and pumps for diverse cooling; Sizewell B Emergency Response Centre; depots of Deployable Back-Up Equipment (DBUE) with a fleet of off-road vehicles; new diesel generators with enhanced seismic qualification and flooding protection.

Additional safety improvements are highlighted in the UK report to the Convention on Nuclear Safety (paragraph 6.28, Reference 22) and include: the provision of super articulated control rods at Hinkley Point B and Hunterston B; and replacement of nitrogen plant and reactor hold-down system at Hinkley Point B.

5 Probabilistic Safety Assessments

Note*: Considering the current actual practices, questions 5.1 to 5.5 are dedicated to nuclear power plant and 5.6 to research reactors.*
Question 5.1: *Do you require the following levels of PSA at any point in your licensing process (whether it be for new or existing reactors)? If so, please precise at which licensing stages, along with the end points (frequencies, doses, etc.) you require from this level of PSA:*

- *PSA level 1:*
- *PSA level 2:*
- *PSA level 3:*

Is the same level of detail (scope, reliability data, uncertainties, sensitivity studies…) required / provided at the different stages (licensing, operation…)?

ONR will assess the Probabilistic Safety Assessment at an early stage of Generic Design Assessment (GDA) which is a prequel to the formal licensing stages. PSA levels are considered to be: Level 1 Core / Fuel Damage Frequency, Level 2 Source Term, Level 3 Off-Site Dose. Level 1 PSA is expected for Stage 2 GDA. A generic Level 3 PSA is expected for Stage 4 GDA. Site specific details are subsequently added for site licensing purposes. Numerical targets 5 to 9 in the ONR SAPs pertain to PSA. The PSA TAG (Reference 20) provides a table of assessment expectations for the various levels of PSA.

Question 5.2: *Do you have requirements / expectations regarding the scope of PSAs? For example, are external events included? Are PSA required / expected for all plant operating states (normal operation, shutdown…)?*

SAP FA.12 regards the scope and extent of PSA. FA.12 states that PSA should cover all significant sources of radioactivity, all permitted operating states and all relevant initiating faults. External events are included in the PSA. The identification of initiating faults should consider the potential for combinations of hazards.

Question 5.3: *Do you require PSAs covering both reactor and spent fuel storage? Do you require PSAs for spent fuel storage as detailed as for the rest of the plant?*

ONR require the PSA to cover all significant sources of radioactivity (SAP FA.12) including the spent fuel pool. The Fuel Route PSA uses the criteria of fuel damage rather than core damage. The Fuel Route PSA is expected to include initiating events covering all disturbances that require mitigation to prevent fuel damage and those that lead directly to fuel damage. (Reference 20)

Question 5.4: *More generally, do you expect PSAs covering multiple installations at a same site?*

In relation to PSA, paragraph 650 of the ONR SAPs states: at multi-facility sites, the analysis should consider the potential for specific initiating faults giving rise to simultaneous impacts on several facilities or for faults in one facility to impact another facility.

Question 5.5: *Are PSAs used to define or justify the technical specifications of a reactor? Are they used for other types of application? If so, please describe your national approach.*

ONR SAP FA.14 states: PSA should be used to inform the design process and help ensure the safe operation of the site and its facilities.

The technical specification of a reactor will be assessed against ONR numerical targets 1 to 9 (Reference 1). Numerical targets 5 to 9 relate to PSA output.

Question 5.6: *When do you require PSAs to be updated to take into account new safety improvements? Please describe your national approach.*

The ONR PSA TAG (Reference 20) identifies that PSA studies performed to support any safety submission, including the justification of any modification to plant or operation, should be comprehensive, technically sound and properly documented. ONR inspectors evaluate dutyholders against these expectations during the 10 yearly PSR and following significant modifications. Inspectors also maintain oversight of progress against any improvement programmes necessary to address regulatory issues.

Question 5.7: *Are PSAs required / developed for new and/or existing research reactors? If so:*

- *What levels of PSA are performed?*
- *What is the scope of PSAs for research reactors (internal events, external events, fire, operating states to consider…)?*
- *How are PSAs used in the safety case (whether it be for the application of a new reactor, a licence modification/renewal of an existing reactor, or the implementation of a safety improvement)?*
- *Do these requirements / developments correspond to a regulatory requirement associated to a threshold (for example thermal power)?*

The ONR SAPs are technology neutral and apply to any type of nuclear facility on a nuclear licensed site. The ONR SAPs recognise that research reactors have significant differences from power reactors. Research reactors need to be considered on a case by case basis. The inspector determines the appropriate principles to apply to a particular situation. This paragraph applies equally to all subsequent questions regarding research reactors.

The need for PSA is defined in FA.10: suitable and sufficient PSA should be performed as part of the fault analysis and design development analysis. ONR are required by the Regulators' Code to be targeted with respect to risk, and would take a proportionate approach if the accident consequences were low.

6 Design extension conditions without fuel melt

Note 1: *For all the questions 6.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.*
Note 2: *The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer.*

Question 6.1*: At what level of DiD do you associate DEC without fuel melt in your regulation and/or in practice? Do you make a difference between DEC without fuel melt and "accidents within the design basis" as mentioned in Article 8b? If so, please explain.*

Internationally, the requirements for DEC and practical elimination remain in the development phase, with both WENRA and the IAEA continuing to draft further guidance and clarification. The ONR are active in influencing and adopting this guidance. Recent examples of ONR assessments in these areas are published at http://www.onr.org.uk/new-reactors/ Generic Design Assessment of new nuclear power stations (Reference 12).

DEC without fuel melt relates to Defence in Depth (DiD) Level 4. Events in this category are typically analysed on a best estimate basis. In contrast, for Design Basis Analysis (DBA), the absence of fuel melt is demonstrated on a conservative basis (DiD Level 3).

Question 6.2: *What is the methodology used by the applicant to come up with the list of DECs without fuel melt? What is the methodology you use to check and approve this list? Is the list revised within the scope of PSR?*

Safety Assessment Principles FA.1 to FA.3 identify how fault analysis events should be identified (Reference 1). SAP FA.1 states: fault analysis should be carried out comprising suitable and sufficient design basis analysis, PSA and severe accident analysis to demonstrate that risks are ALARP. SAP FA.2 states: fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement. SAP FA.3 states: fault sequences should be developed from the initiating faults and their potential consequences analysed.

The PSA TAG identifies that the fault schedule should list all the identifiable initiating faults and hazards within the scope of the PSA which could lead directly or in combination with other failures to a release of radioactive material (Reference 20). DEC events without fuel melt are contained within the PSA.

Question 6.3: *Does the list of DECs without fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.*

ONR require the fault analysis to cover all significant sources of radioactivity (SAP FA.12) including the spent fuel pool. An example from the UK EPR for DEC A is the loss of the two main trains of the Spent Fuel Pool Cooling System during core refuelling (Reference 7).

Question 6.4: *What are your requirements on the provisions to limit the consequences of DECs without fuel melt? In particular:*

- *What are the rules to analyse DEC without fuel melt?*
- *Do you require provisions dedicated to the limitation of the consequences of DECs without fuel melt?*
- *Do you require provisions associated with DECs without fuel melt to be independent from provisions associated with other plant states?*

The PSA TAG (Reference 20) identifies that best estimate methods and data should be used for the transient analyses, accident progression analyses, source term analyses, and radiological analysis that support the PSA. Where no credible best estimate is possible, reasonably conservative assumptions should be made and the sensitivity of the risk to these assumptions should be established.

The consequences of the fault analysis are assessed against the numerical targets. Assessment against numerical target 4 ensures adequate levels of independent provision for the basis of the design. DEC events without fuel melt are addressed in the PSA. SAP FA.14 identifies that PSA should be used to inform the design process and help ensure the safe operation of the site and its facilities.

Question 6.5: *What is the safety classification of the systems, structures and components for limiting the consequences of DECs without fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to DEC without fuel melt conditions, consideration of hazards etc.)?*

Reference 21 is the ONR TAG for the categorisation of safety functions and the classification of Structures, Systems and Components (SSC).

Safety functions are categorised based on their significance with regard to safety (SAP ECS.1). The categorisation is determined from the consideration of the unmitigated radiological consequences of failing to deliver the safety function; societal risk of a severe accident; longer term risks associated with accident recovery and remediation; the role and position of the safety function in the hierarchy of defence in depth; the influence to consequences of a partial delivery of the safety function; and the significance of the safety function in achieving a stable, safe state.

Structures, Systems and Components are classified as follows: Class 1 - any SSC that forms a principal means of fulfilling a Category A safety function; Class 2 - any SSC that makes a significant contribution to fulfilling a Category A safety function, or forms a principal means of ensuring a Category B safety function; Class 3 - any other SSC contributing to a categorised safety function. The class of the SSC determines the required reliability. For either failure frequency per year or failure per demand as appropriate, the following reliability expectations are defined: Class 3 10^{-1} to 10^{-2}, Class 2 10^{-2} to 10^{-3}, Class 1 10^{-3} to 10^{-5}.

The provision of properly defined safety functions and SSCs are fundamental to the development of robust safety cases and well-engineered protective measures for all of the possible states in the lifecycle of a facility. This includes:

1. normal operational states including power generation, standby states, shutdown states, outage or maintenance states;
2. other lifecycle states including construction, commissioning, post-operational;
3. clean-out, decommissioning;
4. operational abnormalities or fault states within the design basis;
5. states which may have arisen because of a beyond design basis event, malicious act or the escalation of a design basis fault;
6. situations in which significant relocations or releases of radioactive material have occurred and need to be managed.

Item 5 of the preceding list is broadly equivalent to DEC A, and item 6 is broadly equivalent to DEC B.

Question 6.6: *If a new reactor is currently being built in your country, did the list of DECs without fuel melt and provisions to limit their consequences change compared to existing reactors?*

The list of DECs without fuel melt for the UK's new EPR is described in Reference 6 pre-construction safety report, sub-chapter 16.1, risk reduction analysis (RRC-A). In the UK EPR defence-in-depth approach, the risk reduction category RRC-A is introduced to complement the deterministic design basis analysis by considering a set of Design Extension Conditions (DECs) due to multiple failure events. The analysis of DECs is performed using both deterministic and probabilistic considerations and leads to the identification of additional safety features (or 'RRC-A features'), which make it possible to prevent the occurrence of a severe accident in these complex situations. Additional RRC-A features are classified consistent with EPR safety classification principles.

Sizewell B first generated power in 1995 and as such was designed to an earlier set of design requirements. That being said, a beyond design basis and incredible initiating faults analysis was conducted for Sizewell B. The delta between modern standards and Sizewell B has been assessed through PSR and post Fukushima stress testing. There have been a number of significant, reasonably practicable improvements that have been carried out at the Sizewell site in a timely manner following the Fukushima accident. These include the installation of seismically qualified passive autocatalytic re-combiners; the replacement of battery charging diesel generators to improve the seismic and flooding resilience of the site; a dedicated off-site emergency response facility; and provision of electrical and fluid system tie-in points to enable connection of portable equipment to the plant in a beyond design basis event (Reference 22).

7 Design extension conditions with fuel melt

Note 1: *For all the questions 7.* in this chapter, please indicate whether your approach is different for existing and new reactors. If it is, please indicate clearly the differences.*
Note 2: *The questions do not differentiate power reactors from research reactors. Nevertheless, considering that DEC is mainly mentioned in IAEA SSR-3 in a chapter dedicated to design and considering that most of research reactors are existing ones, if you believe a question is not relevant for research reactors, please precise it in your answer..*

Question 7.1: *Do you make a distinction between DEC with fuel melt and "severe conditions" as mentioned in Article 8b in your regulation and/or in practice? If so, please explain.*

Internationally, the requirements for DEC and practical elimination remain in the development phase, with both WENRA and the IAEA continuing to draft further guidance and clarification. The ONR are active in influencing and adopting this guidance. Recent examples of ONR assessments in these areas are published at http://www.onr.org.uk/new-reactors/ Generic Design Assessment of new nuclear power stations (Reference 12).

Article 8b requires severe conditions to be controlled, including prevention of accident progression and mitigation of the consequences of severe accidents.

In the ONR Defence in Depth, Level 4 relates to the control of severe plant conditions in which the design basis may be exceeded, including protecting against further fault escalation and mitigation of the consequences of severe accidents. Defence in Depth Level 5 refers to the mitigation of radiological consequences of significant releases of radioactive material.

In terms of safety case analysis in the UK, DEC with fuel melt is analysed in the SAA including control of severe conditions and mitigation of the consequences of severe accidents.

In terms of emergency preparedness, Symptom Based Emergency Response Guidelines (SBERGs) provide advice for recovery from a situation beyond the design basis of the plant, such that conditions are controlled to prevent the onset of a degraded core. If core degradation does occur then the management of the accident escalates to the Severe Accident Guidelines (SAGs) to mitigate uncontrolled off-site radiological releases.

Question 7.2: *What is the general approach (deterministic and/or probabilistic) used by the applicant or licensee to ensure that a relevant scope of DECs with fuel melt conditions has been considered in its design? Is the scope revised during PSR?*

Safety Assessment Principles FA.1 to FA.3 identify how fault analysis events should be identified (Reference 1). SAP FA.1 states: fault analysis should be carried out comprising suitable and sufficient design basis analysis, PSA and severe accident analysis to demonstrate that risks are ALARP. SAP FA.2 states: fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement. SAP FA.3 states: fault sequences should be developed from the initiating faults and their potential consequences analysed.

SAP FA.15 defines the scope of the severe accident analysis: fault states, scenarios and sequences beyond the design basis that have the potential to lead to a severe accident should be analysed. The SAA should, through a systematic approach, analyse beyond design basis states and scenarios arising from the following circumstances: high consequence events of very low frequency for which the design safety measures may be ineffective; and design basis events where, conservatively, the safety provisions are assumed to fail. SAA addresses DEC events with fuel melt. SAP FA.15 applies equally to an initial safety case assessment and a PSR safety case assessment, however the expectations of RGP may have changed over time.

Question 7.3: *Does the list of DECs with fuel melt include sequences that might happen on the spent fuel pool? If so, please provide examples of such sequences.*

ONR require the fault analysis to cover all significant sources of radioactivity (SAP FA.12) including the spent fuel pool. The UK EPR spent fuel pool DEC does not result in fuel melt (Reference 7).

Question 7.4:

What are your requirements on the provisions to limit the consequences of DECs with fuel melt? In particular:

- *What are the rules to analyse DEC with fuel melt?*
- *Do you require provisions dedicated to the limitation of the consequences of DECs with fuel melt?*
- *Do you require provisions associated with DECs with fuel melt to be independent from provisions associated with other plant states?*

The ONR SAPs identify that for SAA a best estimate approach should normally be followed. However, where uncertainties are such that a realistic analysis cannot be performed with confidence, a conservative or bounding case approach should be adopted to avoid optimistic conclusions being drawn. Where a best estimate approach is not followed, the extent to which the analysis could nevertheless be used to inform emergency response activities (e.g. in regard to the expected timings of escalations in the accident sequence) should be considered.

The consequences of the fault analysis are assessed against the numerical targets. Assessment against numerical target 4 ensures adequate levels of independent provision for the basis of the design. DEC events with fuel melt are addressed in the SAA. SAP FA.16 identifies that Severe Accident Analysis should be used in the consideration of further risk-reducing measures.

Question 7.5: *What is the safety classification of the systems, structures and components for limiting the consequences of DECs with fuel melt? What are the associated requirements (redundancy, electrical supply, qualification to severe accident conditions, consideration of hazards etc.)?*

Please see response to question 6.5. Item 6 in the list relates to DEC with fuel melt.

Question 7.6: *If a new reactor is currently being built in your country, how did the list of DECs with fuel melt and provisions to limit their consequences change compared to existing reactors?*

The list of DECs with fuel melt for the UK's new EPR is described in Reference 6 pre-construction safety report, sub-chapter 16.2, severe accident analysis (RRC-B). The UK EPR's safety concept ensures that highly energetic accident situations with core melt leading to large early release are practically eliminated (noting that the Hinkley Point C safety case continues to be refined and remains under on-going assessment by the national regulator ONR). Low pressure core melt sequences (RRC-B) only necessitate protective measures for the public which are limited in area and time. The radiological objectives associated with RRC-B accidents are expressed on the basis that only very limited countermeasures should be necessary in such situations i.e.

- limited sheltering duration for the public,
- no need for emergency evacuation beyond the immediate vicinity of the plant,
- no permanent relocation,
- no long-term restrictions on the consumption of foodstuffs.

Sizewell B first generated power in 1995 and as such was designed to an earlier set of design requirements. That being said, a beyond design basis and incredible initiating faults analysis was conducted for Sizewell B. The delta between modern standards and Sizewell B has been assessed through PSR and post Fukushima stress testing. There have been a number of significant, reasonably practicable improvements that have been carried out at the Sizewell site in a timely manner following the Fukushima accident. These improvements are described in the response to question 6.6.

8 Design of the containment

Question 8.1: *What are your technical expectations / requirements (and their associated parameters) regarding the confinement function in accident conditions (independence of barriers, leak tightness, integrity, etc.)?*

The ONR SAPs state that containment and associated ventilation systems should confine the radioactive material within the facility and prevent its leakage or escape to the environment in normal operation and fault conditions, except in accordance with authorised discharge conditions, or as part of a planned transfer to another facility.

ONR SAP ECV.3 regards means of confinement. ECV.3 states that the primary means of confining radioactive materials should be through the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components. The safety case should minimise the size and number of service penetrations in the containment boundary, which should be adequately sealed to reduce the possibility of radioactive material escaping via routes installed for other purposes.

Containment and associated nuclear ventilation systems will normally form part of systems important to safety and so the safety assessment principles applicable to engineering, safety systems and essential services will apply in the usual manner. Recent examples for ONR assessment of confinement function can be found at: http://www.onr.org.uk/new-reactors/ Generic Design Assessment of new nuclear power stations (Reference 12) and including the

UK ABWR) Step 4 reports on fault studies (Reference 7), PSA, severe accidents (Reference 8), civil engineering (Reference 13) and structural integrity.

Question 8.2: *How are design pressure and temperature defined for the containment design (from accident studies, from an envelope load/bounding case scenario…)?*

The ONR SAA TAG (Reference 4) identifies that pressure and temperature in the containment shall be managed. Equipment qualification procedures should, where appropriate, address severe accident conditions. Where equipment needs to perform safety functions in severe accident conditions, it should be qualified to do so. This requires consideration of post-accident conditions (e.g. high pressures and temperatures). The TAG further states that SAA should be used to predict the maximum containment pressure to protect the containment against overpressure.

Question 8.3: *What is the approach for containment penetrations? Is their leak tightness regularly tested?*

ONR SAP ECV.1 regards the prevention of leakage from containment. ECV.1 states that radioactive material should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented. ONR SAP ECV.7 regards leakage monitoring. ECV.7 states that appropriate sampling and monitoring systems should be provided outside the containment to detect, locate, quantify and monitor for leakages or escapes of radioactive material from the containment boundaries. Provision should be made for testing the leakage monitoring systems at suitable intervals to confirm continuing system performance. Such testing may include, for example, monitoring depressions, airflows, inerting gas concentrations, filter performance or valve response times and consider EMT.6. ONR SAP EMT.6 regards reliability claims. EMT.6 states that provision should be made for testing, maintaining, monitoring and inspecting structures, systems and components (including portable equipment) in service or at intervals throughout their life, commensurate with the reliability required of each item.

License Condition (LC) 28 regards examination, inspection, maintenance and testing. LC28 states that the licensee shall make and implement adequate arrangements for the regular and systematic examination, inspection, maintenance and testing of all plant which may affect safety. Licence Condition 34 regards leakage and escape of radioactive material and radioactive waste. LC 34 states that the licensee shall ensure, so far as is reasonably practicable, that radioactive material and radioactive waste on the site is at all times adequately controlled or contained so that it cannot leak or otherwise escape from such control or containment.

9 Practical elimination

Question 9.1: *Do you have a definition of practical elimination in your country? If so, please provide it, if not, please provide your interpretation of practical elimination.*

Internationally, the requirements for DEC and practical elimination remain in the development phase, with both WENRA and the IAEA continuing to draft further guidance and clarification. The ONR are active in influencing and adopting this guidance. Recent examples of ONR assessments in these areas are published at http://www.onr.org.uk/new-reactors/ Generic Design Assessment of new nuclear power stations (Reference 12).

The ONR SAPs state in paragraph 611 that in line with wider international guidance, the SAA should form part of a demonstration that potential severe accident states have been ‘practically eliminated’. To demonstrate practical elimination, the safety case should show either that it is physically impossible for the accident state to occur or that design provisions mean that the state can be considered to be extremely unlikely with a high degree of confidence.

Question 9.2*: Do you have requirements related to practical elimination?*

The ONR SAA TAG highlights that special attention should be paid to events initiated by gross failures of plant and equipment that rely on claims of high structural integrity in the safety case, particularly those relating to maintaining pressure / containment boundaries e.g. where the licensee has made such a claim or where the fault has been 'practically eliminated'. Here the analysis will need to go beyond 10^{-7} per year so that cliff edge effects from these failures can be suitably addressed.

Question 9.3: *Is the concept of practical elimination used (if so, please provide examples and describe differences in its application between different types of reactors, if any):*

- *For existing power reactors?*
- *For new power reactors?*
- *For existing research reactors?*
- *For new research reactors?*

The UK EPR uses the concept of practical elimination with respect to the following scenarios (Reference 6):

- High pressure core melt accident and Direct Containment Heating (DCH)
- Steam explosions leading to failure of the containment
- Hydrogen combustion processes endangering containment integrity
- Rapid reactivity insertion
- Containment bypass (e.g. Loss of Coolant Accident leading to containment bypass via the Medium Head Safety Injection system)
- Fuel damage in the spent fuel pool

The design of Sizewell B predates the emergence of the practical elimination terminology; however the concept of 'incredibility of failure' was applied to key components (e.g. gross failure of the RPV). Incredibility of failure serves a similar function in the safety case to practical elimination i.e. if the consequences of a specific failure are unacceptable then stringent requirements for design, manufacture and construction along with appropriate preventative measures must be applied to ensure that the failure can be discounted.

Question 9.3: *Please describe safety improvements (for operating reactors) or safety features (for new reactors) implemented or to be implemented to practically eliminate sequences leading to early or large releases.*

With respect to the UK EPR (Reference 6), a design objective is to convert high pressure core melt sequences into low pressure sequences with high reliability so that high pressure core melt situations can be practically eliminated. This objective implies the need to limit the pressure in the reactor coolant system by the time of Reactor Pressure Vessel (RPV) rupture. This objective is ensured by installing two dedicated Severe Accident Depressurisation Valves (SADVs) within the Primary Depressurisation System (PDS). PDS severe accident analysis has shown that even for bounding core melt scenarios the pressure is adequately low at the time of RPV failure. Thus the risk of high pressure core melt scenarios can be considered as practically eliminated, as it is physically impossible for the accident state of high pressure core melt to occur.

For the practical elimination of fuel damage in the spent fuel pool for the UK EPR, the designer must make provisions to avoid the total loss of the cooling system of the spent fuel pool and to avoid pool draining. The spent fuel pool water cooling system removes residual heat from the spent fuel assemblies stored in the pool. This system consists of two identical principal trains, each equipped with two pumps and a heat exchanger; plus a third train equipped with one pump and one heat exchanger diverse from the principal trains. The third train is completely independent with respect to electrical supply and cooling system. Following a Loss of Offsite Power (LOOP), the third train can be supplied by either the Emergency

Diesel Generators (EDGs) or the Station Blackout (SBO) Diesel Generators. Automatic isolation of the lines connected to the bottom of the pool has been implemented to prevent accidental draining of the spent fuel pool. In addition to the deterministic evaluation of the design, a probabilistic analysis of loss of cooling and accidental draining scenarios has been conducted to demonstrate that the risk of fuel assembly damage is extremely unlikely with a high degree of confidence.

Question 9.4: *What are your requirements / expectations regarding the demonstration of the practical elimination of accidental sequences? What principles do you require or expect the applicant / licensee to apply in order to demonstrate the practical elimination of a situation? Is it based on a case-by-case approach?*

Each instance where practical elimination is claimed should be assessed separately, taking into account relevant uncertainties, particularly those due to limited knowledge of extreme physical phenomena (e.g. the behaviour of molten reactor cores). Moreover, an accident state should not be considered to have been practically eliminated simply on the basis of meeting probabilistic criteria. Instead, any claims made on SSCs in relation to practical elimination need to be substantiated appropriately.

Question 9.5: *Is the practical elimination concept applicable also for fuel storage pools?*

The practical elimination concept is applicable for fuel storage pools as described in the response to question 9.3.

10 Periodic safety review

Question 10.1: *Is the scope of the Periodic Safety Review (PSR) defined in your regulatory framework? If not, what are your expectations and how is it defined?*

ONR's regulatory expectations for Periodic safety review are defined in Reference 14. In line with IAEA guidance we expect PSR to be a comprehensive assessment of:

- The extent to which the nuclear facility and the safety case conform to modern standards and good practices.
- The extent to which the safety documentation, including the licensing basis, remains valid.
- The adequacy of the arrangements in place to maintain safety until the next PSR or the end of life.
- Safety improvements to be implemented to resolve safety issues.

The review should not be limited to design basis events, but should also consider the resilience of the plant, staff and processes to events beyond the design basis.

Question 10.2: *What are the safety objectives for a reactor when performing the periodic safety reviews? How are they defined? Do they take the following safety factors according to WENRA Safety Reference Level P2.2 into account:*

- *Plant design*
- *Actual condition of structures, systems and components (SSCs) important to safety*
- *Equipment qualification*
- *Ageing*
- *Deterministic safety analysis*
- *Probabilistic safety assessment*
- *Hazard analysis*
- *Safety performance*
- *Use of experience from other plants and research findings*

- *Organization, management system and safety culture*
- *Procedures*
- *Human factors*
- *Emergency planning*
- *Radiological impact on the environment*

A PSR includes a comprehensive assessment of the facility's condition, operating experience, safety case, management arrangements and culture, looking forward at least the next ten years and normally to the end of life. The review is carried out at appropriate intervals through the different lifecycle phases of the facility, usually every ten years starting at the commencement of active commissioning. (ONR SAPs, Reference 1)

The ONR PSR TAG (Reference 14) identifies that ONR regard the WENRA Safety Reference Levels (SRL) as relevant good practice. ONR therefore expect the safety factors identified in the SRLs to be addressed in the PSR. PSRs are directly addressed in Issue P of WENRA's report on reactor SRL (Reference 23). Issue I on ageing management also requires the PSR to confirm whether ageing and wear-out mechanisms have been correctly taken into account. Reference level P2.2 is directly reproduced in the ONR PSR TAG including the list of safety factors to take into account (as a minimum).

Question 10.3: *How do you ensure that PSR takes into account the most recent developments in international standards, notably to identify potential safety improvements?*

The ONR SAPs identify that the principle of continuous improvement is central to achieving sustained high standards of nuclear safety. The legal requirements for risk reduction SFAIRP, and for PSR as required by Licence Condition 15, underpin this principle. Application of this principle ensures that, no matter how high the standards of nuclear design and subsequent operations are, improvements should always be sought. Seeking and applying lessons learned from events, new knowledge and experience, both nationally and internationally, must be a fundamental feature of the safety culture of the nuclear industry.

As discussed in the response to question 3.7, the UK regulatory framework automatically recognises new IAEA Safety Standards, WENRA Reference Levels and WENRA Safety Objectives on publication. ONR is active within IAEA and WENRA, sitting on both approval and drafting committees.

11 Safety culture and operational experience

Question 11.1: *Please describe your national approach to encourage the development of a robust safety culture as asked in paragraph 2 of Article 8b?*

ONR considers that Leadership and Management for Safety (LMfS) is a key determinant of safety culture and nuclear safety outcomes. ONR's expectations for LMfS are set out in ONR's Safety Assessment Principles (SAPs) for Nuclear Facilities (MS.1 to MS.4); these set the foundation for the effective delivery of nuclear safety, including the development and maintenance of a positive safety culture. Once per year ONR programmes undertake evaluations of licensee's performance against each of the four LMfS SAPs facilitated by specialist LMfS inspectors (Reference 24). These reviews are intended to inform development and resourcing of ONR's future intervention plans. Where appropriate, specific themed inspections are undertaken and corrective action sought from the licensee.

ONR has recently published its expectations in relation to the development and maintenance of an embedded security culture within the dutyholder organisation (Reference 25). The guidance document (Reference 26) provides assistance to inspectors in making regulatory judgements and decisions in relation to organisational security culture. This guidance has been developed to be consistent with the goals of the IAEA nuclear security programme.

Question 11.2: *Paragraph 2 of Article 8b related to safety culture requires having an appropriate feedback from operating experience. Please describe the requirements/ arrangements in place to register, evaluate and document internal and external operating experience.*

The arrangements to register, evaluate and document internal operating experience is described in ONR Guidance Document Reference 27. This document identifies that Operating Experience (OPEX) is a valuable source for learning about and improving the safety and security of nuclear facilities and activities. As part of any OPEX process it is essential to collect information from incidents and events occurring in nuclear facilities. Each incident should be raised on a separate INF1 (Incident Notification Form, Reference 28) and sent to ONR as a discreet incident. Where several incidents have occurred and are seen to be inextricably linked, a single INF1 may be raised, which should include clarification on the connection of the incidents notified. The approach to external lessons learned is described in the response to question 10.3, both in terms of continuous improvement safety culture and PSR.

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PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

FINAL STUDY

VOLUME 4

REPORT ON SELECTED EXAMPLES OF SAFETY UPGRADES PERFORMED IN EXISTING NUCLEAR REACTORS

August 2020

EC Contract No. ENER/17/NUCL/S12.769200

Contents

1 BACKGROUND

As stated in the inception report [1], the objective of Task 3 is to perform an analysis of the practices and approaches used to implement safety upgrades in existing nuclear reactors in selected Member States.

The selection of the case studies in this task offers the widest possible range in terms of nuclear strategies involved, namely prolonged operation (e.g. Slovenia), construction of new units (e.g. France, Finland, Hungary), phase-out (Germany), western LWR designs (Finland, France, Germany and Slovenia) and specific technologies (e.g., CANDU reactors in Romania, VVER reactors in Hungary). The situation in these selected countries was analysed and specific country reports ([2]-[7]) were issued.

The selection of countries aimed to address different boundary conditions as well as different technologies. It was nevertheless set up as a spot sample rather than a comprehensive overview and analysis of the practices in the European Union.

The first detailed study on safety upgrades was performed for Slovenia [2]. All the other reports have used a similar structure as proposed in the Slovenian report and are focusing on country-specific situations, as follows:

- France [3]: the study focused on the impact of PSRs on upgrades of different series of reactors. Further illustrations were given on the features to control or mitigate severe accidents.
- Finland [4]: the study has taken the new projects into account.
- Germany [5]: the main focus was on safety improvements enhancing preventive and mitigative accident management measures.
- Romania [6]: illustration of the implementation of plant safety upgrades (bottom-up and top-down approaches) for the Cernavoda NPP (CANDU-type reactors).
- Hungary [7]: the study outlines the legal framework for nuclear safety in Hungary, safety improvements at the Paks nuclear power plant, and efforts to implement Articles 8a-8c of Directive 2014/87/Euratom in practice. The report mainly takes into account the fact that safety upgrades have been continuously implemented and that the fleet of VVER reactors is limited and focuses on the possible distinctions in existing and new plants.

The competent authorities in all respective countries were consulted on the contents of the country reports. Taking into consideration the following extracts from the *Summary Note* *of Quadripartite Meeting – ENSREG WG1, EC, ETSON, WENRA RHWG (FANC offices, Brussels, 30th January 2019):*

> *"Concern expressed by WENRA and accepted by ETSON, that the ETSON study is not a comprehensive analysis of all nuclear countries and could result in a potentially subjective description of the overall situation across the EU, highlighting differences in approach.*
>
> *The point made was that if there are no clear criteria for decisions, clear consistent judgments would be difficult to make, especially across a range of TSOs reviewing different countries."*

and consistent with the tender specifications and the technical discussions of the first workshop [8], the country-specific reports were used as an input to this final report for Task 3, summarising the national practices when implementing safety upgrades in operating nuclear reactors. For this purpose, the country reports were reviewed by the consortium partners. The review work has focused on:

- identification of the technical rationales for safety upgrades in the selected countries;
- identification of similarities and differences in the national practices to implement safety upgrades.

The outcomes of this report were presented and discussed during the second workshop of this project, held on the European Commission premises on November 13 and 14, 2019.

The national reports in the annex 1 to 6 have been prepared solely on the views and knowledge of the national TSO. The presented opinions and conclusions in the national reports may differ from the views of national regulators.

2 SCOPE OF THE REPORT

The current report presents the results of the analysis of the practices and approaches used to implement safety upgrades in existing nuclear reactors in the following selected Member States: Finland, France, Germany, Hungary, Slovenia and Romania.

As such, the technical overall scope of the report covers the implementation of safety upgrades in existing nuclear reactors in selected countries.

According to the inception report [1] and the outcomes of the first workshop [8], this report focuses on:

- providing insights into the technical rationales for safety upgrades;
- identification of similarities and differences in the national practices to implement safety upgrades;
- identification of potential technical areas on which further discussions and an exchange of experience between Member States may be worthwhile.

The report is based on the evaluation of the information contained in the national reports, regarding the actual situation in the selected countries.

Regarding the similarities between the national practices as well as topics where further international cooperation may be beneficial, the consortium provides some suggestions for further discussions and potential future activities.

The similarities and differences discussed in this report have been chosen primarily to illustrate the similarities and diversities in the national approaches and to facilitate future discussion leading towards a profound understanding and potentially also to learning from each other. The intention of the authors was neither to define nor to promote any good practices.

3 ANALYSIS OF NATIONAL APPROACHES FOR SAFETY UPGRADES

3.1 SLOVENIA case study summary

The Slovenian report [2] documents the legal and technical implementation of plant safety upgrades in Slovenia after the implementation of Council Directive 2014/87/Euratom. The legislation and rules are in place, the regulator has confirmed the plan of safety upgrades at the Krško nuclear power plant, and the upgrades of the nuclear power plant are being commissioned. Currently, the safety upgrades are estimated to be completed beyond 2/3.

The report focuses on Design Extension Conditions (DECs), chosen as main topic for the report because they were conceived by the regulators to provide a reasonably practicable approach to avoid, prevent and/or mitigate the beyond-design accidents and their consequences, especially early and/or large radioactive releases. In addition, the Defence-in-Depth (DiD) principle and the 10-year periodic safety review had already been an integral part of the Slovenian nuclear legislation when Council Directive 2014/87/Euratom of 8 July 2014 was published.

In the report, the outline and analysis of the technical and legal implementation of the DEC concept is supplemented with the outlines of the Slovenian nuclear regulatory framework and the timeline of the major decisions leading to the present state of reactor safety upgrades in Slovenia.

The most important information is summarised below:

The Slovenian safety upgrade programme has been mostly driven by the licensee, who has been proposing the concepts, technical solutions and the schedules for the implementation.

The SNSA supported these activities by approvals of those concepts, technical solutions and the implementation schedules. The legal implementation of the related concepts and rules has been effected mostly in parallel and, in part, also after the approvals of the technical solutions proposed by the licensee.

A licensee-driven safety upgrade process is fully consistent with the primary responsibility for nuclear safety, and it can be seen as one of the best approaches available, especially if fully supported in real time by the regulators and TSOs. A challenge could be the timely mobilisation of sufficient resources at all relevant stakeholders (licensee, regulator, TSO, policy makers).

Periodic safety reviews (PSRs) seem to be an efficient way to identify the weak points in technology. Perhaps it would be useful to develop in the future reviews of a comparable level of detail for the rest of the man-technology-organization triad.

The DEC concept is considered by the Jožef Stefan Institute as a relatively complex concept, which brings rather complex changes to the relatively simple landscape of “design” and “beyond design”. There are different interpretations of DECs by IAEA and WENRA (e.g., “considerations of events more severe than design basis events” is expected for natural hazards according to WENRA SRLs and not in IAEA SSR-2/1). The introduction of DECs has brought a change to the realm of the regulations and an improved approach with regards to the former “beyond-design-basis approach”.

Complex changes require sufficient attention, time and resources, and therefore they are justified if they bring substantial added value. In depth analyses of safety benefits would be most beneficial before any changes in the legislation. Inadequate and/or delayed implementation could bring much fewer benefits than anticipated. The substitution of the “Design Extension Condition“ concept with the concept of the “Extended Design Bases” (for the discussion on the linguistic nuances in the ruling in Slovenian language, see [2]) could for example easily result in a routine updating of design bases, which could eventually prevent any facility from being operated for more than a decade or two.

An additional challenge could be to safely combine different safety philosophies, as for example the safety philosophy of the vendor's country’s regulations (the US NRC, which explicitly rejects DECs and focuses on the improvements of the “design” – “beyond design” regulatory framework) and the WENRA philosophy. The challenge might grow larger if the philosophies followed started focusing on the conditions with substantially different probabilities and/or consequences.

Legislation that is only available in the national language (as in the case of Slovenia) could present a potential challenge for international vendors, especially in cases with very specific and/or complex regulatory requirements and small number of facilities affected.

Regulatory-driven safety upgrades require substantial human resources for preparing well-defined and easy-to-understand regulation. Regulators in countries with fewer nuclear facilities could be affected more. The implementation of rather generic requirements in the national regulations through translations and adaptations might bring additional complexity and uncertainties. A possible solution may be the critical endorsement of international requirements that requires profound understanding of the basis of applied requirements.

A parallel implementation of technical solutions and regulatory requirements, and especially the definition of the regulatory requirements after the technical solution and their parameters have been selected and contracted, might increase the licensee’s risk of the commissioned solution not being licensed. If perceived as such by the licensee, the motivation of the licensee for a proactive and bottom-up approach to safety upgrades might be reduced.

The DEC concept should probably aim to expand beyond the technology part of the man-technology-organization triad, which controls the safety of complex technologies, including nuclear power.

3.2 FRANCE case study summary

The French report [3] highlights the features to control or mitigate severe accidents, notably safety upgrades that have been successfully implemented following PSRs in France, together with upgrades proposed in the French Action Plan following the Fukushima Daiichi accident. France is the country with the largest NPP fleet in Europe, including the construction of a new plant (EPR at Flamanville). As such, most of the safety upgrades are not specific to one NPP and have to be implemented fleetwide. There are three series of PWRs operated in France: 900 MWe (3-loops RCS, 2 safety-trains (containment heat removal system, high-pressure safety injection, low-pressure safety injection), single large dry containment with a steel liner), 1300 MWe and 1450 MWe (4-loops RCS, double large dry containment, 2 safety trains (containment heat removal system, medium-pressure safety injection, low-pressure safety injection)).

The report provides information on a part of those fleet-wide safety upgrades, including on-going activities associated with long-term operation (4th PSR).

PSRs of nuclear installations have been provided since the 1960s in a French decree, and ten-yearly PSRs have been performed for NPPs from the beginning of the nuclear programme. Nevertheless, this ten-yearly PSR was only required by law in 2006 with the adoption of the law on transparency and security in the nuclear field (TSN act, whose reference in France is “Loi n° 2006-686 du 13 juin 2006 relative à la transparence et à la sécurité en matière nucléaire”).

The current practice is to organise PSRs in 2 phases regarding the wide fleet of French PWRs and the existence of only few different plant series (900, 1300 and 1450MWe). The first phase is a generic “plant series approach” whose conclusions are applicable to all NPPs belonging to the corresponding series. The second phase takes into account specificities of each NPP (e.g. related to the site, the results of the compliance assessment, the ageing management programme).

For the next PSR of the 900MWe series, the licensee has presented a programme for Long Term Operation (LTO), with the aim to extend the operating lifetime of the current fleet beyond 40 years. Such a programme includes not only an ageing programme but also some important safety upgrades aiming to reduce the gap to the safety objectives of the generation III reactors, like the EPR.

The French Safety Authority ASN (supported by IRSN technical reviews) has confirmed the importance of the general objective to reduce the differences to the safety objectives defined for the EPR but also to achieve the implementation of all reinforcements decided after the Fukushima accident. One major objective is to obtain a significant reduction of the radioactive releases in case of an accident (with or without core melt).

In 2001, the French Safety Authority ASN requested the NPP operator to develop a “severe accident safety standard” containing the approach and objectives for the prevention and the mitigation of risks associated with severe accidents, the studies necessary to demonstrate compliance with the objectives, and the practical provisions and their design basis. It includes the severe accident conditions to be considered for the qualification of structures, systems and components. For each PSR, the “severe accident safety standard” is updated by the operator, taking into account the results of research programmes (computational codes, experiments …).

More recently, given the scale of natural phenomena observed in Japan and whatever the origin of the under-sizing of the Fukushima Daiichi plant against natural hazards, it was decided that:

- the robustness of the French facilities should be ensured for levels of external hazards exceeding those considered in the most recent safety framework,
- their robustness for some accidents that have not been retained so far should also be ensured (long-duration loss of sources that may affect several facilities simultaneously on one site).

There was a series of additional resolutions issued by the ASN in 2014, updated in 2017, with a list of requests regarding the prevention of core melt and, should it occur, mitigation of the consequences of severe accidents. Licensees were asked to implement a so-called “hardened safety core”, i.e. a set of human, organizational and equipment provisions that withstand rare and severe external hazards. It aims to limit the consequences of an accident, including severe accident situations. The approach is based on the DiD principle, strengthening each level.

The “Hardened Safety Core” should be able to manage accident situations of long duration, affecting several plants at the same site, considering induced events, before the arrival of off-site support organisations such as FARN. FARN is the Rapid Nuclear Action Force of the French utility EDF and was fully operational by the end of 2015. FARN can provide assistance to a damaged site by providing specialised teams to back up those of the plant concerned and mobile equipment to supply additional water and electricity. Several modifications were therefore made to the reactors to make it easier to connect equipment brought on site by FARN.

3.3 FINLAND case study summary

The Finnish legal framework for nuclear safety was briefly presented in the Finnish report [4], and its development since 2011 was described. The safety upgrades performed in the Finnish reactors since 2011 were also specified.

The principle of continuous safety improvements was adopted in Finland already in the 1970's when nuclear power plant operation was started. In 2008, this principle was included in the Nuclear Energy Act and the Government Decree on the Safety of Nuclear Power Plants, currently STUK Regulation Y/1/2018.

The wording of the Finnish classification of accidents differs from that of many other countries. Finland does not use the term *design basis accident*, but the term *postulated accident* has about the same meaning. DECs are accidents that have a smaller probability than postulated accidents, but the reactor must withstand them without fuel damage. The term *severe accident* is defined as an accident with fuel damage. In the DiD principle, both postulated accidents and DECs are on the third level of defence, while severe accidents are on the fourth (and fifth) level.

Finland's regulatory guides, called YVL Guides, are continuously re-evaluated for updating, considering operating experience, research, and advances in technology. The updated regulations apply as such to new reactors, but application to existing reactors is decided on a case-by-case basis. This practice follows the guiding principle of the Nuclear Energy Act: "The safety of nuclear energy use shall be maintained at as high a level as practically possible" and allow the implementation of reasonable safety improvements since modern technology allows a higher safety level for new reactors, but it would be unreasonable to require the same safety level for old reactors by back-fitting.

Finland's regulator STUK started a project of completely rewriting the YVL Guides in 2008, and the lessons learned from the Fukushima accident were taken into account in the new YVL Guides before their publication in 2013. Prior to the Fukushima accident, Finland already had strict requirements for safety-classified severe accident management systems with redundancy and independency from the safety systems designed for postulated accidents. Thus, major changes to the regulations were not needed for severe accident management. The new guides require that nuclear facilities have to withstand more severe natural phenomena and power failures.

In the stress tests that were performed after the Fukushima accident, hazards or deficiencies requiring immediate actions were not identified, but many opportunities for safety improvements were found. Safety improvements have been implemented, following the safety objectives and requirements of STUK Regulation Y/1/2018 and more detailed requirements in the YVL Guides.

The most significant plant modifications are related to a loss of electric power. In Olkiluoto 1&2, two new systems for pumping water into the reactor during a loss of AC power were constructed. The low-pressure system works with pumps that are independent from the plant electric systems, and the high-pressure system pumps operate with steam turbines, similarly to the HPCI (high pressure coolant injection) and RCIC (reactor core isolation cooling) systems at the Fukushima Daiichi NPP. New mobile generators and pumps have been acquired, and they can be used if all AC power is lost. New cooling towers were constructed in Loviisa in order to cope with a loss of the ultimate heat sink. For Olkiluoto 3, no major options for safety improvements were found in the stress tests. Only the possibility to pump cooling water to the steam generators and to the spent fuel pool with mobile pumps has been implemented.

A potential challenge related to the construction of new safety systems is the increased complexity. It becomes more challenging for the operators to understand the behaviour of all the plants systems. The increasing number of systems also increases challenges in plant maintenance. Nevertheless, there is no evidence of these potential challenges actually occurring.

Severe accident mitigation systems were constructed at all reactors in Finland in the 1980s and 1990s, and therefore only minor improvement possibilities for severe accident mitigation were found as result of the stress tests.

An important part of the stress tests was a reassessment of natural hazards, including earthquakes, flooding, and extreme weather phenomena. Probability estimates of these threats have been verified, and the effects of external hazards exceeding the design bases have been assessed. Only minor improvements were found to be beneficial, such as improving the flood protection of certain buildings and anchoring some components to the floors in order to improve their earthquake resistance. As regards emergency preparedness, organisations have been improved so that they can operate during a multi-unit accident.

The effects of safety improvements were quantified by a PSA.

3.4 GERMANY case study summary

The German report [5] describes the legal and regulatory framework for the continuous improvement of nuclear safety in Germany. Implemented safety improvements of German NPPs before as well as after the accident at the Fukushima Daiichi NPP site are presented. The focus is laid on those safety improvements that enhance accident management measures in the preventive and mitigative domain.

To ensure that nuclear safety is continuously improved in the operating NPPs in Germany, a comprehensive programme to monitor the progressing state of the art in nuclear safety has been established, firstly to identify potential safety improvements and secondly to assess the necessity to implement a safety improvement measure based on its safety significance and the applicability of findings to German NPPs.

The interaction of research in nuclear safety, continuous evaluation of operating experiences of German NPPs and NPPs abroad, analysing lessons learned from major accidents and deriving actions to further optimise nuclear safety, insights from deterministic and probabilistic safety analyses as well as the concept of continuous supervision are considered an effective set of instruments to identify and implement safety improvements and a continuous optimisation of nuclear safety.

German NPPs in operation have been designed according to safety standards of the 1970s and 1980s. The practical elimination of events leading to early or large releases at German NPPs is demonstrated by the interaction of plant operation, high reliability of the safety system, and comprehensive accident management. It can be illustrated by five tiers:

The first tier forms the design of systems, structures and components of high reliability and quality. One example is the application of the concept of basic safety developed in the late 1970s to prevent a catastrophic failure of those components, characterised by the following principles:

- high-quality materials, especially with respect to fracture toughness;
- conservative stress limits;
- avoidance of peak stresses by optimisation of the design;
- ensured application of optimised manufacturing and test technologies;
- knowledge and evaluation of existing flaws;
- accounting for the operating medium.

Later, this concept was developed further to the integrity concept, proven in practice and presenting an important contribution in terms of damage precaution.

The second tier is characterised by the highly reliable safety systems. It has to be demonstrated for postulated design basis accidents, e.g. large-break loss-of-coolant accidents, that the capacity of the emergency core cooling systems can cope with a guillotine break of a reactor coolant line. In addition, safety analyses for different leak sizes and locations have to be performed to prove conformance with established acceptance criteria, like the emergency core cooling criteria.

The third tier is represented by the measures of preventive plant-internal accident management. The licensee has to retain pre-planned measures to restore the fundamental safety functions. The effectiveness of these measures has to be demonstrated by safety analyses (realistic models and boundary conditions).

Mitigative plant-internal accident management can be considered as the fourth tier to achieve avoidance of large and early releases. Again, the effectiveness of these measures has to be demonstrated by safety analyses applying realistic models and boundary conditions.

The above-mentioned four tiers are based on deterministic approaches. A complementary probabilistic safety analysis (PSA) can be considered as the fifth tier. By PSA level 1 and PSA level 2, the achievement of practical elimination can be substantiated. It can be demonstrated that the implemented design features, periodic testing and in-service inspections, together with preventive and mitigative measures of plant-internal accident management, will lead to very low values of large early release frequencies (LERF).

The continuous improvement of German NPPs resulted in a safe fleet of NPPs with highly reliable safety systems and additional safety features for plant-internal accident management. Confidence that today’s radiological objective for nuclear power plants - allowing only limited off-site countermeasures in area and time - can be met is achieved by the existing NPPs due to the interaction of safe plant operation, highly reliable safety systems, and a comprehensive plant-internal accident management programme.

As the responsibility for nuclear safety rests with the licensee by law, a system of independent review of proposed safety improvements by the competent authority with consultation of authorised experts, in particular TSOs, is seen as an effective approach to increase trust in the reliability and efficiency of the proposed measures to improve safety. In addition, experts of the RSK provide independent recommendations for further improving nuclear safety in Germany.

3.5 ROMANIA case study summary

The DiD principle and the 10-year periodic safety review had already been integral parts of the Romanian nuclear legislation before the implementation of Council Directive 2014/87/Euratom. The approach for legal implementation of Articles 8a-8c of Directive 2014/87/EURATOM was consistent with developments in the international nuclear communities.

The Romanian report [6] outlines the Romanian legal framework for nuclear safety and the legal and technical implementation of the safety improvements in nuclear power plants in Romania.

The legislation is available only in Romanian language. The national requirements of the plant vendor's country (Canada) together with international requirements have also been endorsed.

Following the Fukushima Daiichi accident, the Romanian Regulatory Body, CNCAN, focused initially on technical reviews of the protection of the plant against extreme external events and of beyond-design-basis accident analysis, severe accident management, and emergency response. After more information became available on the organisational factors that have contributed to the accident, CNCAN used the lessons learned to improve the national regulatory framework, its practices for regulatory oversight of licensee's safety culture, and its own safety culture.

New regulations and regulatory guides have been issued or are in the process of being issued and work is on-going for the implementation of the National Strategy for Nuclear Safety and Security.

The main highlights of the information provided in the report are given below:

The implementation of plant safety upgrades in Romania is driven by both approaches, bottom-up and top-down.

Safety upgrades can result from the following situations: due to the development of technology (resulting in improved performance or reliability of operation); following PSR results; following international recommendations; following the revision of codes, standards, and regulatory requirements.

The Romanian safety upgrades have been driven mainly by the licensee, benefiting from the strong support of the regulatory body in elaboration of the concepts, technical solutions, and the schedules for the implementation. The licensee-driven safety upgrade process is fully consistent with the primary responsibility for nuclear safety. CNCAN has actively monitored the implementation status of the safety improvements. Most relevant safety upgrades were implemented after the Fukushima accident.

Maintaining an equivalent level of nuclear safety for all units (of the same design) on site could be also a good driver for safety upgrades. Taking advantage of the similar design of all nuclear units at the Cernavoda site, and due to the fact that commissioning was realised one by one for the units (they were not commissioned in the same year), the feedback programme at the Cernavoda site was very useful, assuring that reasonably practicable design modifications and improvements carried out for Unit 2 were also implemented for Unit 1.

A PSR represents an efficient means and opportunity to identify and implement the necessary safety improvements.

The licensee benefits from the membership in international organisations, seeking to achieve the highest levels of operational safety and performance. The membership in similar technology owners' associations provides the necessary support to resolve technical and operating problems for all members, and the recommendations arising from international memberships may constitute drivers for safety upgrades in the plant.

Safety benefits should be the strongest argument for changing the regulations. Regulatory-driven safety upgrades require substantial resources for the elaboration of well-defined regulations. There is a continuous effort of CNCAN to harmonise its practices with those already implemented by the international community. Being a member of international working groups relevant to nuclear safety, CNCAN is following the developments and recommendations arising from these dedicated group efforts. This effort also leads to the development of new regulations and, as result, to requirements for new safety upgrades in the plant.

Complex changes in the field of safety should receive sufficient attention, time and resources, and should involve all the necessary parties in order to achieve good results.

3.6 HUNGARY case study summary

The study [7] outlines the legal framework for nuclear safety in Hungary, safety improvements at the Paks nuclear power plant, and efforts to implement in practice Articles 8a-8c of Directive 2014/87/Euratom.

The Hungarian Parliament approved the Act CXVI on Atomic Energy in December 1996. It stipulates that "In the use of atomic energy, safety has priority over all other aspects", and that “the Licensee is obliged to undertake continuous activities to improve safety, taking account of its operational experience and the new safety related information”.

Several government decrees and ministerial decrees have been issued to implement the requirements of the Act on Atomic Energy, the most important being the Govt. decree 118/2011. (VII. 11.) Korm. on the nuclear safety requirements (Nuclear Safety Code) for nuclear facilities, and Govt. decree 487/2015. (XII. 30.) Korm. on the protection against ionising radiation and the corresponding licensing, reporting and inspection system.

Nuclear safety requirements (Nuclear Safety Code) were issued as an annex to Govt. Decree 118/2011 (VII. 11.) Korm. Volume 3 (“Design requirements for nuclear power plants”) contains general nuclear safety-related requirements concerning the design of nuclear power plants. In order to be prepared for the construction of the new nuclear power plant units, Volume 3/A (“Design requirements for new nuclear power plant units”) was published. It is required by the Act on Atomic Energy that regulations shall be reviewed at least every 5 years, and in practice regulations are revised more frequently than the 5-year review period. An important aspect of regulation development was to take into account the recommendations of international organisations, e.g. WENRA reference levels.

At the Paks site, there are four VVER-440/213 type reactor units in operation. The four units were commissioned between 1983 and 1987 and are currently in good technical condition. Improvement of safety of Paks NPP has been implemented in several steps in the last decades, milestones are summarised in the report [7].

The reassessment of safety of the Paks NPP according to internationally recognised criteria was performed during the AGNES (Advanced, General and New Evaluation of Safety) project between 1991 and 1994. It was demonstrated that there were several alternatives to enhance the safety of the nuclear power plant and feasible measures were identified and implemented.

A periodic safety review (PSR) is performed every ten years, during these the technical status and safety level of the facility are reviewed. A Safety Analysis Report is regularly updated, follows and analyses the safety impact of different measures and modifications, and evaluates safety performance according to international practice.

After the accident at the Fukushima Daiichi NPP, the Hungarian Atomic Energy Authority issued the requirements for the licensee's re-assessment. During the re-assessment it was demonstrated that the Paks NPP is in compliance with the licensing conditions and that the design basis established during the construction of the plant was extended through a series of safety improvement programmes. Several corrective actions were proposed in order to increase the safety margins, and a detailed implementation action plan was elaborated and executed.

The Paks NPP pays special attention to utilising international experience; more than 40 international reviews have taken place since 1984.

It should be emphasized that Hungary has committed to a new NPP construction project in 2014 with the objective to build two VVER-1200 type units next to the existing units at the Paks site. The new NPP project is currently in its early phase, detailed information will be available in the coming years.

It was demonstrated that safety of the units was improved continuously before as well as after the accident at the Fukushima NPP. Requirements of Article 8a to 8c of Directive 2014/87/EURATOM are implemented in the Hungarian regulatory framework.

4 ANALYSIS

4.1 Technical rationales for safety upgrades

The most important drivers for the safety improvements in the European Nuclear power plants, reflected in all country-specific reports, include:

- defence in depth (DiD), including the concept of Design Extension Conditions (DEC);
- WENRA reference levels (2014) [9];
- requirements for new reactors, as applied also to existing reactors
- periodic safety reviews (PSR)
- probabilistic safety analyses (PSA)
- operating experience.

A brief discussion of each of the above drivers is given below.

4.1.1 Defence in depth

Defence in depth (DiD) is mentioned in all country reports as the fundamental technical rationale to evaluate safety and as an important driver to identify any further proposals for safety improvements.

Further, the concept of Design Extension Conditions (DECs) has been integrated in all respective regulatory frameworks and is clearly considered as an important concept for identification of further safety improvements. This is consistent with the fact that the heads of the competent national authorities of all countries for which specific reports for Task 3 have been prepared, are members of WENRA.

It is noted that the wording for the Finnish classification of DiD differs from that used by many other countries. Finland, for example, does not use the term design basis accident. Although this seems to be a difference in wording, there is probably no difference regarding safety applications. Nevertheless, for DiD, the classification seems to be the same as in other countries (severe accidents with core melt are on level 4).

It can be summarized that priority is given to preventive measures (to prevent core degradation) in all 6 countries.

4.1.2 WENRA Reference Levels 2014

The WENRA Reference levels (2014) [9] have been implemented (Slovenia) or are in the process of implementation in the national regulatory frameworks as stated in all country specific reports and reported by WENRA [11]. This is probably the main reason for the rather high degree of harmonisation noted in reviewing the country-specific approaches.

4.1.3 Requirements for new reactors

Countries with a new NPP under construction (i.e. France and Finland) use the requirements for the improved new designs as references to be considered also for existing NPPs. Countries without new build projects apply the new requirements as a reference target to identify safety improvements for existing reactors.

The applicability of requirements for new reactors to existing reactors is not explicit but may be decided on a case-by-case basis. In this way, the requirements for new reactors serve as an important and continuous driver for identifying potential safety improvements in existing installations. In addition, the WENRA safety reference levels for existing reactors [9] and the WENRA safety objectives for new NPP designs [10] are consulted too, as they represent the consensus of all competent European authorities regulating NPPs.

In Hungary, different targets and requirements exist for old and planned reactors. Volume 3 of Annex of Govt. Decree 118/2011 (VII. 11.) deals with the design requirements for operating nuclear power plants, Volume 3a of the Annex is for the design requirements for new nuclear power plant units.

4.1.4 Periodic Safety Review (PSR)

We observe that in all countries the PSR is now a regulatory requirement, typically with a periodicity of 10 years. The results of PSR serve as another important driver for potential safety upgrades. The studies also revealed difference in practices of performing PSRs. This is mainly due to country-specific situations. For example, in France, a two-step approach is followed: In the first step, a PSR for the reactor series under review is performed and in a second step, a plant-specific PSR is performed in addition. This approach requires a standardised fleet of reactors with only minor plant specific modifications.

Stress tests and subsequent safety improvements also represent an important step.

4.1.5 Probabilistic Safety Analysis (PSA)

The PSA is identified in all national reports to be an important tool for evaluating safety improvements in the NPP. Two types of the use of a PSA can be distinguished:

- to identify potential safety improvements (for instance by examining how dominant accident sequences can be made less probable); this is an “a priori” use of a PSA;
- to evaluate, once a safety improvement has been decided upon (based on other “drivers”) and implemented, to which extent it reduces the CDF of LERF; this is “a posteriori” use of a PSA.

4.1.6 Operating experience

An important rationale for safety upgrades is operating experience. Obviously, the most significant operating experience in the last decade was the Fukushima accident leading to safety improvements in many European NPPs. It also initiated the EU stress tests, in which many opportunities for safety improvements were identified. The country-specific studies demonstrated that further major or minor events past and present have also triggered certain safety improvements, as e.g. described in the specific reports of Finland and Germany.

The EU Clearinghouse on Operating Experience Feedback for Nuclear Power Plants also had an active role in analysing the results of EU stress tests. The EU Clearinghouse was established in 2008 to enhance nuclear safety through improvement of the use of lessons learned from Operating Experience. The central office of this network is located at the European Commission Joint Research Centre (JRC) in Petten, (Netherlands). The Clearinghouse is comprised of dedicated staff from JRC and nuclear safety regulators of Member States and their TSOs.

4.2 Diversity in country-specific approaches

Some level of diversity in country-specific approaches has been observed in the country-specific reports. It is noted that further in-depth technical analyses of the seemingly diverse approaches discussed below might reveal on the one hand more similarities or, on the other hand, more opportunities for further harmonization or perhaps also for potential further safety improvements. Such in-depth analyses would be useful for many reasons, including the excellent opportunity for younger experts to get in-depth knowledge on the technical safety reasoning in EU member countries. It would also require a lot of time and resources in the potential follow-up projects.

The diversities observed have been grouped by the authors of the report into the following groups:

- Feedback from external advisors, research, operating experience;
- Probabilistic safety targets;
- Diversity in drivers and status of regulations;
- Documenting the technical rationales for exceptions;
- Design Extension Conditions;
- Culture for safety and increasing technical complexity.

A brief discussion of each of the above groups is given below.

The differences discussed below have been chosen primarily to illustrate the diversities in the national approaches and to facilitate future discussion leading towards more profound understanding and potentially also to learning from each other. The intention of the authors was neither to define nor to promote any good practices.

4.2.1 Feedback from external advisors, research, operating experience

The German report identifies that the Reactor Safety Commission (Reaktorsicherheitskommission - RSK), an external independent advisory body to the federal regulator, has a proactive role in identifying safety improvements. This seems to be different from other countries, where such external independent bodies are rather more endorsing or advising on safety improvements. In addition, it appears that the German report, compared with other national reports, puts more emphasis on the “state of the art in science and technology”, “research” and “operating experience feedback” as “drivers” for identifying further safety improvements.

Similarly, the Romanian report puts a comparatively strong emphasis on operating experience feedback.

Safety research seems to be perceived in all countries as a very useful driver for safety. It also seems that operating experience has a much more immediate impact on the safety improvements than research.

4.2.2 Probabilistic safety targets

Three country reports (Slovenia, Romania, Finland) mention quantitative probabilistic safety limits or targets. These could be used as “drivers” among other types of drivers for safety improvements, provided that the assumptions and uncertainties in the input data and models of a PSA, which inevitably lead to uncertainties in the absolute results of a PSA, are properly understood and documented.

4.2.3 Diversity in drivers and statuses of regulations

A variety of approaches in the development of the regulations has been noted in the country-specific reports in terms of the balance between the bottom-up and top-down approach. Slovenia, for example, seems to rely comparably more on the legal solutions based on the technical solutions proposed by the licensee (bottom-up), whereas for other countries a more proactive approach of the regulators (top-down) may be noted.

A good example of the top-down approach may be a global concept known as the “hardened safety core”, which was decided upon and implemented in the French NPPs after the Fukushima accident. It appears that this approach brought numerous safety improvements, which are also reported in other countries without any explicit reference to the “hardened safety core”.

Also, diverse levels of the involvement of TSOs in the conceptualisation and drafting of the regulations can be noted. Slovenia, for example, does not formally involve TSOs.

It appears that not all regulators can issue binding regulations. In Slovenia, for example, the binding regulations can be proposed by the regulator to be issued by the ministers of the government. In Romania and in Finland, the regulator can do that independently of the government.

The regulations are available in national languages and in some cases also in foreign languages (e.g. for information purposes). Direct endorsement (e.g. without transposition into the national regulations) of the WENRA Reference Levels (2014) [9] has not been noted.

The diversity in the development and issuance of the regulations appears to lead to similar results.

4.2.4 Documenting the technical rationales for exceptions

For some exemptions noted in the country reports (e.g., Slovenia, PSA Levels 1, 2 and 3 are required for new reactors and PSA Levels 1 and 2 for existing reactors), the technical rationale is not given in the regulations and might not be easily available in the public domain.

It appears that we pay a lot of attention to the “technical rationale” for safety improvements that are implemented. It also appears to be worthwhile to focus on documenting sufficiently the “technical rationale” for exceptions or exemptions in the future.

4.2.5 Nuclear rapid action force

The creation of a FARN (Nuclear rapid action force) is described in the French report, without similar actions reported in other country reports. While it is clear that such an approach may be very valuable in a country with many nuclear sites, it may also be not very useful for countries with only a few sites. A potential consensus on the technical rationale behind nuclear rapid action forces might potentially bring an opportunity for cross-border support at EU or regional level.

4.2.6 Design Extension Conditions (DEC)

The objectives of implementations of design extension conditions in the national regulatory frameworks are clearly similar. The applied terminology could be different and aligned to the national regulatory framework and possibly also the native language. For example, the wording for the Finnish classification of DECs includes accident conditions without core melt, but more severe than design basis accidents:

- DEC-A: An accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function.
- DEC-B: An accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment.
- DEC-C: An accident caused by a rare external event and which the facility is required to withstand without severe fuel failure. This includes for example some extreme weather phenomena and the impact of a large aircraft.

For postulated core melt accidents, the traditional term ‘severe accidents’ is still used, albeit as part of the design.

An interesting topic for future discussions might be the role of hazards more severe than those considered for the design basis of an NPP and its relation to design extension conditions.

4.2.7 Culture for safety and increasing complexity

It is noted that the main focus of this particular project and the post-Fukushima improvements in the EU and WENRA regulations has been on the improved safety of nuclear installations mainly through improvements (and inevitably increasing complexity) of the technology (e.g., design extension conditions, hardened safety core, etc.).

It is also noted here that the more complex technologies, introduced through the Safety Directive, the WENRA RL [9] and the related national regulations, will have to be ordered and constructed by the organisations and operated by personnel. It is well known that increased complexity (in technology or in organizations) could impede or even counteract safety improvements. With this in mind, we highly recommend that improvements in personnel (education and training) and organizations (culture for safety, knowledge management) should also continue to receive appropriate attention in the future.

Another point to be discussed in the future could be increasing complexity versus independence of levels of defence in depth to verify or possibly improve the balance between both concepts.

5 CONCLUSIONS

This report summarises the country reports from Slovenia, France, Finland, Romania, Hungary and Germany.

After having analysed the national reports, the consortium highlights that a rather high degree of harmonization in technical rationales for safety improvements of nuclear installations has been observed.

Some level of diversity in country-specific approaches has also been observed in the country-specific reports. Differences can be detected between (1) small and big countries with NPPs at 1-2 or many sites and (2) countries planning or commissioning new nuclear power plants. It is noted that a further in-depth technical analysis of the seemingly diverse approaches discussed might reveal more similarities on the one hand or more opportunities for further harmonization or perhaps also for potential further safety improvements on the other hand, including:

- **Suggestion 1:** Diverse implementations of PSRs and DECs exist in the analysed countries, aligned among others with the size of their fleets, their legal frameworks and possibly also with their national languages. It is therefore suggested to **enhance the mutual understanding of approaches for periodic safety reviews with specific focus on the implementation of the DEC approach in member states**;
- **Suggestion 2:** The regulatory requirements and technical solutions designed to improve safety might increase the technical complexity of the plants and/or demote the independence of the levels of defence. It is therefore suggested to **promote discussions on the balance between the increasing complexity in plant designs and the independence of the levels of defence in depth to avoid a potential decrease in nuclear safety. Practical examples might include side effects of modifications, shared equipment in multi-unit sites, and I&C. A development of guidance may be needed to adequately support decision-making**;

Such in-depth analyses, coordinated by WENRA or ENSREG, would be useful for many reasons, including the excellent opportunity for younger experts to gain hands-on experience in multicultural working environments and in-depth knowledge on the technical safety reasoning in EU member countries. It would also require significant resources in the potential follow-up projects.

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Reactor Safety Upgrades in Slovenia

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Praktična uveljavitev členov 8a-8c direktive 2014/87/EURATOM: Varnostne nadgradnje reaktorjev v Sloveniji

Povzetek

Poročilo povzema varnostno nadgradnjo jedrske elektrarne in zakonodaje v Sloveniji po uveljavitvi Evropske Direktive 2014/87/Euratom 8.7.2014. Slovenijo smo izbrali kot referenčni primer zato, ker so zakon in spremljajoči pravilniki že sprejeti, jedrski upravni organ je tudi že potrdil načrt varnostnih nadgradenj v edini slovenski jedrski elektrarni v Krškem, kjer varnostne nadgradnje že tudi vgrajujejo. Ocenjujemo, da je dokončanih že okoli 2/3 načrtovanih varnostnih nadgradenj.

Na kratko je orisan slovenski zakonodajni okvir za jedrsko varnost, ozadje koncepta »razširitve projektnih stanj« (v angleščini Design Extension Conditions, DEC) ter tehnična in pravna implementacija DEC v Sloveniji. Sledijo razpoznani primeri potencialno dobrih praks, potencialnih izzivov in potencialnih vrzeli v dosedanjem pristopu k varnostnim nadgradnjam.

Koncept »razširitve projektnih stanj (DEC)« smo v središče postavili zato, ker po mnenju združenja evropskih jedrskih regulatorjev WENRA opredeljuje smiseln (angleško reasonably practicable) pristop k izogibanju (angl. avoid), preprečevanju (angl. prevent) oz. blaženju (angl. mitigate) težkih jedrskih nesreč in njihovih posledic, še posebej zgodnjih (angl. early) in velikih (angl. large) izpustov radioaktivnih snovi v okolico. Pristopa globinske obrambe in rednih 10 letnih občasnih varnostnih pregledov sta bila v Sloveniji uzakonjena in v uporabi že pred uveljavitvijo Evropske Direktive 2014/87/Euratom.

Poročilo sodi v delovni sklop 3 projekta ENER/17/NUCL/S12.769200, ki ga je Direktorat za energijo Evropske komisije naročil pri združenju ETSON v podporo državam članicam pri vzpostavljanju konsistentne praktične implementacije zahtev Evropske Direktive 2014/87/Euratom v nacionalne zakonodaje.

Implementation in Practice of Articles 8a-8c of Directive 2014/87/EURATOM

Reactor Safety Upgrades in Slovenia

Abstract

The report documents the safety upgrades in Slovenia related to the Council Directive 2014/87/Euratom of 8 July 2014. Slovenia has been chosen as the reference case, since the legislation and rules are in place, the regulator has confirmed the plan of upgrades in the country's only nuclear power plant at Krško and the upgrades of the nuclear power plant are being commissioned. Currently, the commissioning of the upgrades has been estimated to be completed beyond 2/3.

The report outlines the Slovenian legal framework for nuclear safety, background on the Design Extension Conditions (DEC) and the legal and technical implementation of the DEC in Slovenia. This is followed by identification of potential good practices, challenges and gaps. The report will serve as a benchmark for the analysis of situations in other countries.

The Design Extension Conditions have been chosen for the central role in the report because they were conceived to provide **reasonably practicable** approach to **avoid**, **prevent** and/or **mitigate** the beyond design basis accidents and their consequences, especially the **early** and/or **large radioactive releases.** The defense in depth principle and the 10 year periodic safety review have been already integral part of the Slovenian nuclear legislation before the implementation of the Council Directive 2014/87/Euratom.

The report is part of the Task 3 of the project ENER/17/NUCL/S12.769200 contracted by the European Commission, DG ENER to ETSON, to support Member States in achieving a consistent practical implementation of the provisions set out in the Council Directive 2014/87/Euratom of 8 July 2014.

Table of Contents

List of Tables

List of Figures

List of Abbreviations

Not Applicable.

1 Introduction

1.1 Background

After the nuclear accident at the Fukushima Dai-ichi nuclear power plant in 2011, several activities have been started to further strengthen nuclear safety worldwide. In Europe, Council Directive 2009/71/Euratom [1] establishing a Community framework for the nuclear safety of nuclear installations was amended by Council Directive 2014/87/Euratom of 8 July 2014 [2]. In particular, Articles 8a to 8c, reproduced for convenience of the reader in Table 1, Table 2 and Table 3, introduced:

- the nuclear safety objective (Table 1),
- requirements for the implementation of the nuclear safety objective (Table 2) and
- requirements for the initial assessment / periodic safety reviews (Table 3).

The aim of this project, contracted to European Technical Safety Organisation Network (ETSON) by the European Commission, DG ENER with the contract no ENER/17/NUCL/S12.769200, is to support Member States in achieving a consistent practical implementation of the provisions set out in the new Directive. To meet this aim the work performed contributes to identify gaps, common approaches, and areas of good practice and makes recommendations on practical steps for ambitious implementation of the Directive.

As defined in the inception report [3], the Task 3 entitled “Performing a detailed study on the safety upgrades in existing reactors in selected Members States” analyzes in some detail the practices and approaches in selected Member States to implement safety upgrades in existing nuclear reactors.

Table 1: Article 8a (Nuclear safety objective for nuclear installations) of [2]

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).

Table 2: Article 8b (Implementation of the nuclear safety objective for nuclear installations) of [2]

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.

Table 3: Article 8c (Initial assessment and periodic safety reviews) of [2]

Member States shall ensure that the national framework requires that: (a) any grant of a licence to construct a nuclear installation or operate a nuclear installation, is based upon an appropriate site and installation-specific assessment, comprising a nuclear safety demonstration with respect to the national nuclear safety requirements based on the objective set in Article 8a; (b) the licence holder under the regulatory control of the competent regulatory authority, re-assesses systematically and regularly, at least every 10 years, the safety of the nuclear installation as laid down in Article 6(c). That safety reassessment aims at ensuring compliance with the current design basis and identifies further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards, using as a reference the objective set in Article 8a.

1.2 Purpose of the report

The first detailed study in Task 3, documented in this report, focuses on safety upgrades in Slovenia. Slovenia has been chosen as the reference case, since the legislation and rules are in place, the regulator has confirmed the plan of upgrades in the country's only nuclear power plant at Krško and the upgrades of the nuclear power plant are being commissioned. Currently, the commissioning of the upgrades has been estimated to be completed beyond 2/3.

The report focuses on Design Extension Conditions (DEC) in the existing plant at Krško. The Design Extension Conditions have been chosen for this report because they were conceived by the regulators to provide **reasonably practicable** approach to **avoid**, **prevent** and/or **mitigate** the beyond design basis accidents and their consequences, especially the **early** and/or **large radioactive releases.** In addition, the defense in depth principle and the 10 year periodic safety review have been already integral part of the Slovenian nuclear legislation when Council Directive 2014/87/Euratom of 8 July 2014 [2] was published (for more details see section 2).

In the report, the outline and analysis of the technical and legal implementation of Design Extension Conditions is supplemented with the outlines of the Slovenian nuclear regulatory framework and the time-line of the major decisions leading to the present state reactor safety upgrades in Slovenia.

The report will serve as a benchmark for the analysis of situations in other countries. The inception report [3] defines that the final selection of the remaining countries and specific topics to be analyzed in more detail – the candidate countries being France, Finland, Germany, Hungary and Romania - will be confirmed after analyzing the results of the survey in Task 2 and the 1st workshop. In this way, the selection of candidate Countries offers the widest possible range in terms of nuclear strategies involved, namely prolonged operation (e.g. Slovenia), construction of new units (e.g. France, Finland, Hungary), phase out (Germany) and specific technologies (e.g., Romania).

The detailed studies of selected countries will be finally used as an input to the final report summarizing the status and identifying best practices and possible gaps in the national implementations of safety upgrades in existing nuclear reactors.

1.3 Organization of the report

The main sections of the report include:

- 2 Slovenian legal framework for nuclear safety – contains a brief overview of the history and main features of the Slovenian legal framework for nuclear safety.
- 3 Background on Design Extension Conditions – attempts a brief overview of the development and current status of the Design Extension Conditions.
- 4 Legal and technical implementation in Slovenia – the outline and analysis of the technical and legal implementation is supplemented with the outlines of the Slovenian nuclear regulatory framework and the time-line of the major decisions leading to the present state of affairs.

- 5 Discussion – attempts to highlight potential best practices, gaps and challenges in the Slovenian implementation.
- 6 Conclusions – summarizes the main messages of the report.

Draft version of this report has been submitted for comments to the Slovenian nuclear regulator (Slovenian Nuclear Safety Administration), the EC DG ENER and to the ETSON consortium. Comments received by the SNSA [4], EC DG ENER [5] and BelV [6] are incorporated in this version of the report.

2 Slovenian legal framework for nuclear safety

The main national and international legislative acts and events that shaped the Slovenian legal framework for nuclear safety are outlined in section 2.1 in the chronological order.

The outline of the national legal framework for nuclear safety in Slovenia follows, describing in some detail the following documents:

- Resolution on Nuclear and Radiation Safety (section 2.2),
- Ionising Radiation Protection and Nuclear Safety Act (section 2.3) and
- Government Decrees, Rules, Regulatory Guides and Regulatory decisions (section 2.4). It is important to mention here that the Decrees, Rules and Regulatory guides in most cases do not provide the sufficiently detailed criteria for technical implementation. These are in most cases detailed in the application for the modification of the license and adopted in the particular Regulatory decisions.

2.1 The context

The construction of the Krško NPP formally started in 1975.

The nuclear safety licensing framework in Slovenia has been established by the Yugoslav Federal law in 1976 [7], complementing the already existing radiation protection rules. The primary responsibility for the licensing has been delegated to the members of the federation (e.g., Slovenia), with the federation preserving the responsibility to resolve possible conflicting decisions of the members.

In 1980, Slovenia adopted detailed regulatory framework, including the role and establishment of TSOs [8]. Four existing entities were instituted as TSO's with well-defined and complementary fields of expertise. The nuclear and radiation safety was attributed to the Jožef Stefan Institute [9].

The formal endorsement of the nuclear safety regulations in the vendor country (USA) in Slovenia and Yugoslavia was established in 1984 [10]. Such approach has been considered adequate for the licensing of the nuclear power plant at Krško (vendor Westinghouse, operator NEK d.o.o., first criticality in 1981, commercial operation since January 1, 1984). The US 10CFR50 approach is based on the **defense in depth** approach.

The nuclear regulatory body SNSA (Uprava Republike Slovenije za jedrsko varnost, Slovenian Nuclear Safety Administration) has been established in 1987 [11].

Slovenia became independent in 1991 and ratified the International Atomic Energy Agency (IAEA) Convention on nuclear safety in 1996 [12].

In 2002, the legal framework for nuclear safety has been redefined with a new "Ionising Radiation Protection and Nuclear Safety Act" [13]. The Act [13] introduced the **10 year periodic safety review** and cancelled the endorsement of the nuclear safety regulations in the vendor (USA) country. It also fundamentally changed the organizations of the TSOs: a license is now given by the regulator to applicants fulfilling certain criteria. The fields of expertise of TSOs are not complementary. The licensee is required to complement any formal request for the license amendment with the safety assessment developed by a licensed TSO. The licensee is free to select and contract any among the licensed TSOs in a competitive process.

This Act [13] has been frequently amended, as discussed in more detail in section 2.3. It also resulted in a series of novel decrees and rules (see section 2.4).

By joining the European Union in 2004, Slovenia inter alia adopted the EURATOM treaty [14].

Council Directive 2009/71/Euratom [1] imposed obligations on the Member States to establish and maintain a national framework for nuclear safety. That Directive reflected the provisions of the main international instruments in the field of nuclear safety, namely the Convention on Nuclear Safety [12], as well as the Safety Fundamentals [15] established by the International Atomic Energy Agency ('IAEA').

The IRRS (Integrated Regulatory Review Service) mission in September 2011 triggered the adoption by the Parliament of the "Resolution on Nuclear and Radiation Safety in the Republic of Slovenia for the period 2013-2023" [16]. Resolution [16] is discussed in some more detail in section 2.2.

The Fukushima Dai-ichi accident in 2011 and the lessons learned during the European Stress tests [17] initiated the Council Directive 2014/87/Euratom [2]. This requires Member States to ensure that the national framework requires that "license holders are to regularly assess, verify, and continuously improve, as far as reasonably practicable, the nuclear safety of their nuclear installations in a systematic and verifiable manner. That shall include verification that measures are in place for the prevention of accidents and mitigation of the consequences of accidents, including the verification of the application of defense-in-depth provisions."

2.2 Resolution on Nuclear and Radiation Safety

The Resolution on Nuclear and Radiation Safety [16] is exhibiting the high-level political commitment to ensure nuclear and radiation safety in Slovenia. It has been prepared by the Government and adopted by the Parliament as a response to the recommendation of the IRRS (Integrated Regulatory Review Service) mission in September 2011, asking for "*the development of a national policy and strategy for nuclear safety which would be supported by a national co-ordinated plan to ensure that appropriate national infrastructure is in place to secure its delivery*".

The commitment to ensure nuclear and radiation safety at the highest political level has been recommended in the 2010 edition of the basic IAEA GSR Part 1 standard: Governmental Legal and Regulatory Framework for Safety (National legal and administrative framework for nuclear and radiation safety) [18].

In the first part, the Resolution highlights the ten fundamental safety principles governing the nuclear safety legislation of the Republic of Slovenia. The description of the main nuclear and radiation safety activities in the country, the compliance of the Slovenian regulations with international regulations, and the description of the existing legislation and organization of the regulatory bodies follows. The Resolution also emphasizes the need for appropriate personnel to ensure nuclear and radiation safety, which includes research and development activities. Of particular importance is commitment to public participation and commitment to quality, excellence in leadership and safety culture.

The Resolution has no executive power.

2.3 Ionising Radiation Protection and Nuclear Safety Act

The main part of the executive legislation is the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV). The first edition of the Act, adopted in 2002 [13], has undergone various modifications. The latest version of the act (ZVISJV-1) has been adopted in 2017 [19] after extensive modifications, related also to the Council Directive 2014/87/Euratom [2] and the WENRA reference levels [20].

In the period 2002-2017, it has been customary to publish the official English translation of the Act in the Official Gazzete and on the web site of the SNSA (http://www.ursjv.gov.si/en/legislation_and_documents/). This practice has been discontinued in 2016, reinstated in June 2018 and again (temporarily?) discontinued in September 2019 with the major refurbishment of the Slovenian .gov websites.

This act shall regulate ionising radiation protection with the aim of reducing as much as possible the detrimental effects on human health and contamination of the living environment, while at the same time enabling the development, production and use of radiation sources and performing radiation practices. With regard to radiation sources intended for producing nuclear energy, this Act shall regulate the implementation of nuclear safety measures and also, in the case of the use of nuclear materials, special protection measures.

The act, among others, also defines the competent regulatory body and the licensing framework. It also concedes the definition of detailed rules to the set of Governmental Decrees (adopted by the Government) and Rules (adopted by the relevant Ministers).

2.4 Decrees, Rules, Regulatory Guides and Regulatory Decisions

Executive regulatory documents, defined by the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19], include:

- Governmental decrees, adopted by the Government (see section 2.4.1);
- Rules, adopted by relevant Ministers (see section 2.4.2);
- Regulatory guides, published by the SNSA (see section 2.4.3) and
- Decisions of the SNSA (granting, modifying or revoking licenses, see section 2.4.4).

A formal role of the TSOs in the preparation of the Decrees, Rules and Regulatory Guides is not defined by the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19]. The TSOs nevertheless have the possibility to participate in the public consultations of the draft documents together with any other interested legal entity or citizen.

With respect to the Regulatory Decisions, the formal role of the TSOs is defined by the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19] indirectly: the licensee is required to complement any formal request for the license amendment with the safety assessment developed by a licensed TSO. The licensee is free to select and contract any among the licensed TSOs in a competitive process. In this way, the TSOs advice is formally considered in the Decisions of the SNSA.

2.4.1 Governmental Decrees

The Governmental Decrees with corresponding topics are defined in the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19]. They are adopted or modified by the Government.

Table 4: Governmental decrees (Status as of May 2018)

No.	Governmental Decree	Valid from	Off. Gaz. of the RS
UV1	Decree on activities involving radiation Decree amending the Decree on activities involving radiation Exemption levels for radionuclides Ba-133, Y-88 and Po-209	1.5.2004	48/2004 9/2006
UV2	Decree on dose limits, radioactive contamination and intervention levels	1.5.2004	49/2004
UV3	Decree on the areas of limited use of space due to a nuclear facility and the conditions of facility construction in these areas	28.4.2004	36/2004
	Decree amending the Decree on the areas of limited use of space due to a nuclear facility and the conditions of facility construction in these areas	7.10.2006	103/2006
	Decree amending the Decree on areas of restricted use due to nuclear facilities and on the conditions for construction in these areas	20.12.2014	92/2014
UV4	Programme on systematic monitoring of working and residential environment and raising awareness about measures to reduce public exposure due to the presence of natural radiation sources for the period 2016-2020	12.3.2016	19/2016
UV6	Decree on safeguarding of nuclear materials	22.4.2008	34/2008
UV8	Decree on the criteria for determining the compensation rate due to the restricted use of areas and intervention measures in nuclear facility areas	1.1.2015	92/2014
UV11	Decree on checking the radioactivity of shipments of scrap metal	1.1.2008	84/2007
	Decree on the implementation of Council Regulations (EC) and Commission Regulations (EC) on the radioactive contamination of foodstuffs and feedstuffs	3.6.2006	52/2006
	Decree amending the Decree on the implementation of Council Regulations (EC) and Commission Regulations (EC) on the radioactive contamination of foodstuffs and feedstuffs	15.5.2010	38/2010
	Decree on the method, subject of and conditions for performing a compulsory public utility service of long term surveillance and maintenance of landfill of mining and hydrometallurgical tailings resulting from extraction of and exploiting of nuclear mineral raw materials	10.10.2015	76/2015

The full list (Status as of May 2018) of Governmental decrees is for completeness given in Table 4. None of them is related to the design extension conditions or has been modified to accommodate the provisions of the Council Directive 2014/87/Euratom [2]. They will not be further discussed in this report.

The full texts of the Governmental Decrees are available (Status as of May 2018) for consultation in the web site of the SNSA (http://www.ursjv.gov.si/en/legislation_and_documents/). Please note that the practice to publish official translations to English language has been temporarily discontinued in 2017 and was resumed in June 2018 and again (temporarily?) discontinued in September 2019 with the major refurbishment of the Slovenian .gov websites.

2.4.2 Rules

The Rules and the topics to be further detailed within them are defined in the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19]. They are adopted or modified by the relevant Ministers.

The list (Status as of May 2018) of Rules to be adopted by the Minister of environment, responsible also for the nuclear safety, is for completeness given in Table 5. The two rules that have been affected by the Council Directive 2014/87/Euratom [2] are:

- JV5, Rules on radiation and nuclear safety factors [21], and
- JV9, Rules on operational safety of radiation and nuclear facilities [22].

Further details on the implementation of the Council Directive 2014/87/Euratom [2] within JV5 and JV9 are discussed in section 4.1.

Table 5: Rules of the Minister of the Environment (Status as of May 2018)

No.	Rule	Valid from	Off. Gaz. of the RS
JV1	Rules on the specialist council on radiation and nuclear safety	26.4.2003	35/2003
JV2/ SV2	Rules on the use of radiation sources and on activities involving radiation	28.3.2006	27/2006
JV3	Rules on authorised experts on radiation and nuclear safety	2.6.2006	51/2006
JV4	Rules on providing qualification for workers in radiation and nuclear facilities	14.5.2011	32/2011
JV5	Rules on radiation and nuclear safety factors	10.12.2016	74/2016
JV7	Rules on radioactive waste and spent fuel management	12.5.2006	49/2006
JV9	Rules on operational safety of radiation and nuclear facilities	31.12.2016	81/2016
JV10	Rules on radioactivity monitoring Rules amending the rules on radioactivity monitoring	21.3.2007 1.12.2009	20/2007 97/2009
JV11	Rules on transboundary shipments of radioactive waste and spent fuel	24.3.2004	22/2009
JV12	Rules on the transboundary shipment of nuclear and radioactive substances Rules amending Rules on the transboundary shipment of nuclear and radioactive substances	6.8.2008 7.6.2014	75/2008 41/2014

Full texts of the Rules are available (Status as of May 2018) for consultation in the web site of the SNSA (http://www.ursjv.gov.si/en/legislation_and_documents/).

Please note that the practice to publish official translations to English language has been temporarily discontinued in 2017 and was resumed in June 2018 and again (temporarily?) discontinued in September 2019 with the major refurbishment of the Slovenian .gov websites.

2.4.3 Regulatory Guides

The Regulatory Guides (the term “Praktične Smernice” in Slovenian language might be closer to “practical guidelines”) are developed and published by the discretion of the SNSA. The full list (Status as of May 2018) of published Guides is shown for completeness in Table 6.

The Council Directive 2014/87/Euratom [2] triggered the definition and development of PS 1.06 “Definitions of nuclear installations states” [23], which is devoted to the definitions related to design extension conditions. In this way, the PS 1.06 “Definitions of nuclear installations states” [23] provides interpretations of requirements of the Act ZVISJV-1 [19] and rules JV5 [21] and JV9 [22], which are considered acceptable to the regulatory body.

Table 6: Regulatory Guides by the SNSA (Status as of May 2018)

No.	The title of the Practical Guideline	Issue
PS 1.01	The content and scope of periodic safety review of radiation or nuclear facility	Issue 1, 6.5.2009
PS 1.02	Management of modifications in a radiation or nuclear facility	Issue 1, 11.1.2013
PS 1.03	The content of the safety analysis report of a low- and intermediate-level radioactive waste repository	Issue 1, 10.7.2012
PS 1.04	The content of the safety analysis report of radiation or nuclear facilities	Issue 2, 18.2.2016
PS 1.05	The use of general reference documents in administrative procedures Appendix to PS 1.05 (List of recognized general reference documents from January 2017)	Issue 1, 8.7.2014
PS 1.06	Definitions of nuclear installations states	Issue 1, 22.1.2018

2.4.4 Decisions of the SNSA

Decisions of the SNSA constitute the basic regulatory document: the licenses for the nuclear facilities are granted, modified or revoked through the Decisions.

The implementation of plant (safety) upgrades in Slovenia is mainly driven by two processes:

- Bottom-up: the conceptual proposals are prepared by the licensee, which are then, possibly with some modifications, formalized as the regulatory decisions. Traditionally, the TSO’s are not part of the discussions between the licensee and the regulatory body during these conceptual stages.
- Top-down: the changes in legislation usually trigger the regulator to require a specific modification in a form of formal Decision.

After the SNSA decision has been issued, the licensee, supported by the vendor, will propose a technical solution and the safety criteria. The safety criteria are usually taken from the US legislation and/or vendor's experience, as the Slovenian legislation (Act, Decrees and Rules) provide generic safety criteria without specifying the limiting values (the limiting dose for the member of the general population from the operation of nuclear facilities is limited to 1mSv/year in the Act [19]).

The safety evaluation of the proposed technical solution is performed by one of the about 20 TSOs (licensed by the regulatory body), selected in a lowest price tender by the licensee, and approved by another regulatory decision.

3 Background on Design Extension Conditions

3.1 Definition by WENRA

For the purpose of this report, the definition adopted by the Western Europe Nuclear Regulators Association (WENRA) in the 2014 Edition of the WENRA Safety Reference Levels for Existing Reactors [20] will be used as reproduced in Table 7.

Table 7: Objective of DEC in WENRA Safety Reference Levels for Existing Reactors 2014 [20]

F1.1

As part of defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety of the nuclear power plant by:

- enhancing the plant's capability to withstand more challenging events or conditions than those considered in the design basis,
- minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.

F1.2

There are two categories of DEC:

- DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;
- DEC B with postulated severe fuel damage.

The analysis shall identify reasonably practicable provisions that can be implemented for the prevention of severe accidents. Additional efforts to this end shall be implemented for spent fuel storage with the goal that a severe accident in such storage becomes extremely unlikely to occur with a high degree of confidence.

In addition to these provisions, severe accidents shall be postulated for fuel in the core and, if not extremely unlikely to occur with a high degree of confidence, for spent fuel in storage, and the analysis shall identify reasonably practicable provisions to mitigate their consequences.

Further explanation is provided in the WENRA Guidance Document Issue F [24]): *"Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions."*

The WENRA guidance document for issue F [24] explains that the definition of DEC in WENRA reference levels [20] is consistent with the definition in IAEA SSR-2/1 [25], published in 2012.

DEC are more complex and/or more severe than conditions caused by the postulated design basis accidents [24].

3.2 Historic outline of DEC definitions

For the convenience of the reader, the most important developments of the definition of DEC are outlined below.

The expression **“design extension conditions”** (DEC) has a rather long history and was possibly introduced by the European Utility Requirement before the 2001 Rev. C. [26], which defines DEC for the design of the reactors of third generation as preferred method for giving due **consideration to the complex sequences and severe accidents at the design stage without including them in the design basis conditions** [26].

In 2007, WENRA [27] recommended a “**design extension**” analysis (it is noted here that the term Design Extension Conditions has not yet been used). Perhaps one might see the rationale for **extending the design bases** in the 2007 position of WENRA [27], or perhaps, in the rationale behind the Periodic Safety Reviews asking for revisiting and updating, if necessary, the facility design bases. As discussed in section 3.1, no potential foundation for **extending design bases** can be found in the WENRA 2014 Safety Reference Levels [20].

In 2012, the IAEA SSR-2/1 Rev. 0 [25] defines: “***Accident conditions*** *that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions.*”

Council Directive 2014/87/Euratom [2] in Article 3 defines: “*11. “severe conditions” means conditions that are more severe than conditions related to design basis accidents; such conditions may be caused by multiple failures, such as the complete loss of all trains of a safety system, or by an extremely unlikely event.*”

In 2016, IAEA SSR-2/1, Rev.1 [28] design extension conditions definition is: “***Postulated accident conditions*** *that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable regulatory limits.*”

It is important to note that 2016 IAEA SSR-2/1, Rev.1 [28] definition of DEC importantly differs from the 2012 IAEA SSR-2/1 [25]. Namely, “accident conditions” have been replaced by “postulated accident conditions”, i.e. emphasis is on postulated. However, IAEA SSR-2/1, Rev.1 [28] does not attempt to specify the list of postulated accident conditions. Second major difference was to remove the following statement: “*Design extension conditions could include severe accident conditions.*”

The main purpose of the Design Extension Conditions therefore appears to be identification of those beyond design plan states, where one could anticipate successful intervention, which would either prevent development of a severe accident or mitigate its consequences, and to develop the procedures and means for such intervention. In other words, the DEC are clearly conditions beyond the design basis conditions and therefore in the realm of more (DEC A) or less (DEC B) controllable beyond design basis accidents. The purpose of DEC is therefore not to extend the plant design bases.

Since the plant states have been traditionally organized in sets (e.g., operational conditions, accident conditions, severe accident conditions…), it may be worthwhile to depict the DEC together with other plant conditions (or states) in the form of Venn's diagram (see Figure 1 and Figure 2).

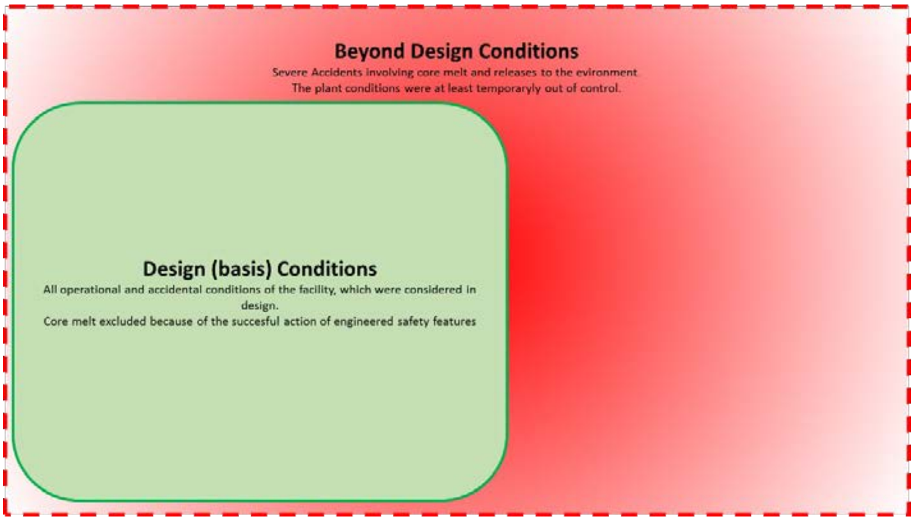

Figure 1 Simplified sketch of all plant conditions before the introduction of DECs. Note that shapes represent the sets in the sense of the Venn's diagram. The set with "Beyond Design Conditions" is an open set.

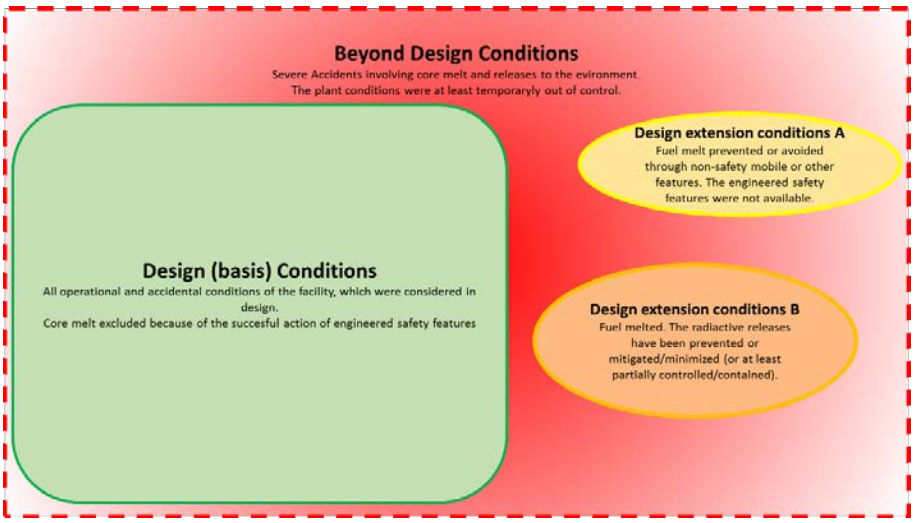

Figure 2 Simplified sketch of all plant conditions. Note that shapes represent the sets in the sense of the Venn's diagram.

Figure 1 shows the situation before the inauguration of DECs. The union of two nonintersecting sets "design (basis) conditions" and "beyond design conditions" composes the set of “all possible plant conditions”. The set with “beyond design conditions” must include the “unknown unknowns” and is therefore open.

The introduction of DEC’s is then sketched in Figure 2. Two new non-intersecting sets (DEC A and DEC B) are defined from the conditions, which were members of the set “Beyond Design Conditions” in Figure 1. In this way, neither DEC A nor DEC B conditions can affect the Design (bases) Conditions. Separation or independence of DBA, DEC A, DEC B and BDBA plant conditions also allows for separate requirements for the equipment devoted to the management of respective plant states.

3.3 What is not part of Design Extension Conditions

Valuable explanations for DEC in IAEA SSR-2/1, rev. 1 [28] are provided in IAEA TECDOC-1791 [51]:

“*In the IAEA terminology, a DEC is a postulated plant state (see Table 1) that is determined by a postulated sequence of events, and for the same reasons that design basis hazards are not considered DBAs, more severe hazards are not considered DECs although they might result in a DBA or possibly in DEC. According to SSR-2/1 […], external hazards are considered in the design by assuming appropriate loads, load combinations and margins as detailed in requirements 5.21 and 5.21a for DBA equipment, and requirements 5.21a and 5.29 for DEC equipment.*

The control of DECs is expected to be achieved primarily by features implemented in the design (safety features for DECs) and not only by accident management measures that are using equipment designed for other purposes. This means that in principle a DEC is such if its consideration in the design leads to the need of additional equipment or to an upgraded classification of lower class equipment to mitigate the DEC.”

3.4 Practice in the USA

An overview of the regulatory practice in the vendor country (USA) for the Krško NPP is given below for completeness. As further detailed below, the US practice currently does not endorse the concept of Design Extension.

Several rules have been issued to address accidents identified in Probabilistic Risk Assessments (PRAs) as having significant risks not recognized in or enveloped by the set of Design Basis Accidents (DBAs). An important example is 10 CFR 50.63, the station blackout rule. However, these events were not added to the set of DBAs, and a consistent framework was not developed to address them. As a result, a “patchwork” of agency guidance was established to address beyond-design-basis accidents. Document NUREG-2150 from 2012 [29] recommends: “The NRC should establish through rulemaking a design-enhancement category of regulatory treatment for beyond-design-basis accidents. This category should use risk as a safety measure, be performance-based (including the provision for periodic updates), include consideration of costs, and be implemented on a site-specific basis.”

It should be noted that Risk Management Task Force (RMTF) has chosen to use the term “design-enhancement” rather than “design extension” [29].

The NRC staff proposed in document SECY-13-0132 from 2013 [30] adoption of a new term—“design-basis extension” for events that are not currently considered to be design-basis events or accidents, but that must be regulated because their prevention and/or mitigation is necessary for reasonable assurance of adequate protection or should be regulated because their prevention and/or mitigation would result in a substantial safety improvement at a cost that is justified in view of the increased protection.

However, SECY-15-0168 from 2016 [31] for Design-Basis Extension Category concludes that a new category of events should not be established [31]. The NRC decided to use existing resources to develop and implement internal rulemaking guidance to ensure that all future nuclear power reactor regulations (especially those imposing beyond-design-basis requirements) include consistent and comprehensive rule language addressing all necessary regulatory attributes (i.e. performance goals, treatment requirements, documentation, requirements, change processes, and reporting requirements).

Activity running in parallel was related to lessons learned from Fukushima Dai-ichi accident. In 2011 a task force of senior US NRC staff made recommendations based on its evaluations of the relevant issues identified from the Fukushima Dai-ichi accident. The regulatory efforts to address lessons learned from Fukushima have evolved over time, and resulted in the Mitigation of Beyond Design Basis Events (MBDBE) rulemaking in 2016. The proposed rule “Mitigation of Beyond Design Basis Events (MBDBE)” [32] would enhance mitigation strategies for nuclear power reactors for beyond-design-basis external events. Draft Final Rule 10 CFR 50.155, “Mitigation of Beyond-Design-Basis Events” is available [33]. The corresponding changes establish requirements for applicants and licensees to develop, implement and maintain strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that result in an extended loss of alternating current power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. The final rule is effective on September 9, 2019 [52].

In parallel the US industry developed the proposal for FLEX strategy NEI 12-06, Rev. 1A [33] for fulfilling the key safety functions of core cooling, containment integrity, and spent fuel cooling. The US NRC proposed Regulatory Guide 1.226 [34] endorsing the FLEX methods [35]. Additionally, the Regulatory Guide 1.226 [34] provides guidance in areas that are not covered in NEI 12-06, for meeting the regulations in 10 CFR 50.155.

4 Legal and technical implementation in Slovenia

The bottom-up approach with the regulations following the plant upgrades proposed by the license has been the major driver for the implementation of the post Fukushima plant upgrades and therefore also the implementation of the Council Directive 2014/87/Euratom [2] in the national legislation. This makes it more convenient for the reader to describe the technical implementation before the legal implementation.

The discussion in this chapter is guided in the level of details, organization and form by the publicly available documents, published by the Slovenian licensee and regulator.

4.1 Technical Implementation

The technical implementation of the Krško NPP safety upgrades, associated with the European Stress tests [17] and the Council Directive 2014/87/Euratom [2], is summarized in the 2017 Update of the Slovenian Post-Fukushima Action Plan [36]. The following main sets of actions are identified:

1. The already implemented short-term improvements
   a) Accelerated B.5.b requirements' actions
   b) Implementation of Slovenian Stress test action plan
2. Safety Upgrade Program (SUP) of Krško NPP
3. Additional long-term improvements/activities

The majority of the DEC related upgrades are enclosed in the Safety Upgrade Program (SUP) of Krško NPP, which is described in more detail in section 4.1.1.

The most of the actions in the SUP have been conceived or conceptualized before the Stress tests in 2011, WENRA definition of DEC in 2014 [20] or the Council Directive 2014/87/Euratom [2]. The conceptualization of actions was done within the context of many other drivers for safety upgrades, including:

- the 1st Periodic Safety Review (2002, [38], [39]);
- the 2nd Periodic Safety Review (2010, [40]);
- the US NRC B.5.b requirements [41], initiated by the 9/11 events, for reasonable measures to assure that available resources are used effectively in responding to beyond design basis threats. The objective of these site-specific assessments was to utilize a threat-independent methodology to identify potential plant specific strategies for preventing or mitigating damage to the fuel;
- the post Fukushima stress tests [42], [43], which were interpreted as an extraordinary periodic safety review by the Slovenian nuclear regulations.

Some generic comments on this context are given below. More specific references to possibly interlinked connections between different drivers for safety upgrades are made, as appropriate, in section 4.1.1.

Krško NPP accelerated implementation of the US NRC B.5.b requirements after the Fukushima Dai-ichi accident and completed them in June 2011. This improved the stress tests [17] assessment of the Krško NPP resilience for coping with the beyond design bases accidents such as loss of all AC power combined with loss of ultimate heat sink or failure of equipment due to natural phenomena exceeding the design bases.

In 2011 Krško NPP has been already commissioning safety upgrades required by the action plan following the 1st periodic safety review (2002, [38]). In particular, these included:

- installation of the third independent diesel generator with a safety bus. The 3rd diesel generator was conceived to serve either as the 3rd safety related emergency diesel generator with provisions to connect to either of the two original plant safety buses, or as an alternative alternate current (AAC) power source
- a mobile diesel generator of capacity 2000 kVA with provisions to be connected to the switchgear of the 3rd diesel generator, and
- flood protection upgrade (strengthening the dikes).

These actions were completed in 2012.

4.1.1 Development of the Krško Safety Upgrade Program

In September 2011, the SNSA issued Decision [44] requesting the Krško NPP to perform a review of already available results, as well as additional PSA and deterministic calculations to assess overall plant vulnerability to low probability extreme external events (earthquake, flood, fire, aircraft accident) and their combinations. Decision [44] also requires reassessment of the severe accident management strategy, existing design measures and procedures, undertake a study of the response of the nuclear power plant to severe accidents and, based on the findings of this study, prepare a program for safety upgrade of existing structures, systems and components (SSCs) and other measures and new systems that are important to provide nuclear safety during severe accidents, with special emphasis on improvements aiming for improved reliability of:

- AC power supply from external and internal sources,
- reactor core cooling through primary (water injection to primary system) and secondary system (primary system/core cooling through the steam generators),
- containment integrity at high temperature conditions, overpressure and high hydrogen concentrations,
- controlled releases from the plant to the environment (< 0.1 % of aerosols and particulates from core fission products),
- core cooling and control during severe accidents from the alternative control room and,
- alternative cooling of spent fuel pool.

The time schedule and implementation plan for the proposed safety upgrades was also requested in [44]. The deadline for the implementation of the upgrades planned in SUP was set to 31.12.2016.

In response, the Krško NPP prepared extended design bases (EDB) Safety Upgrade Action Plan [42], which included a review of all site-specific natural and man-made accident initiators to verify current plant licensing design bases, followed by the development of the extended design bases requirements. Additional safety systems were proposed using defense-in-depth gap analysis and results of PSA analyses of extended plant models. The SUP was approved by the SNSA Decision [45] in January 2012.

It is clear that the Safety Upgrade Action Plan [42] has been developed in parallel with the IAEA SSR-2/1, rev. 0 [25] and before the 2014 WENRA Reference levels and related changes in the Slovenian national legislations. This was also recognized by the SNSA in the 2016 National report on the IAEA nuclear safety convention [46]:

"*On its own initiative and based on various industry issues the Krško NPP initiated some safety improvement projects. The Safety Upgrade Program aim is to improve plant safety against extreme external hazards and to increase plant capabilities for prevention or mitigation of severe accidents. In the year 2013 the Krško NPP installed Passive Containment Filtered Venting System (PCFVS) and Passive Autocatalytic Recombiners (PAR) in the containment. In the year 2015 the flood protection of safety important plant buildings against extreme flooding was implemented*."

Further revisions of the Krško Safety Upgrade Plan were therefore needed and were approved by the SNSA Decisions in 2013 [47] and 2017 [48]. The 2017 Decision [48] extended the deadline for the implementation of the upgrades until 31.12.2021. The reasons for the extended deadlines have been reported by the regulator in the Update of the Slovenian Post-Fukushima Action Plan (NAcP) for 2017 [36]:

"*In September 2013, the Krško NPP applied for the extension of the final SUP deadline. The main reasons for the delay were the magnitude of the project, complexity of design documentation, delivery times of some of the main components, as well as inclusion of the Krško NPP into the Public Procurement in Water Management, Energy, Transport and Postal Services Area Act, which further complicated, delayed, and finally failed the bidding of the project. The SNSA approved the extension of the deadline until the end of 2018.*

Then, in the beginning of 2014 the Krško NPP notified the SNSA that the implementation of the SUP until the end of 2018 is going to be challenged due to financial constraints. Namely, the two owners of the Krško NPP (the Slovenia's state owned GEN Energija d.o.o. and the Croatia's stated owned HEP d.d.) became unwilling to finance the SUP (especially the larger part of it, the "BB2 project") due to doubts that the plant could, after the implementation of the project, still continue to provide electricity at a competitive price. The owners ordered the financial viability study, after which they would decide about the continuation of the "BB2 *project". The result was in favor of the SUP implementation and life time extension of the Krško NPP, thus the owners decided to continue with the implementation of SUP improvements. Due to the delay of the implementation of the BB2 project, the Krško NPP again applied for the extension of the deadline for the third SUP phase (the BB2 project), and also some conceptual changes of the SUP. The major change is the revision of alternative ultimate heat sink, which will be assured with the use of steam generators fed by additional dedicated sources of water capable of replenishing from underground wells. This way the cooling of the reactor will be assured for at least 30 days even with the complete loss of the existing UHS. The SNSA reviewed the revision of the SUP and supporting analyses and in the beginning of 2017 approved the new SUP program and the extension of the third SUP phase's deadline until the end of 2021*."

4.1.2 Selection of Design Extension Conditions for Krško

The most recent publicly available document summarizing the Krško Safety Upgrade Plan is the Update of the Slovenian Post-Fukushima Action Plan (NAcP) for 2017 [36], published by the SNSA.

The relationship with the Design Extension Conditions is described in the 2017 NAcP [36] as:

"Additional systems, structures and components, which will be implemented within the SUP, will be designed and structured in accordance with the design extension conditions (DEC) requirements specific for the Krško NPP design and site location."

"A set of DEC is derived on the basis of engineering judgment, deterministic assessment and probabilistic assessment based on the IAEA methodology defined in SSR-2/1, Safety of Nuclear Power Plants: Design Specific Safety Requirements […], Krško NPP's Individual Plan Examination and the Krško NPP Analyses of Potential Safety Improvements […]".

The definition of the Design Extension Conditions in the 2017 NAcP [36] follows as:

- ***"earthquake****, extended design condition seismic value is 2xSSE (0.6 g PGA)";*
- ***"flooding****, new maximum flood level is 157.53 m above sea level. The preceding flood protection dikes were at 157.10 m";*
- ***"earthquake + flooding****, flood due to dikes damaged by earthquake with the river flow at current maximum PMF flow";*
- ***"earthquake + fire****, fire caused by DEC earthquake";*
- *"external low and high temperatures, air temperatures with a return period of 10,000 years",*
- ***"aircraft crash accident****, crash of large commercial aircraft at the maximum landing velocity",*
- ***"fire****, fire due to DEC aircraft crash".*

All other combinations of events/accidents are considered as Beyond Design Basis Accidents (BDBA) and will be addressed by mobile equipment. Severe accident management guidelines are in place.

The above mentioned Design Extension Conditions are assumed to persist for the following amount of time [36]:

- *“loss of off-site power (LOOP) for 7 days”,*
- *“station black-out (SBO) for 72 hours”,*
- *“loss of ultimate heat sink (UHS) for 30 days”,*
- *“loss of UHS combined with SBO for 72 hours”,*
- *“flooding water (from Sava river) retains for 7 days”.*

DEC systems, structures and components will be located in two new bunkered buildings.

The new DEC equipment can be separated into the prevention and mitigation (DEC B) part. The prevention part of the equipment serves to preserve adequate fuel cooling in case of DEC events, taking into account the duration of these events specified above.

Please note here, that although the prevention and mitigation parts of DEC clearly correspond to DEC-A and DEC-B, respectively, the notions of DEC-A and DEC-B were not yet used in the 2017 NAcP [36].

For the mitigation part, it is assumed that preventive DEC equipment will not be available for 24 hours and that the core will melt and corium relocate into containment. This is the basic assumption for DEC containment filtered vent system and passive autocatalytic recombiners. This assumption also led to the requirement that batteries for DEC equipment and emergency control room shall have a 24 hour capacity.

4.1.3 Status of the Krško Safety Upgrade Program

The Krško Safety Upgrade program is being commissioned in three phases. The main safety upgrades are, together with the main functional descriptions, summarized for each of the three phases in Table 8, Table 9 and Table 10, respectively.

Table 8: Equipment planned in the 1st phase of SUP (adapted after NAcP [36])

Equipment	Description	Purpose	Scheduled finish
Installation of passive autocatalytic recombiners in the containment	Replacement of electric DBA recombiners with passive BDBA auto-catalytic recombiners in the containment. The two electric DBA recombiners have been replaced with two DEC PARs. Additional DEC PARs were installed into different containment compartments for managing severe accidents hydrogen.	Mitigation	2013 implemented
Filtered venting system	Filtered venting system capable of depressurizing containment and filtering over 99.9% of volatile fission products and particulates (not including noble gasses).	Mitigation	2013 implemented

The history of the Core Damage Frequencies (CDF) for the Krško NPP is depicted in Figure 3 A history of the Core Damage Frequency for Krško NPP [37]Figure 3. The upgrades completed in years 2013 and later brought very modest decreases in the CDF, which is in part consistent with the nature of the traditional PSA models (focus on safety systems) and the fact that the new DEC systems do not have to match the redundancy and diversity of the DBA safety systems.

Table 9: Equipment planned in the 2nd phase of SUP (adapted after NAcP [36])

Equipment	Description	Purpose	Scheduled finish
Additional flood protection of nuclear island and newly installed equipment	Nuclear island and the above-described newly installed equipment will be additionally flood protected against the failure of flood protection dikes or high river flows exceeding flood protection dikes by up to 0.4 m.	prevention and mitigation	2016 implemented
Emergency operating facilities	Establishment of new technical support center and upgrade of existing operational support center (emergency operating facilities).	prevention and mitigation	2018
Installation of additional pressurizer PORVs	Additional pressurizer PORVs will be installed, qualified for DEC events	prevention and mitigation	2018 implemented
Installation of permanent sprays around the SFP	Installation of permanent sprays (2×SSE qualified) around the SFP with provisions for quick connection of mobile equipment and different sources of water. Spraying of SFP is needed in case of loss of SFP integrity.	prevention	2016
Mobile HX	Mobile heat exchanger with provisions to quick connect to SFP.	prevention	2018
Establishment of emergency control room	Relocation and expansion of existing remote shutdown panels into a new emergency control room in the separate bunkered (2×SSE and PMF flood protected) building with all I&C needed for safe shutdown of the plant and maintaining the safe shutdown conditions.	prevention and mitigation	2019
Long term habitability of emergency control room and support staff facility	The above-mentioned emergency control room will enable long term habitability of control room staff even during severe accidents (air filtering, radiation protection). For the same conditions also new facility for supporting staff will be designed and build.	prevention and mitigation	2018
Installation of separate dedicated BDBA I&C	Installation of separate dedicated BDBA I&C capable of monitoring and controlling both from the existing as well as the new emergency control room.	prevention and mitigation	2018
Safety upgrade of AC supply	Within this action several modifications/upgrades will be performed on the AC power supply, including modification of alternative supply of non-safety related buses, requalification of 3rd 6.3 kV safety related bus (MD3), upgrade of connection between 400 V safety related bus (for charging batteries) and mobile diesel generators,…	prevention	2015

Table 10: Equipment planned in the 3rd phase of SUP (adapted after NAcP [36])

Equipment	Description	Purpose	Scheduled finish
Alternative ultimate heat sink	Alternative ultimate heat sink will be assured for at least 30 days with the use dedicated pool of water in the bunkered building with replenishment from underground wells.	prevention	2021
Alternate AF pump	Additional high pressure pump for feeding SGs in the separated bunkered (2×SSE and PMF flood protected) building with dedicated source of water for 8 hours with provisions to refill by mobile equipment from different water sources.	prevention	2021
Dry spent fuel storage facility	The results showed that best alternative spent fuel strategy would be storing the spent fuel in dry cask storage with a possibility to combine it with later reprocessing. Dry cask storage with a possibility to combine it with later reprocessing.		2021
Alternate SI pump	Additional pump for injecting into the reactor primary system, in a bunkered building, with a dedicated (borated) water supply.	prevention and mitigation	2021

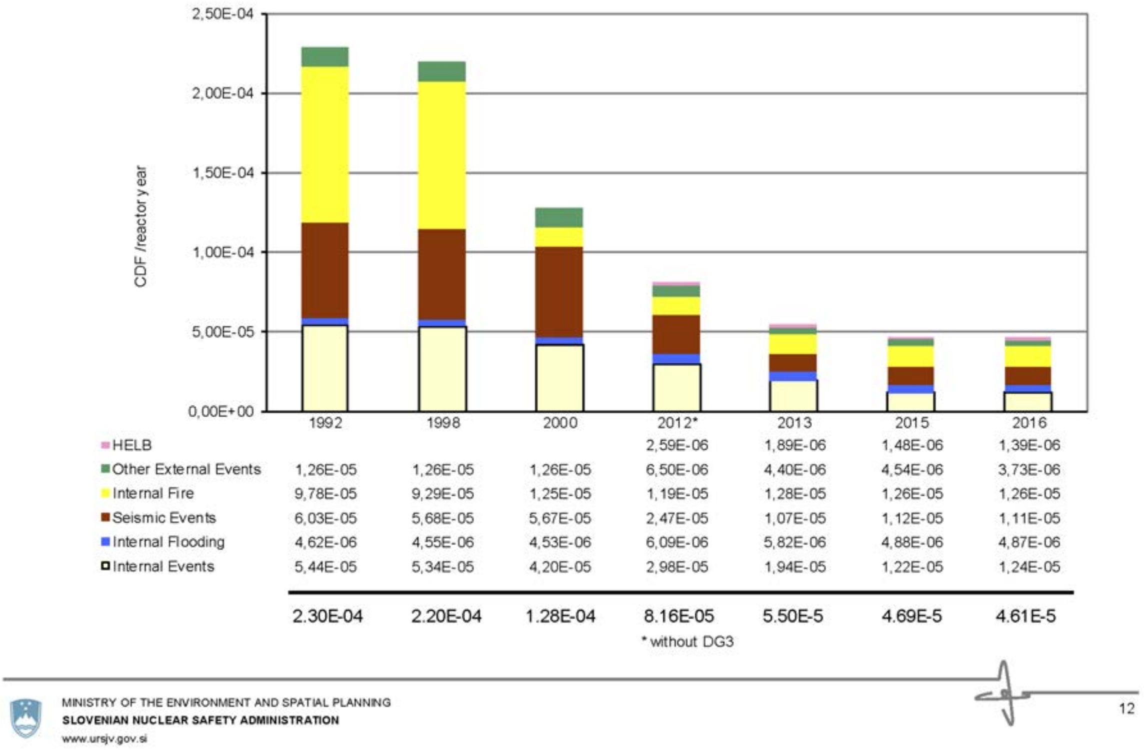

Figure 3 A history of the Core Damage Frequency for Krško NPP [37]

4.2 Legal Implementation

The changes of the national legislation with respect to the Council Directive 2014/87/Euratom [2] were completed in 2016. This is reported in page 17 of the 2017 Update of the Slovenian Post-Fukushima Action Plan [36] as:

"The SNSA finished amending / revising its legislation based on the above stated commitments and/or considerations. The two amended rules, Rules on radiation and nuclear safety factors – JV5 and Rules on operational safety of radiation and nuclear facilities – JV9, which incorporate the latest WENRA SRL updates (adopted in September 2014), were adopted in December 2016."

More details are given below for the convenience of the reader.

4.2.1 Article 8a

Article 8a of the Council Directive 2014/87/Euratom [2] is for the convenience of the reader reproduced in Table 1.

The 1st paragraph of the Article 8a is directly translated into the Article 4 (chapter on the basic principles) of the 2017 edition of the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19]. The definition of the **early radioactive releases** is slightly modified and would in an unofficial translation read "*early radioactive releases that would require quick off-site protective measures*". Similarly, the definition of the **large radioactive releases** seems to be slightly modified to read "*large radioactive releases that would affect large areas and persist for a long time*".

The following probabilistic limits for design basis are defined in part 1.1. of Appendix 1 to the Rule JV5 (Rules on radiation and nuclear safety factors, [21]):

- the frequency of the core melt is lower than 10^{-5}/year (lower than 10^{-4}/year during power operation for Krško NPP);
- the frequency of early or large releases is lower than 10^{-6}/year (lower than $5 \cdot 10^{-6}$/year for large uncontrolled releases during power operation for Krško NPP).

Part (a) of the 2nd paragraph is fulfilled as the (ZVISJV-1) [19] is valid for any (new or old) nuclear facility. The existing Krško NPP unit has been granted certain exceptions as compared to potential new builds, as detailed in the Rule JV5 [21]:

(1) The minimum time before the first credited operation action is defined as 30 minutes and 15 minutes for Krško NPP (Article 16 of JV5).
(2) All three levels of PSA analysis are required (Article 16 of JV 5). The Krško NPP is required to develop at least Levels 1 and 2.
(3) Seismic design must account for the seismic loadings estimated at the site with the frequency of 10^{-4}/year. For less frequent loadings, a lower limit of horizontal acceleration of 0.1g should be used. Krško NPP may fulfill the requirements through the estimated seismic fragilities and related protection measures.
(4) Functionality of the containment should be maintained in the case of the crash of a large commercial aircraft. Krško NPP must show that all

reasonable measures have been implemented to mitigate the crash of the large commercial aircraft.

Part (b) of the 2nd paragraph is implemented in the Article 75 of the Rule JV5 [21], which requires:

(1) Krško NPP should fulfill the following requirements before December 31, 2018:
   a. Establish an Emergency Control room, which is physically, electrically and functionally separated from the Main Control Room. The Emergency control room should enable monitoring and control of the following: achieve and maintain the reactor in safe shutdown, remove the residual heat from the reactor and the spent fuel storage, monitor essential parameters of the plant including those of the spent fuel storage.

(2) Krško NPP should fulfill the following requirements before December 31, 2021 (4 and 5 in Appendix 1 of JV5):
   a. Extended design bases. Define, perform the safety analyses, ensure safety functions and management of severe accidents within the extended design bases, periodically review. The definition of the "extended design bases" is given by the Ionising Radiation Protection and Nuclear Safety Act (ZVISJV-1) [19] and is further detailed in section 4.2.4).
   b. Natural hazards: define and assess those relevant for the site, define design basis events, define protection against design basis events, define the extended design events and update the extended design bases.

In addition, Appendix 1 of JV 5 requires in paragraph 4 of chapter 1.1 that the plant design must practically exclude severe accidents or render them practically impossible. For those core melt accidents, which cannot be excluded, practical solutions must be in place to protect the public with the very basic protection measures (e.g., excluding the need for permanent relocation, evacuation from the vicinity of the plant, limited shelter or long-term food restrictions). Also, there must be enough time to implement the basic protection measures. Krško NPP uses this requirement as a reference for the timely implementation of practicable safety improvements in the framework of periodic safety reviews.

4.2.2 Article 8b

Article 8b of the Council Directive 2014/87/Euratom [2] is for the convenience of the reader reproduced in Table 2.

The main changes in the national legislation have been triggered by the parts (a) and (e) of the 1st paragraph of Article 8b. These changes are related to the management of beyond design basis plant conditions and is further detailed in the section 4.2.4 Definition of Design Extension Conditions below.

It is noted that requirement "arrangements of education and training" (Part (d) of the 2nd paragraph of Article 8b) is delegated to the employer(s) (part (3) of Article 92 in ZVISJV-1 [19]).

4.2.3 Article 8c

Article 8c of the Council Directive 2014/87/Euratom [2] is for the convenience of the reader reproduced in Table 3.

Site and facility specific license (as defined by the US 10CFR50) and Periodic Safety Review every ten years were part of the Slovenian legislation since 1984 and 2002, respectively.

4.2.4 Definition of Design Extension Conditions

The concept of Design Extension Conditions exists in the Slovenian nuclear regulation in an implicit manner. An associated concept of “razširjene projektne osnove”, in free translation “**extended design bases**”, has been introduced in the 2014 edition of the ZVISJV-D [49] (now superseded by the 2017 edition of ZVISJV-1 [19]). As already mentioned, the practice of publishing official English translations of the Slovenian nuclear legislation has been temporarily discontinued with the latest few editions of the Act and Rules and has been resumed in June 2018, when the majority of conceptual and design work for the relevant plant modifications has already been completed.

The translations in this section are therefore unofficial and have been developed by the authors of the report for the convenience of the readers and to shed some light on the regulations in Slovenian language, which represented the legal bases for the conceptual and design work for the relevant plant modifications.

Extended design bases as defined in ZVISJV-1 [19] shall ensure robustness of the facility to reduce the probability of the radioactive releases in the case of extended design basis events to negligible amount. One should mention here that the purpose of DEC B in WENRA [20] is to “minimize the radioactive releases harmful to the public” rather than to aim at “negligible amount of radioactive releases”, which seems to be the intention of the ZVISJV-D [49].

The concept of “extended design bases” is then further elaborated in the Chapter 4 of Annex 1 of rule JV5 [21], which also introduces and defines the terms “Extended design bases of category A” and “Extended design bases of category B”. The definitions of these two terms are fairly consistent with DEC A and DEC B as defined in WENRA RL, Issue F [24]. Further details on the definitions of extended design bases and related concepts in JV5 [21] are given in Table 11 and Table 12.

The requirements of the JV5 [21] with respect to the “extended design bases” are more demanding that the requirements of the WENRA RL for the “design extension (of existing reactors)”. Indeed, the conditions being part of the extended design bases can not at the same time belong to the design extension conditions, which are defined as beyond design conditions (see section 3). Further, this is also consistent with the apparent linguistic explanation of differences between the extended design conditions/bases and design extension conditions/bases, namely: (extended design = design + design extension).

Table 11: Definitions related to DEC in Slovenian regulations JV5 [21].

Origin	Definition
extended design bases (razširjene projektne osnove) – design extension conditions	
ZVISJV-1, Article 3, item 64	Razširjene projektne osnove objekta opredeljujejo njegovo zmogljivost za preprečevanje nesprejemljivih radioloških posledic zaradi nesreč, težjih od tistih dogodkov, ki so podlaga za projektne osnove, ali ki vključujejo več odpovedi, kakor so predpostavljene pri projektnih osnovah. Razširjene projektne osnove je treba pripraviti na podlagi inženirske ocene ter z determinističnimi in verjetnostnimi metodami z namenom, da se prepoznajo dodatni scenariji nesreč in načrtujejo praktične rešitve za njihovo preprečitev ali blaženje njihovih posledic.
Translation by SNSA	Extended design bases of facility define the facility's capacity to prevent unacceptable radiological consequences due to accidents, more serious than situations which are the basis of design bases, or which also include terminations as presumed in the design bases. The extended design bases are prepared based on an engineering assessment and the deterministic and probability methods with the intention to identify additional scenarios of disasters and to plan practical solutions for their preventions or for mitigating their consequences.
Translation by JSI	Extended design bases of nuclear facility define it's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than events considered in design bases or that involve more failures than considered in design bases. The extended design bases shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments with the purpose 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.
WENRA	As part of defence in depth, analysis of Design Extension Conditions (DEC) shall be undertaken with the purpose of further improving the safety of the nuclear power plant by (1) enhancing the plant's capability to withstand more challenging events or conditions than those considered in the design basis, (2) minimising radioactive releases harmful to the public and the environment as far as reasonably practicable, in such events or conditions.
extended design accidents (razširjene projektne nesreče) – design extension conditions	
JV5, Art. 2, item 45	Razširjena projektna nesreča je nesreča, ki jo povzročijo razširjeni projektni dogodki.
Translation by JSI	Extended design accident is an accident, which is caused by extended design events.
WENRA	Not explicitly defined in Issue F, but WENRA Design Extension Conditions are consistent with the IAEA SSR-2/1 rev. 0 [25]: "Accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions could include severe accident conditions."
extended design event (razširjeni projektni dogodek)	
JV5, Article 2, item 46	Razširjeni projektni dogodek je dogodek ali kombinacija dogodkov z izredno majhno verjetnostjo in težjimi posledicami od projektnih dogodkov oziroma vključuje več odpovedi, kot so predpostavljene pri projektnih osnovah jedrskega objekta.
Translation by JSI	Extended design event is an event or combination of very low probability events with consequences more severe than those of design basis events, or that involve more failures than considered in the design bases of nuclear facility.
WENRA	Not Applicable

* for the existing reactors only, while for new reactors several such scenarios (i.e. the postulated scenarios) are part of "design basis"

It is however noted here that if one assumes (following SNSA Regulatory Guide [23] and document [50]) that the "extended design bases" is an alternative expression for the "design extension conditions", then the JV5 [21] can be seen as a reasonable implementation of the WENRA RL for Issue F [20].

Similar position has also been taken by the SNSA in the regulatory guide PS 1.06 [23], which defines "design extension conditions" as equivalent to "extended design bases" ("razširjene projektne osnove"). English terms in PS 1.06 [23] for plant states are "Design Extension Condition A" and "Design Extension Conditions B". English terms in PS 1.06 [23] for Initiating events are "Design Extension Conditions A" and "Design Extension Conditions B". Finally, English term in PS 1.06 [23] for Design Conditions is "Design Extension Conditions". From this example it is seen that English term "Design Extension Conditions" is used for the following three purposes: as plant state, as initiating event and as design condition (there are two: design basis and design extension condition). It is not so in Slovenian, they have each their own expression, which can be distinguished: "Razširjena projektna nesreča kategorije A", "Razširjeni projektni dogodek kategorije A" and ""Razširjene projektne osnove".

Table 12: Definitions related to DEC A and B in Slovenian regulations JV5 [21].

Origin	Definition
Extended design accident of category A, B (razširjena projektna nesreča kategorije A, B)	
JV5, Article 2, items 45, 46	"45. razširjena projektna nesreča je nesreča, ki jo povzročijo razširjeni projektni dogodki. Obsega razširjene projektne nesreče kategorije A in kategorije B;" "Obstajata dve kategoriji razširjenih projektnih dogodkov: - razširjeni projektni dogodki kategorije A, pri katerih se lahko zagotovi preprečitev poškodbe goriva v reaktorju ali skladišču z izrabljenim gorivom; - razširjeni projektni dogodki kategorije B, za katere se predvideva težka poškodba goriva, ki presega projektno poškodbo goriva;"
Translation by SNSA	There are two categories of design extension conditions events: - Design extension conditions category A, for which prevention of severe fuel damage in the reactor or spent fuel storage can be achieved, - Design extension conditions category B with postulated severe fuel damage, exceeding the design basis fuel damage;
Translation by JSI	"45. Design extension accident is an accident, which is caused by extended design event. It includes extended design accidents of category A and B." "There are two categories of extended design events: – extended design event of category A for which prevention of severe fuel damage in the reactor or in the spent fuel storage can be achieved; – extended design event of category B with postulated severe fuel damage, exceeding the design basis fuel damage."
WENRA (F1.2)	There are two categories of DEC: – DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved; – DEC B with postulated severe fuel damage.

In the 2016 National report on the IAEA nuclear safety convention [46] about the implementation of the 2014 WENRA reference levels [20] for the Slovenian JV5 rule [21] is stated:

"*The amended Rules JV5 stipulate (in line with the WENRA reference levels and requirements for new designs) that the accidents with core melt, which would lead* *into early or large releases, shall be practically eliminated, meaning that these kind of accidents shall be almost impossible by design. Yet for accidents that cannot be practically eliminated, solutions shall be in place to assure 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).*"

4.2.5 Selection of Design Extension Conditions

WENRA, issue F2 [20] defines the requirements for "selection of design extension conditions". The implementation in JV5, Annex 1, Section 4.1 requires "selection of extended design bases" ("izbira razširjenih projektnih osnov").

Further, the WENRA selection process for DEC A (F2.2) is very similar to JV5 "selection of extended design based of category A" (Annex 1, Ch. 4.1) with the following notable differences:

- WENRA requires "**defined** operational states of the plant" while JV5 requires "**all possible** plant operational conditions"; and
- WENRA requires "events resulting from internal or external **hazards**" while JV5 requires "events resulting from internal or external **postulated initiating events**".

JV5, Article 2, item 1, requires deterministic analyses of sequences starting with postulated initiating events ("predpostavljenih začetnih dogodkov"). JV5 defines the postulated initiating events ("predpostavljeni začetni dogodek") as events, which are part of **design basis** and can actuate anticipated operational occurrences ("pričakovani obratovalni dogodek") or accidents ("nesrečo"). In this way, the most severe accidents to be analyzed would never be beyond design basis accidents. This renders the current JV5 requiring definition of "postulated initiating events" (instead of "hazards" by WENRA) potentially significantly different. JV5 by such definition is therefore limited to design basis accidents, while WENRA considers DEC A. However, when looking closely to the SNSA translation of JV5 Attachment 1, section 4, item 2, the SNSA translation in document [50] returns WENRA F2.2 text, with the note that "DEC" is consistently translated to "design extension conditions".

4.2.6 Analyses of Design Extension Conditions

JV5, Annex 1, Chapter 4.2 deals with **safety analyses** of extended design bases ("varnostne analize razširjenih projektnih osnov"). WENRA F2.3 request safety analysis of design extension conditions (DEC). Again, this dissimilarity appears minor with the assumption of equivalence between the JV5 "extended design bases" and WENRA "design extension conditions".

5 Discussion

The Slovenian safety upgrade program outlined in section 4 above has been mostly driven by the licensee, who has been proposing the concepts, technical solutions and the schedules for the implementation.

The SNSA supported these activities by approvals of those concepts, technical solutions and the implementation schedules. The legal implementation of the related concepts and rules has been made mostly in parallel and in part also after the approvals of the technical solutions proposed by the licensee. This was consistent with developments in the international regulatory (WENRA) or regulatory guidance (IAEA) communities.

Some proposals for good practices, opportunities, challenges and gaps were noted in this approach and are discussed below. The main intent is to stimulate the discussion, which could lead with further analyses to the proposal of optimal solutions.

Most of the potential challenges discussed below could be avoided or mitigated by a stronger involvement of TSOs already in the conceptualization phase for both the technical solutions and regulations.

5.1 Potential best practices and opportunities

- Licensee driven safety upgrade process is fully consistent with the primary responsibility for the nuclear safety. It can be seen as one of the best approaches available, especially if fully supported in real time by the regulators and TSO. A challenge could be timely mobilization of sufficient resources at all relevant stakeholders (licensee, regulator, TSO, policy makers).
- Periodic safety reviews seems to be efficient way to identify the week points in technology. Perhaps it would be useful to search for similar opportunities to develop comparable reviews for the rest of the man-technology-organization triad.

5.2 Potential Challenges

The concept of Design Extension Conditions (DEC).

- DEC is a relatively complex concept, which brings rather complex changes to the relatively simple landscape of “design” and “beyond design” (compare Figure 1 and Figure 2). The introduction of DEC brought a complex change to the realm of the regulations.
- Complex changes should be given sufficient attention, time and resources. In other words, complex changes could be (only?) justified, if they would bring substantial added value. Please compare the US decision to abandon DEC and to focus (the limited resources available) on the improvements of the relatively simple “design” – “beyond design” regulatory framework (section 3.4).

- In depth analysis of safety benefits would be most beneficial before the changes in the legislation. Inadequate and/or delayed implementation could bring much less benefits than anticipated. The concept of “Extended design bases” could for example easily result into the routine updating of design bases, which could eventually prevent any facility to operate more than a decade or two.
- The complexity of the DEC approach could be overwhelming for the professionals and is hard to be explained to the general public.
- An additional challenge could be to safely combine different safety philosophies, as for example the safety philosophy of the vendor's country regulation (US NRC, explicitly rejects the DEC) and the WENRA philosophy. The challenge might grow larger if the philosophies followed start focusing on the conditions with substantially different probabilities and/or consequences.
- Different interpretation of DEC by IAEA and WENRA (more severe than design basis hazards are not considered DEC by IAEA [51], while WENRA include on the list of DEC initiating events induced by earthquake, flood or other natural hazards exceeding the design basis events).

What and how to implement in national and what to be kept in the international legislation:

- Legislation only available in the national language (case of Slovenia) could bring a potential challenge for international vendors, especially in the cases with very specific and/or complex regulatory requirements and small number of affected facilities. A possible example could be the national requirements based on the “extended design bases” with vendors mostly working in the regulatory systems based “design extension conditions” or “design enhancement”.
- Implementation of rather generic requirements in the national regulations through translations and adaptations might bring additional complexity and uncertainties. Small nuclear countries might be at larger risk here. A possible solution may be direct endorsement of (selected) international or vendor’s country requirements.
- Parallel implementation of technical solutions and regulatory requirements, and especially the definition of the regulatory requirements after the technical solution and their parameters have been selected and contracted, might increase the risk of the licensee that the commissioned solution will not be licensed. If perceived as such by the licensee, the motivation of the licensee for proactive and bottom up approach to the safety upgrades might be reduced.
- Regulatory driven safety upgrades require substantial human resources for preparing well defined and easy to understand regulation. Regulators in countries with less nuclear facilities could be more affected. A possible solution may be critical endorsement of international requirements that requires profound understanding of the basis of applied requirements.

5.3 Potential Gaps

- The concept of DEC focuses mostly on the improvements/upgrades of technology. Since the safety depends on the interaction between man-technology-organization, the focus on the technology only might represent a substantial gap.

6 Conclusions

The Council Directive 2014/87/Euratom of 8 July 2014 introduced (1) the nuclear safety objective, (2) requirements for the implementation of the nuclear safety objective, and (3) requirements for the initial assessment / periodic safety reviews.

The report documents the legal and technical implementation of plant safety upgrades in Slovenia after the implementation of the Council Directive 2014/87/Euratom. The legislation and rules are in place, the regulator has confirmed the plan of safety upgrades in the country's only nuclear power plant at Krško and the upgrades of the nuclear power plant are being commissioned. Currently, the safety upgrades are estimated to be completed beyond 2/3.

The focus is on Design Extension Conditions (DEC) in the existing plant at Krško. The Design Extension Conditions have been chosen for this report because they were conceived to provide **reasonably practicable** approach to **avoid**, **prevent** and/or **mitigate** the beyond design basis accidents and their consequences, especially the **early** and/or **large radioactive releases.**

Potential good practices, challenges and gaps are identified in the report to stimulate the discussion facilitating the Member States in achieving a consistent practical implementation of the provisions set out in the Council Directive 2014/87/Euratom of 8 July 2014.

Licensee driven safety upgrades and periodic safety reviews are identified as potentially good practices. Most of the challenges are related to the complexity of the Design Extension Conditions concept, the balance between the national and international regulations and the possible role of TSOs in the conceptualization and implementation of new safety regulations and technical solutions.

The primary focus of the Design Extension Conditions on the technology part of the man-technology-organization triad, which controls the safety of complex technologies including nuclear power, is identified as a potential gap.

7 References

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PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

REPORT ON DETAILED STUDY ON SAFETY UPGRADES IN FRANCE

Revision R1c

October 2018

EC Contract No. ENER/17/NUCL/S12.769200

Table of contents

1 BACKGROUND

After the nuclear accident at the Fukushima Dai-ichi nuclear power plant in 2011, several activities have been started to further strengthen nuclear safety worldwide. In Europe, Council Directive 2009/71/Euratom [1] establishing a Community framework for the nuclear safety of nuclear installations was amended by Council Directive 2014/87/Euratom of 8 July 2014 [2]. In particular, Articles 8a to 8c introduces:

- a general nuclear safety objective of preventing accidents and, should they occur, mitigating their consequences, avoiding in particular early and large releases,
- general principles to be implemented in order to achieve the nuclear safety objective,
- the need for a robust initial assessment of the safety demonstration of the nuclear installations and the importance of periodic safety reviews.

As defined in the inception report [3], the task 3 of this project, entitled “Performing a detailed study on the safety upgrades in existing reactors in selected Members States” analyzes in some detail the practices and approaches in selected Member States to implement safety upgrades in existing nuclear reactors. This specific report is dedicated to France.

2 SCOPE OF THE REPORT

According to the inception report [3], the report highlights features to control or mitigate severe accidents, notably safety upgrades that have been successfully implemented following Periodic Safety Reviews (PSR) in France. Upgrades proposed in the French Action Plan [6] following the Fukushima Dai-ichi accident are also considered. France is the country with the NPP largest fleet in Europe, including construction of a new plant. As such, most of the safety upgrades are not specific to one NPP, and have to be implemented fleet-wide. There are 3 series of operated PWRs in France : 900 MWe (3-loops RCS, 2 safety-trains (containment heat removal system, high pressure safety injection, low pressure safety injection), single large dry containment with a steel liner), 1300 MWe and 1450 MWe (4-loops RCS, double large dry containment, 2 safety-trains (containment heat removal system, medium pressure safety injection, low pressure safety injection).

This document provides information on a part of those fleet-wide safety upgrades, including on-going activities associated to long term operation (4th PSR).

The following table provides schedule information on the recent and future periodic safety review (PSR) / periodic safety visit (PSV):

Table 1 PSR / PSV schedule for the French PWRs

	1st PSR (1st PSV)	2d PSR (2d PSV)	3rd PSR (3rd PSV)	4th PSR (4th PSV)
900 MWe PWRs (34 reactors)		(1999-2010)	2002-2008 (2009-2020)	2013-2019 (2019-2030)
1300 MWE PWRs (20 reactors)		2002-2005 (2005-2014)	2010-2014 (2015-2024)	2020-2024 (2025-2032)
1450 MWe PWRs (4 reactors)	2002-2005 (2009-2012)			

3 FRENCH PWRS UPGRADES REGARDING SEVERE ACCIDENT MANAGEMENT

3.1 The periodic safety review process for the French NPPs

PSR of nuclear installations were foreseen since the 60's in a French decree and ten-yearly PSR have been performed for NPPs from the beginning of the nuclear program. Nevertheless, this ten-yearly PSR was only required by law in 2006 with the adoption of the law on transparency and security in the nuclear field (TSN act[1]).

The current practice is to organize PSR in 2 phases regarding the wide fleet of French PWR and the existence of only few different plant series (900, 1300 and 1450 MWe). The first phase is a generic "plant series-approach" whose conclusions are applicable to all NPPs belonging to the corresponding series. The second phase takes into account specificities of each NPP, e.g. those related to the site, the results of the compliance assessment, the decennial tests (reactor building, reactor vessel), the ageing management program...

At the very beginning of a PSR process, the operator (EDF) submits to the Safety authority its orientations (objectives) for the review. The Safety authority approves these objectives and (in general) requests additional ones.

For instance, for the next periodic safety review of the 900 MWe series, the operator has presented a program for Long Term Operation (LTO), with the aim to extend the current fleet operation duration beyond 40 years. Such a program includes not only an ageing program but also some important safety upgrades aiming to reduce the gap towards the safety objectives of the generation III reactors like the European Pressurizer Reactor (EPR).

This approach is of particular importance in France considering that the current fleet of reactors will co-exist with the new EPR Flamanville 3 reactor. This new reactor is designed to mitigate core melt accident and its safety objectives encompass this aspect.

The French Safety Authority (supported by IRSN technical reviews) has confirmed the importance of the general objective to reduce the gaps towards the safety objectives defined for the EPR but also to achieve the implementation of all reinforcements decided after the

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Fukushima accident. One major objective is to obtain a significant reduction of the radioactive releases in case of accident (with or without core melt).

Regarding the control and mitigation of severe accidents, safety upgrades are of particular importance considering that 58 reactors of the current fleet were not originally designed to mitigate a core melt accident. Thus, several plants reinforcements have been discussed in France since the 80's and progressively implemented by the operators to allow the management of severe accidents. Since 2000's the PSR process is the main framework to discuss objectives, design and implementations of PWRs reinforcements for severe accident management.

Note: a concept similar to the DEC (design extension condition) have been developed from the mid-70's and the French regulator required the consideration of a first list of multiple failure situations through the "orientation letter" SIN 1076/77 (1977, July 11th). After the Three Miles Island accident, new equipment and procedures were added to manage situations like SBO or LUHS and situations with multiple failures identified with L1 PSA or to mitigate severe accidents. Today, a DEC concept (as defined today) is applied during PSR for internal/external hazards, severe accident reinforcement and management of any new accident scenario that is significant for PSAs.

3.2 French PWRs safety upgrades regarding the management of severe accident decided before 2011

Since the 1990's, Severe Accident Management Guidelines have been developed in France to help the PWR plant operators and emergency teams in limiting the consequences of any postulated severe accident. These guidelines have been progressively improved by taking into account the results of R&D activities and reactor studies.

Some plants modifications have been decided accordingly and have been implemented, such as containment filtered venting system and hydrogen recombiners. The objective was to prevent a containment failure following a core melt accident and to limit the releases into the environment. For instance, the containment venting system has been installed on all French PWR in the 90's to avoid any containment failure in the long term phase of accident. A metallic filter in the containment can retain a large part of aerosol and a sand filter, outside the containment, should retain a major part of the remaining aerosols. The venting line is heated to avoid the steam condensation and to limit the risk of hydrogen combustion within the venting line. This system is supposed to retain efficiently the aerosols and limit the long term impact of a severe accident.

In 2001, the French safety authority requested the NPP operator to develop a "severe accident safety standard" containing the approach and objectives for the prevention and the mitigation of risks associated with severe accidents, the studies necessary to demonstrate compliance with the objectives and the practical provisions and their design basis. It includes the severe accident conditions to be considered for the qualification of structures, systems and components.

For each PSR, the "severe accident safety standard" is now updated by the operator, taking into account research programs (computational codes, experiments …) results.

Level 2 Probabilistic Safety Assessments (L2 PSA) have also been developed in France and are used since 2003 during PSR to discuss the safety upgrades to be implemented. The severe accident reinforcement programme can now be discussed consistently with both deterministic and probabilistic approaches.

Additional modifications for severe accident management have been introduced since 2005:

- a modification of the pressurizer safety valves to improve their reliability in case of station black-out and control the vessel pressure during core melt (avoid direct containment heating, induced steam generator tube rupture and containment bypass);
- the reinforcement of electrical supply of the containment isolation system and optimization of procedures for the manual actions to limit the risk of containment bypass;
- the reactor pit reinforcement (increase of the basemat width and creation of a corium spreading area) for Fessenheim NPPs;
- the reinforcement of the closure system of equipment access penetration for the 900 MWe PWRs reactor building : it allows the closure system to resist to containment exceeding significantly the design pressure;
- the possibility of re-injection of contaminated water from auxiliary buildings (in case of leakage on safety systems recirculation circuits (safety injection or containment spray systems) into the reactor building;

- the reinforcement of the annulus ventilation system of the 1300 and 1450 MWe PWRs double-wall reactor building to limit the releases in case of severe accident;
- the instrumentation to detect hydrogen release in the reactor containment due to core melt;
- the instrumentation to detect vessel rupture due to core melt.

3.3 Safety upgrades specifically decided for the Long term operation programme and after the Fukushima accident (after 2011)

The Fukushima accident had the following impacts on the French severe accident programme for existing PWRs:

- the acceleration of the installation of some of the reinforcements described in section 3.2;
- the decision to verify or reinforce equipment needed for severe accident management against extreme hazards, especially earthquake (before the Fukushima accident, earthquake was not considered in the design of equipment dedicated to severe accident management (FCVS for example));
- the development of a reinforced (against external hazards) crisis centre for each site;
- the development of the FARN (Nuclear rapid action force): rescue team able to help a damaged site 24 hours after a postulated initiating event (with no site access difficulty) or 48 hours after an extreme hazard (with site access difficulty).

The following modifications will contribute to reduce the gap with the EPR safety objectives:

- some sodium tetraborate (borax) baskets have been installed by EDF on the reactor building lower floor of 1300 and 1450 MWe PWRs in order to passively alkalize the sumps water and consequently trap iodine in the liquid phase (this strategy reduces the radioactive iodine release in case of containment venting);
- the reactor vessel zone will be modified to allow corium stabilization and avoid containment basemat melt through in case of reactor vessel failure after core melt :
  - ✓ some arrangements will be added to prevent water entry in the vessel cavity before vessel failure;

- ✓ a channel will be added to connect a room to the vessel cavity in order to allow the corium spreading on larger area than the vessel cavity;
- ✓ an appropriate concrete layer will be added on the corium spreading area to help the corium stabilization;
- ✓ a passive system will activate the corium submersion by water after spreading;

- a new containment heat removal system will be added by EDF on all PWRs, qualified to severe accident conditions to avoid the use of the filtered containment venting during a severe accident and then limit significantly the release for those situations.

The corium stabilization will be a major benefit of these reinforcements and will increase the chances to reach a stable state of the reactor without any large radioactive. However, the radiological consequences cannot be (for a 900 MWe PWR) as low as for EPR due the organisation of the buildings (there is no possibility of direct release from the reactor containment on EPR due to the surrounding buildings).

3.4 Information on the post-Fukushima "complementary assessment" in France (stress-test)

In France, the Fukushima-Daichi accident initiated a reassessment of the safety of NPPs called "complementary safety assessments" (CSA). To quote the French national report [4], "*these complementary safety assessments are part of a two-fold approach: on the one hand, performance of a nuclear safety audit on the French civil nuclear facilities in the light of the Fukushima event, which was requested from French Nuclear Safety Authority (ASN) on 23rd March 2011 by the Prime Minister, pursuant to article 8 of the TSN Act and, on the other, the organisation of "stress tests" requested by the European Council at its meeting of 24th and 25th March 2011*".

The relevance of the studies and positions taken for many years, especially considering PSR implementation, on-going research to improve the safety guidelines for the extension of the duration of operation of the facilities, R&D and improvement of severe accident management arrangements, limitation of releases was confirmed. But given the scale of natural phenomena observed in Japan and whatever the origin of the under-sizing of the plant against natural disasters, it was decided that:

- the robustness of the French facilities should be ensured for levels of external hazards exceeding those considered in the safety framework,
- their robustness for some accidents not retained so far should be as well ensured (long-duration loss of sources that may affect several facilities simultaneously on one site).

Thus the CSA lead to a series of additional resolutions issued by the ASN in 2014, updated in 2017, with a list of requests regarding the prevention of core melt and, should it occurs, mitigation of the consequences of severe accidents. Licensees were asked to implement a so-called “hardened safety core”, i.e. a set of human, organisation and material provisions that withstand rare and severe external hazards. It aims to limit the consequences of accident, including severe accident situations. The approach is based on the defence-in-depth principle, strengthening each level.

The « Hardened Safety Core » should be able to manage accident situations of long duration, affecting several plants of the same site, considering induced events (for example : multi-units LUHS will be considered in the safety report after the 4th PSR). It aims to limit the consequences of very « extreme » situations (but not impossible indeed…), includes « on site » structures, systems and components to cope with the first hours after the accident, before the arrival of « off-site » supports (such as FARN, EDF’s Rapid Nuclear Action Force). The FARN, fully operational by the end of 2015, can provide assistance to a damaged site by providing specialised teams to back up those of the plant concerned and mobile equipment to supply additional water and electricity. A number of modifications were therefore made to the reactors to make it easier to connect equipment brought on-site by the FARN.

Meanwhile, licensees were asked to strengthen their emergency capabilities. New premises are being built on sites. They ensure the protection of the staff in case of emergency, including in case of severe accidents occurring in the site reactors.

Some resolutions related to severe accident management that apply to the whole French fleet of NPPs are outlined in the following table, and detailed in appendix 1.

ASN Resolution	State of Progress
Stress Test 1: Defining the structures and components of the "hardened safety core", including the emergency management premises. Defining the	13/12/2012: EDF sent a first set of provisions of the hardened safety core. 21/01/2014: List of additional resolutions issued by the ASN.

requirements applicable to this hardened safety core. Hardened safety core based on diversified structures and components.	30/06/2014: EDF (operator) submitted the list of new and existing equipment items intended to form part of the hardened safety core, the general hypotheses for the design, construction, verification, qualification and testing of these new or existing equipment items, and the seismic levels for each site. July 2017: List of requests regarding the mitigation of the consequences of severe accidents issued by the ASN.
Stress test 10: Reinforcement of team preparation in the event of an earthquake	Completed (31/12/2013)
Stress test 14.I: Integration of industrial risks in extreme situations.	Completed (30/06/2018)
Stress test 14.II: Coordination with neighbouring industrial operators in the event of an emergency	Alert system implemented on all sites (31/12/2013).
Stress test 19: Redundancy of instrumentation for detecting reactor vessel melt-through and hydrogen in containment	Studies submitted, the modifications are partially deployed and expected to be completed mid-2019.
Stress test 20: Reinforcement of pool condition instrumentation	Additional requests from the ASN as part of the “hardened safety core” resolution (Stress test 1) issued in January 2014.
Stress test 27.I: Study of the feasibility of installing a geotechnical containment or a system with the same effect	Hydrogeological data sheets submitted (30/06/2013) July 2016: ASN asked EDF for more detailed feasibility studies on the implementation of geotechnical containments July 2017: Additional requests concerning the severe accident management strategy. To be examined in 2018-2019.
Stress test 29: Reinforcement of the reactor building filtered containment venting system ("sand-bed filter")	Additional requests from the ASN as part of the “hardened safety core” resolution (Stress test 1) issued in July 2017.
Stress test 30: Designing the emergency premises to withstand earthquakes and flooding	Construction of Crisis centre on sites – 1st centre built in Flamanville.
Stress test 31 : Modifications to ensure facility management further to releases	The licensee has submitted the dose estimates for different scenarios and different parties involved in emergency management. The instrumentation specific to the hardened safety core shall be put in place along with the hardened safety core.
Stress test 32: Multiple plant unit emergency organisation	Completed (13/11/2014)
Stress test 35.I and II: Feasibility of	EDF has transmitted the human actions required for

emergency management actions in extreme situations	managing the extreme situations studied in the stress tests, along with the measures planned to ensure the availability of specialised teams able to intervene on sites. EDF presented ASN with a methodology for determining the personnel numbers permanently present on a nuclear site to deal with an extreme situation. ASN issued a position statement in 2017 and asked EDF for further explanations, in order to improve the overall approach for verifying the correct definition of the personnel number for management of extreme situations. This subject will be examined in 2019 when the hardened safety core operating emergency procedures become available.
Stress test 36: The nuclear rapid intervention force (FARN)	Completed (31/12/2015)

4 References

[1] Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations.

[2] Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations.

[3] Inception Report “Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom”, ENER/D3/2017/-209/-2, Rev 2, 21.03.2018.

[4] Complementary Safety Assessments of the French Nuclear Power Plants (European “stress tests”) - Report by the French Nuclear Safety Authority, December 2011

[5] Peer review country report - Stress tests performed on European nuclear power plants, April 2012

[6] Follow-up to the French Nuclear power Plant Stress Tests - Update of the Action Plan of the Nuclear Safety Authority (ASN), December 2017

5 List of Appendixes

Appendix 1: List of ASN resolutions in the National Action Plan

Appendix 1: List of ASN resolutions in the National Action Plan

Stress Test 1: Defining the structures and components of the "hardened safety core", including the emergency management premises. Defining the requirements applicable to this hardened safety core. Hardened safety core based on diversified structures and components.

I. Before 30 June 2012, the licensee shall propose to ASN a hardened safety core of robust material and organisational measures designed, for the extreme situations studied in the stress tests, to:

- prevent an accident with fuel melt, or limit its progression,
- limit large-scale radioactive releases,
- enable the licensee to perform its emergency management duties.

II. Within this same time-frame, the licensee shall submit to ASN the requirements applicable to this hardened safety core. In order to define these requirements, the licensee adopts significant fixed margins in relation to the requirements applicable on 1 January 2012. The systems, structures and components (SSC) which are included in these measures shall be maintained in a functional state, in particular for the extreme situations studied in the stress tests. These SSC shall be protected against the internal and external hazards induced by these extreme situations, for example: falling loads, impacts from other components and structures, fires, explosions.

III. For this hardened safety core, the licensee installs SSC that are independent and diversified in relation to the existing SSC, in order to limit common mode risks. If applicable, the licensee shall justify the use of undiversified or existing SSC.

IV. The licensee shall take all necessary steps to ensure that the emergency organisation and resources are operational in the event of an accident affecting all or some of the facilities on a given site.

To this end, the licensee includes these steps in the hardened safety core defined in § I. of this resolution and, in accordance with II of this resolution, more specifically sets requirements concerning:

- the emergency situation management premises, so that they offer considerable resistance to hazards and remain accessible and habitable at all times and during long-duration emergencies, including in the event of radioactive releases. These premises shall enable the emergency teams to diagnose the status of the facilities and control the resources of the hardened safety core,
- the availability and operability of the mobile means vital for emergency management,
- the means of communication essential to emergency management, in particular comprising the means of alerting and informing the emergency teams and the public authorities and, should this prove necessary, the arrangements for alerting the population if the off-site emergency plan is triggered in reflex phase by delegation from the Prefect,
- the availability of parameters used to diagnose the status of the facility, as well as meteorological and environmental measurements (radiological and chemical, inside and outside the emergency situation management premises) enabling the radiological impact on the workers and general public to be evaluated and predicted,
- the active dosimetry resources, radiation protection measuring instruments and individual and collective protective means. These means shall be available in sufficient quantity by 31 December 2012.

Stress test 10: Reinforcement of team preparation in the event of an earthquake

Before 30 June 2012, the licensee shall send ASN a training programme for the operating teams to enhance their level of preparedness for earthquake situations. This programme shall in particular include regular in-situation training exercises. This programme shall have been followed by the reactor operating personnel in charge of the seismic instrumentation rack and of the associated operating measures no later than 31 December 2012. The other site operating teams shall receive information by 31 December 2012 and shall have followed the entire programme no later than 31 December 2013.

Stress test 14.I: Integration of industrial risks in extreme situations.

I. No later than 31 December 2013, the licensee shall supplement its ongoing studies with the inclusion of the risk arising from activities taking place near its facilities, in the extreme situations studied by the stress tests and in conjunction with neighbouring licensees responsible for these activities (nuclear facilities, installations classified on environmental protection grounds or other facilities liable to constitute a hazard). By that deadline, the licensee shall propose any modifications to be made to its facilities or their operating procedures as a result of this analysis.

Stress test 14.II: Coordination with neighbouring industrial operators in the event of an emergency

II. No later than 30 September 2012, the licensee shall take all steps, for example by means of agreements or detection and alert systems, to ensure that it is rapidly informed of any event liable to constitute an off-site hazard for its facilities, in order to protect its staff against these hazards and to ensure that emergency management is coordinated with the neighbouring operators.

Stress test 19: Redundancy of instrumentation for detecting reactor vessel melt-through and hydrogen in containment

I. As early as possible, given the constraints of cross-fleet deployment, and in any case before 31 December 2017, the licensee shall install redundant means in the reactor pit to detect vessel melt-through as well as in the containment to detect the presence of hydrogen.

Instrumentation in the control room shall indicate corium melt-through of the vessel.

II. Before 31 December 2013, the licensee shall propose final requirements to ASN for these provisions and shall indicate whether or not they are part of the hardened safety core.

Stress test 20: Reinforcement of pool condition instrumentation

I. Before 30 June 2012, the licensee shall present ASN with the modifications to be made, for measuring both the condition of the fuel storage pool (temperature and water level in the spent fuel pool) and the radiological atmosphere in the fuel building hall.

II. Pending their implementation:

- By 31 December 2012 at the latest, the licensee shall provide its national emergency organisation with charts indicating the times to reach boiling point in the event of total loss of cooling, according to the residual power of the fuel stored in the spent fuel pool.

- No later than 31 December 2013, the licensee shall ensure that level measurement in the event of total loss of electrical power supplies is available.

Stress test 27.I: Study of the feasibility of installing a geotechnical containment or a system with the same effect

I. Before 31 December 2012, the licensee shall send ASN a feasibility study for the installation or renovation of a geotechnical containment or equivalent technical measure to prevent the transfer of radioactive contamination to groundwater and, by means of underground flow, to the surface waters, in the event of a severe accident leading to corium melt-through of the reactor vessel

II. Before 30 June 2013, the licensee shall submit to ASN an updated hydrogeological data sheet for the site, containing the current geological and hydrogeological data.

Stress test 29: Reinforcement of the U5 venting-filtration system ("sand-bed filter")

Before 31 December 2013, the licensee shall submit to ASN a detailed study of the possible improvements to the U5 venting-filtration system, taking account of the following points:

- resistance to hazards,

- limitation of hydrogen combustion risks,

- efficiency of filtration in the case of simultaneous use on two reactors,

- improved filtration of fission products, in particular iodine,

- radiological consequences of opening the device - in particular for accessibility of the site - and the radiological atmosphere of the emergency premises and control room.

Stress test 30: Designing the emergency premises to withstand earthquakes and flooding

I. The licensee shall ensure that the emergency situation management premises can withstand flooding in the event of the flood safety margin level being reached. Before 30 June 2012, it shall present ASN the conclusions of this verification and any modifications considered necessary. Before 30 June 2013, it shall perform any necessary reinforcement work.

II. No later than 30 June 2012, the licensee shall set up independent communication resources allowing direct contact between the site and the national emergency organisation defined in the interministerial directive of 7 April 2005.

III. No later than 30 June 2013, the licensee shall store its mobile resources necessary for emergency management in appropriate premises or zones able to withstand the SSE and flooding in the event of the flood safety margin level being reached.

Stress test 31: Modifications to ensure facility management further to releases

Before 31 December 2012, the licensee shall send ASN a file presenting the planned modifications on its site to ensure that, in the event of release of dangerous substances or opening of the U5 venting-filtration system, operation and monitoring of all the facilities on the site are guaranteed until a long-term safe state is reached; the corresponding deployment schedule shall also be provided.

Stress test 32: Multiple plant unit emergency organisation

Before 31 December 2012, the licensee shall reinforce its material and organisational measures to take account of accident situations simultaneously affecting all or some of the facilities on the site.

Stress test 35.I and II: Feasibility of emergency management actions in extreme situations

I. No later than 31 December 2012, the licensee shall define the human actions required for management of the extreme situations studied in the stress tests. It shall check that these actions can effectively be carried out given the intervention conditions likely to be encountered in such scenarios. It shall for instance take account of the relief of the emergency teams and the logistics necessary for the interventions. It shall specify any material or organisational adaptations envisaged. On the deadline date, the licensee shall transmit a summary of the results of this work and the envisaged measures. On 30 June 2012, the licensee shall send ASN an interim report.

II. Before 31 December 2012, the licensee shall send ASN a list of the necessary emergency management skills, specifying whether these skills could be provided by outside contractors. The licensee shall provide proof that its organisation ensures the availability of the necessary skills in an emergency situation, including if outside contractors are used.

Stress test 36: The nuclear rapid intervention force (FARN)

I. Before 30 June 2012, the licensee shall present ASN with the measures it intends to take in order to provide specialised teams capable of relieving the shift crews and deploying emergency response resources in less than 24 hours, with operations starting on the site within 12 hours following their mobilisation. This system may be common to several of the licensee's nuclear sites.

These teams shall be sized so that they can respond on all the reactors of the site and have measuring instruments that can be deployed as of their arrival. The licensee shall specify the organisation and sizing of these teams, in particular:

- the activation criteria,
- the tasks incumbent upon the teams,
- the material and human resources at their disposal,
- the personal protective equipment,
- the system put into place to ensure the maintenance of these material resources and their permanent operability and availability;
- the training of their staff and the skills currency process.

II. On 31 December 2012, this organisation will be deployable for intervention on a reactor on the site. It shall be able to intervene simultaneously on all the reactors of the site by the end of 2014.

III. Before 30 June 2012, the licensee shall also present the measures for adapting the organisation to simultaneous intervention on several of its nuclear sites.

Comments: The FARN is more specifically tasked with deploying emergency response resources within 24 hours and has its own mobile resources. The ramp-up of this organisation is monitored and checked during inspections. Deployment of the FARN and personnel recruitment are in accordance with the regulatory schedule.

Since 01/01/2016, the FARN has been capable of intervening on 6 reactors on a given site in less than 24 hours, with operations starting on a site 12 hours after mobilisation.

PROJECT

Analysis to Support Implementation in Practice of Articles 8a-8c of Directive 2014/87/Euratom

REACTOR SAFETY UPGRADES IN FINLAND

Revision R5

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Prepared by:

ORGANISATION	NAME	DATE
The contractor	Tuomo Sevón	7 January 2021

Table of contents

List of acronyms

AC	Alternating Current
ACIS	Auxiliary Coolant Injection System
BWR	Boiling Water Reactor
CC BY SA	Creative Commons, Attribution, Share Alike
DC	Direct Current
DEC	Design Extension Condition
EPR	European Pressurized Water Reactor
ETSON	European Technical Safety Organisations Network
EU	European Union
FiR 1	Finland Reactor 1
HPCI	High Pressure Coolant Injection
IRRS	Integrated Regulatory Review Service
NPP	Nuclear Power Plant
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SG	Steam Generator
SMR	Small Modular Reactor
STUK	Säteilyturvakeskus
TRIGA	Training, Research, Isotopes, General Atomics
TSO	Technical Support Organization
TVO	Teollisuuden Voima
VTT	Technical Research Centre of Finland Ltd, Finnish TSO
VVER	Vodo-Vodjanoi Energetitšeski Reaktor

List of figures

1 Introduction

The objective of this project is to support EU member states in achieving a more consistent practical implementation of the Nuclear Safety Directive (Council Directive 2009/71/Euratom), which has been amended by Council Directive 2014/87/Euratom. Articles 8a–8c of the directive set out a new nuclear safety objective, targeting the prevention of accidents and mitigation of their consequences. The directive had to be enacted into national law by 15 August 2017. The role of the European Commission is to ensure that the directive is applied appropriately. This project examines how the directive is being applied in practice at reactor installations in the EU. The project work is performed by a consortium composed of ten ETSON partners.

This report is a part of Task 3 of the project. The objective of this task is performing detailed studies on the safety upgrades performed in selected member states. The studies will be used as an input to the project final report, summarizing the status and identifying best practices and possible gaps in the national implementations of safety upgrades.

This report concerns Finland. The report was written by VTT Technical Research Centre of Finland. A draft of this report was reviewed by Finland's regulator STUK, but the presented opinions and conclusions may differ from the views of STUK.

Chapter 2 of this report contains a brief overview of the Finnish legal framework for nuclear safety and its development since 2011. Chapter 3 concerns safety upgrades performed in Finnish reactors since 2011. The year 2011 was chosen as the starting point because the Fukushima accident in March 2011 triggered several activities to further strengthen nuclear safety worldwide. Chapter 4 discusses the application of digital instrumentation and control (I&C) in Finnish reactors. Chapter 5 includes some discussion on the nuclear safety development in Finland and summarizes the main messages of this report.

Finland has four operating nuclear reactors. Figure 1 shows their locations. Loviisa 1 and 2 are VVER-440 PWRs, designed by the Russian company Atomenergoexport and operated by Fortum. They started commercial operation in 1977 and 1981, respectively. Their electric power is about 510 MW. Olkiluoto 1 and 2 are BWRs, designed by the Swedish company Asea Atom and operated by TVO. They started commercial operation in 1979 and 1982, respectively. Their electric power is about 890 MW.

Finland has one nuclear reactor under construction, Olkiluoto 3. It is an EPR, designed by the French company Areva, and it will be operated by TVO. The construction license was granted in 2005 and the operating license in 2019. The reactor is expected to start operation in 2021. Its electric power will be 1600 MW.

Finland has one nuclear reactor in the planning phase. Construction license application for Hanhikivi 1 was submitted in 2015. It will be an AES-2006 PWR, designed by the Russian company Rosatom, and it will be operated by Fennovoima. The target of Fennovoima is to obtain the construction license in 2021 and start operation of the reactor in 2028. Its electric power will be 1200 MW.

Finland's only research reactor FiR 1 started operation in 1962 and was shut-down permanently in 2015. It was a TRIGA Mark II reactor, designed by the American company General Atomics and operated by VTT. The reactor was located in Espoo, 10 km west from Helsinki. Its thermal power was 250 kW. VTT is going to dismantle the reactor in 2021–2022.

Figure 1. Locations of nuclear power plants in Finland.
Map from Wikimedia Commons by NordNordWest, license CC BY SA 3.0.

2 Finnish Legal Framework for Nuclear Safety

The Radiation and Nuclear Safety Authority STUK is Finland's regulator of radiation and nuclear safety. The abbreviation comes from the Finnish word *Säteilyturvakeskus* ("radiation safety center"). STUK operates under the Ministry of Social Affairs and Health.

Unofficial English translations of Finland's nuclear safety regulations and their explanatory memorandums are available at https://www.stuklex.fi/en. The Nuclear Energy Act lays down general principles for the use of nuclear energy. The Nuclear Energy Decree specifies and complements the act.

STUK is authorized by virtue of the Nuclear Energy Act to issue binding Regulations on technical details concerning nuclear safety. Five Regulations have been issued:

- Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018)
- Regulation on the Emergency Arrangements of a Nuclear Power Plant (STUK Y/2/2018)
- Regulation on Security in the Use of Nuclear Energy (STUK Y/3/2016)
- Regulation on the Safety of Disposal of Nuclear Waste (STUK Y/4/2018)
- Regulation on the Safety of Mining and Milling Operations Aimed at Producing Uranium or Thorium (STUK Y/5/2016)

In addition to the Regulations, STUK publishes regulatory guides on nuclear safety (the YVL Guides). Currently 47 YVL Guides are in force. They are binding, but the licensees have the right to propose an alternative solution, and STUK may approve it if the licensee can convincingly demonstrate that the same safety level is achieved. [1]

The YVL Guides are continuously re-evaluated for updating. When a new YVL Guide is published, it applies as such to new reactors. When considering how new YVL Guides apply to the operating nuclear facilities or to those under construction, STUK takes into account Section 7a of the Nuclear Energy Act: "The safety of nuclear energy use shall be maintained at as high a level as practically possible. For the further development of safety, measures shall be implemented that can be considered justified considering operating experience and safety research and advances in science and technology." [1] STUK can approve exemptions from new requirements if it is not technically or economically reasonable to implement respective modifications to operating reactors, and if the safety justification is considered adequate. [2]

2.1 Classification of Plant States

The Regulation STUK Y/1/2018 classifies plant states into five categories:

- *Normal operating conditions*: planned operation of a nuclear power plant (NPP). This includes testing, plant start-up and shutdown, maintenance, and refueling.
- *Anticipated operation occurrence*: a deviation from normal operation, frequency ≥ 1 / 100 operating years.
- *Postulated accident*: a deviation from normal operation, frequency < 1 / 100 operating years, excluding design extension conditions and severe accidents. Postulated accidents are grouped into two classes on the basis of the frequency of their initiating events:
    - Class 1 postulated accident, frequency ≥ 1 / 1000 operating years
    - Class 2 postulated accident, frequency < 1 / 1000 operating years.
- *Design extension conditions* (DECs) consist of three classes:
    A. An accident where an anticipated operational occurrence or class 1 postulated accident involves a common cause failure in a system required to execute a safety function.
    B. An accident caused by a combination of failures identified as significant on the basis of a probabilistic risk assessment.
    C. An accident caused by a rare external event and which the facility is required to withstand without severe fuel failure. This includes for example some extreme weather phenomena and the impact of a large aircraft.
- *Severe accident*: an accident in which a considerable part of the fuel in a reactor or in a spent fuel storage loses its original structure.

The Finnish classification differs from many other countries. Finland does not use the term *design basis accident*. STUK considers it misleading because severe accidents have to be taken into account in the design of reactors in Finland. A postulated accident in Finland has about the same meaning as a design basis accident in many other countries. In many countries DECs may involve fuel melting, but in Finland reactors have to withstand DECs without fuel damage. In Finnish regulations the term *accident* includes postulated accidents, DECs, and severe accidents. In many other countries accidents refer to design basis accidents only.

The classification of plant states has a direct connection with the defense-in-depth principle. The first level of defense applies to normal operating conditions, preventing any deviations from normal operation. The second level of defense applies to anticipated operational occurrences, preventing their escalation into accidents. The third level of defense applies to both postulated accidents and DECs, preventing their escalation into severe accidents. The fourth and fifth levels of defense apply to severe accidents. The fourth level limits releases to the environment by ensuring the leak-tightness of the containment during a severe accident.

The fifth level mitigates the consequences of severe accidents by means of emergency arrangements outside the NPP.

The five levels of defense-in-depth shall be independent of each other. Because postulated accidents and DECs are on the same level of defense, the same systems can be used for controlling both. However, deterministic safety analyses of postulated accidents shall be made with conservative assumptions, while best-estimate assumptions can be used for DECs. In addition, acceptance criteria are stricter for analyses of postulated accidents than for DECs. Severe accident mitigation systems shall be independent from the systems that are dedicated to postulated accidents and DECs because severe accidents are on a different level of defense.

Currently, there is a lot of discussion about Small Modular Reactors (SMRs) in Finland. They could be used for replacing fossil fuels in district heating. Since many SMR designs have strengthened the first three levels of defense, it could be argued that the fourth and fifth levels of defense could be accordingly weakened or even declared unnecessary. However, no decisions have been made in this issue. The regulator has not made any decisions, since SMR license applications have not been submitted. And potential license applicants would need to know the regulator’s position before they could apply for a license for an SMR. [3]

2.2 Development of the Regulations since 2011

STUK started a project of completely rewriting the YVL Guides in 2008. The project was ongoing when the Fukushima accident took place. Lessons learned from the accident were taken into account in the new YVL Guides, which came into force in December 2013. Prior to the Fukushima accident, Finland already had strict requirements for safety-classified severe accident mitigation systems with N+1 redundancy and independency from the safety systems designed for postulated accidents. Thus, major changes to the regulations were not needed for severe accident management. The new YVL guides require that nuclear facilities have to withstand more severe natural phenomena and power failures. [2]

The Fukushima accident showed that extreme external events can make it difficult to transport equipment and supplies to the site for a long period of time. As a result, the new YVL Guide B.1 (2013) states that, in rare external events, decay heat removal and reactivity control shall be possible “for at least eight hours without any material replenishments or recharging of the DC batteries. In addition, a sufficient inventory of water and fuel and capability to recharge the DC batteries shall exist on site to enable decay heat removal for a period of 72 hours.” “The battery sets supplying severe accident management systems shall be dimensioned to provide a 24-hour discharge time under the highest conceivable load.”

The Fukushima accident highlighted the possibility of a severe accident in several reactors on the same site at the same time. STUK has requested the power companies to ensure the applicability of procedures and the availability of personnel and material resources in case of a multi-unit accident situation in extreme external conditions. In particular, water reserves on the site shall be sufficient for 72 hours even when all reactors on the same site are in an emergency. More consideration will be given to training exercises with multi-unit accidents and long-duration events.

The revised YVL guides did not contain notable technical modification needs with regard to operating facilities since several plant improvements were already initiated after the Fukushima accident. Several plant modifications have also been implemented during the last decades based on previously updated regulatory requirements, probabilistic risk assessment (PRA), and periodic safety reviews. [2]

In 2012, the Finnish regulatory framework for nuclear and radiation safety was reviewed in the IRRS (Integrated Regulatory Review Service) peer review process. According to the IRRS recommendations, some amendments were made to the legislation aimed to increase the independence of STUK and to extend its authorities. The Nuclear Energy Act was amended in 2015 giving STUK a mandate to issue binding STUK Regulations concerning the areas of previous Government Decrees. [2]

The Council Directive 2014/87/Euratom did not cause many changes to Finland’s legislation. The regulations were already in line with the new directive for almost all parts. Article 8c of the directive requires periodic safety reviews at least every 10 years. The requirement of regular periodic safety reviews was already in the Nuclear Energy Act, but the frequency was not specified, so the 10 year interval was added. In practice the periodic safety reviews have been performed every 10 years, so the change in the law did not change the current practice. Peer reviews of the national nuclear safety framework and the regulatory authority (article 8e of the directive) were added to the Nuclear Energy Act. Minor changes to the act were required by the transparency requirements (article 8 of the directive). [4]

3 Safety Upgrades in Finland

As a response to the Fukushima accident, the European Union started so-called stress tests. They did not identify hazards or deficiencies requiring immediate actions at Finnish power plants. There was no indication that any of the plants did not comply with its licensing basis. However, opportunities were found for safety improvements.

3.1 Natural Hazards

The earthquake risk in Finland is very low. No earthquake disasters in the area are mentioned in the written history. The peak ground acceleration that corresponds to the frequency of 10^{-5} /year is 0.056 g for Loviisa and 0.082 g for Olkiluoto. However, the design basis earthquake is set to 0.1 g, as it is the minimum level suggested by IAEA. [5] According to PRAs, earthquakes' contribution is only 2 % of the total core damage frequency in Olkiluoto 1 and 0.6 % in Loviisa 1 [6].

Finland's regulator STUK required both Olkiluoto and Loviisa to update the seismic fragility analyses of the spent fuel pools and firefighting systems. The analyses were updated in 2012 and 2013. The seismic resistance of the firefighting system in Olkiluoto was improved by reinforcing pipe supports in 2014. The licensees have performed plant walkdowns for assessing the seismic risk. They have plans to carry out seismic safety improvements for some components when they are replaced due to ageing. For example, battery stands were replaced with models designed to endure earthquakes at the same time as the batteries were replaced in Loviisa in 2013 and 2014. [2][7]

The flooding risk in Finland is related to high seawater levels. There are no rivers or dams nearby that could cause flooding. Tsunami waves are impossible because the Baltic Sea is shallow. [7]

In Olkiluoto, the probability of exceeding the seawater level design basis is extremely small, less than 1E−9 /year. STUK requested carrying out a more detailed assessment of the effects of exceptionally high seawater level on the cooling systems of the spent fuel interim storage and their electric power supply. The licensee tightened the seam between the seawater pumping station and seawater pipe culvert. [7]

In Loviisa the safety margin against flooding is smaller than in Olkiluoto. The probability of exceeding the seawater level design basis is 4E−7 /year. Although the probability is very small, the consequences of flooding of the basement of the Loviisa NPP could be severe, as all cooling systems might be lost. STUK requested improving protection against external flooding. In 2012 the flood protection was improved during the annual outage, when hatches are open in the condenser cooling seawater system. The flood protections of the buildings most important to safety (the auxiliary emergency feedwater and auxiliary residual heat removal buildings) were improved in 2018. [2][7][8]

Extreme weather phenomena constitute about 12 % of the total core damage frequency in Olkiluoto 1 and about 15 % in Loviisa 1 [6]. The most significant risks are related to very high wind speeds, which could throw debris onto some important components, for example the electric switchyard, and cause a loss of off-site power. Other weather phenomena analyzed in the PRA are high and low air temperatures, high seawater temperature, low seawater level, heavy rainfall, lightning, snow storms, and geomagnetic storms. The probabilities of extreme weather conditions are determined by the Meteorological Institute and reviewed by STUK. The estimates are reviewed and updated every few years. [2][5][7]

As a result of the stress tests, STUK requested both Olkiluoto and Loviisa licensees to analyze the consequences of beyond design basis low and high temperatures, tornados and downbursts. The analyses were performed in 2011, and the safety margins were considered sufficient. [7]

3.2 Loss of Electric Power

The systems for decay heat removal from the reactor, containment and spent fuel pools require external power at both Finnish NPPs. As a result of the Fukushima accident, changes are aimed at decreasing the dependency on the plants' normal electricity supply and distribution systems. Experiences from the Fukushima accident were taken into account in the renewal of the YVL Guides. Now the capability is required to remove decay heat for 8 h without the need to replenish diesel fuel tanks or to charge batteries. Decay heat removal shall be possible for 72 h without any external help from outside the plant. This implies that sufficient diesel fuel must be stored at the plant area for 72 h. [7]

In Olkiluoto 1&2, an independent way of pumping water to the reactor in the case of a loss of AC power was designed by the licensee and approved by STUK in 2015. The arrangement consists of two systems, high and low pressure systems. The low pressure system pumps water to the core from the firefighting water system with additional booster pumps trough the reactor spray system. The system has a dedicated diesel aggregate. The high pressure system, the Auxiliary Coolant Injection System (ACIS), is based on a steam-driven turbine pump (Figure 2). The steam is drawn from the main steam line, and the exhaust from the turbine is routed to the wetwell suppression pool. The water is pumped from the demineralized water tank to the reactor via an auxiliary feedwater line. The system can operate when the reactor pressure is greater than 7 bar. The steam-driven pumps were delivered by SPX Clyde Union. The system was implemented in 2017–2018. [2][8][9]

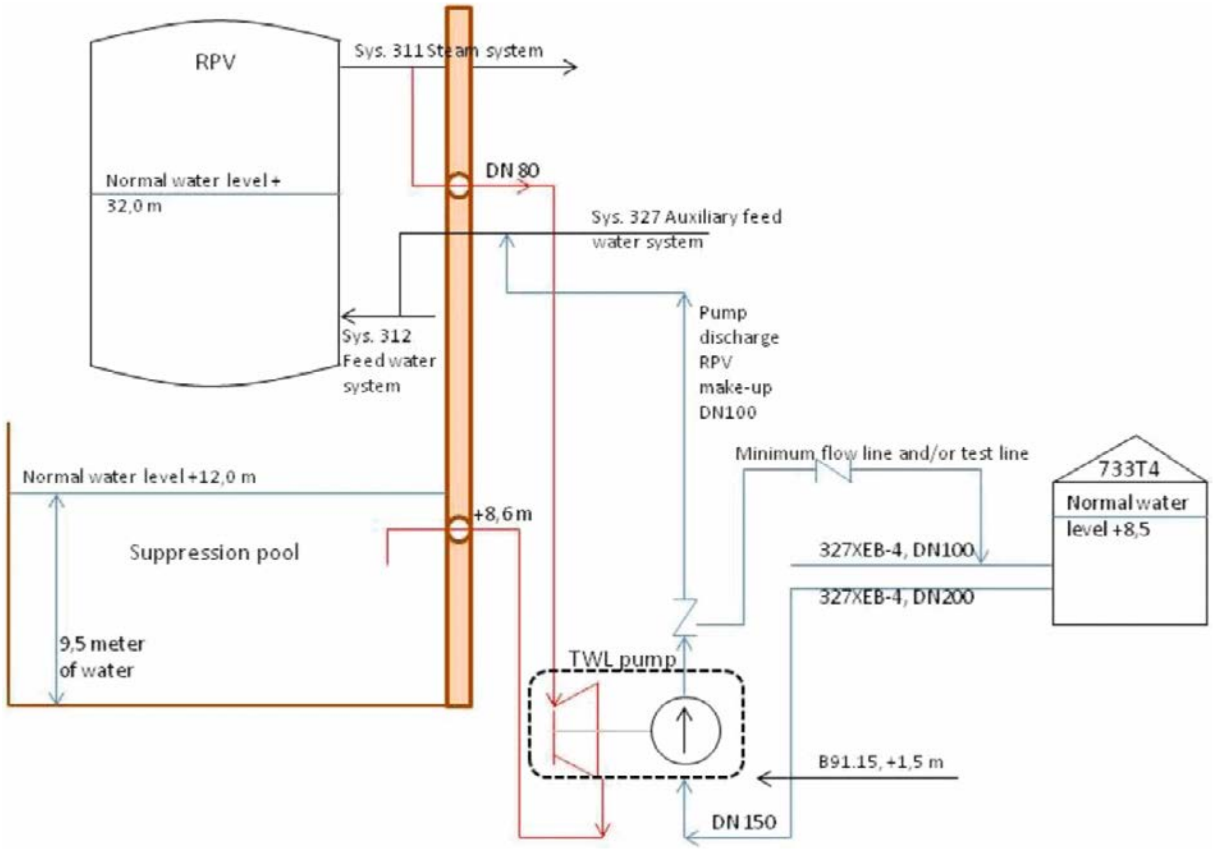

Figure 2. The Auxiliary Coolant Injection System (ACIS) that was constructed in Olkiluoto 1&2. [9]

A project to renew all eight emergency diesel generators and to add one extra generator has been started in Olkiluoto 1&2. In addition to improving safety, another reason for the renewal is that the availability of spare parts to the old generators is declining. It is planned that the first new generator will be installed in 2019, and the project should be completed by 2023. The cost of the investment is over 100 million euros. The new diesel generators are made by Wärtsilä, and they can be cooled by both seawater and air, while the existing generators have only seawater cooling. This will improve their reliability. There is enough diesel fuel for more than one week of operation of the generators, if fuel transfer between different tanks is considered. [2][7][8]

In addition to the fixed emergency diesel generators, the Olkiluoto plant operator has acquired four new 48 kW movable diesel aggregates (Figure 3). The aggregates were delivered by the Finnish company AGCO Power. The aggregates can be used not only at the reactors but also elsewhere at the site, e.g. at the emergency organization’s support facilities, or at the weather mast to secure the weather measurements during an emergency. [10]

Figure 3. A new movable diesel aggregate in Olkiluoto. [10]

For Olkiluoto 3, the possibility to add water to the steam generator secondary side and to the spent fuel pool from the fire water distribution system is being implemented. Additional mobile pumps to provide water to the firefighting system will be acquired before the reactor starts operation. [2]

In Loviisa, the licensee has evaluated measures needed to secure the availability of the diesel-driven auxiliary emergency feedwater system and the electricity supply for the instrumentation needed in accidents. No modifications are planned for the current design concerning AC power supply. [7]

There is enough diesel fuel in the Loviisa generator tanks for at least 72 h of operation, and with realistic loads the duration is evaluated twice as long. In 2012 the licensee purchased a container to transfer diesel fuel at the site. The purpose of the container is to make fuel transfer between the tanks on-site easier and faster. In addition, the licensee has built a new fuel storage tank, from which it is possible to deliver fuel to the diesel generators' day tanks. [7][11]

Conventional diesel fuel is nowadays available in Finland only in a limited scope because the government has ordered mixing of biodiesel into car fuel. Currently the emergency diesel generators at Olkiluoto and Loviisa use conventional diesel fuel. The licensees together with the diesel engine manufacturers have carried out investigations of replacing the conventional diesel with widely available biodiesel. Based on these investigations, biodiesel is allowed to be used in exceptional circumstances. [7]

At Olkiluoto 1&2, the depletion times of DC batteries are well above 10 h, in some cases tens of hours. It is possible to charge the batteries using AC power sources. The batteries supplying the severe accident monitoring systems could also be chargeable by mobile generators. [7] The licensee has decided to install fixed connection points for recharging of all safety important batteries using transportable power generators [6].

In Loviisa, the duration of DC power supply from batteries is considered to be enhanced. The licensee submitted a plan regarding these improvements to STUK at the end of 2012. Some safety important batteries are under a changing process for a larger capacity. Additionally, within the ongoing automation renewal project, the depletion time of the batteries will also be substantially lengthened. [7]

3.3 Loss of Ultimate Heat Sink

The sea is used as the ultimate heat sink in Finnish NPPs. In Loviisa there could be a risk of a loss of ultimate heat sink if an oil tanker accident takes place near to the NPP. An important oil transportation route from Russia passes near Loviisa. Seawater that is badly contaminated with oil cannot be used as a coolant. Another risk of loss of ultimate heat sink could be a very high concentration of algae in the sea, or frazil ice.

In Olkiluoto 1&2, in the event of a loss of ultimate heat sink, the decay heat could be removed by pumping water to the reactor and dumping the steam to the containment and from there to the atmosphere. However, cooling of the pumps requires seawater. The plant operator is modifying the auxiliary feedwater system so that cooling water for the pumps can be taken from the demineralized water tank in addition to the seawater-based cooling chain. By this modification the system will remain operational for a significant period of time even during a loss of the primary ultimate heat sink. The modification was implemented in 2014 at unit 1, and it will be implemented later at unit 2. [2][7]

In Olkiluoto 1&2, new pipeline junctions for injecting water to the spent fuel pools with mobile pumps were completed in 2015. New mobile pumps have been acquired. [2]

In order to secure decay heat removal in the event of a loss of ultimate heat sink, cooling towers were installed in Loviisa. Two air-cooled cooling units per reactor were installed. One unit removes decay heat from the reactor (Figure 4). The other unit removes decay heat from the spent fuel pool and cools other safety-critical equipment. The reactor cooling unit is dimensioned so that it can remove the decay heat 72 h after scram. Until that time the heat can be removed by injecting water to the steam generator secondary side by two diesel-driven auxiliary emergency feedwater pumps and dumping steam into the atmosphere. The cooling towers were designed by GEA EGI Contracting/Engineering Co. Ltd. They are powered by an additional air-cooled diesel generator, so they are independent from the emergency diesel generators. They were taken into use in 2015. A cooling tower during its construction is shown in Figure 5. The cooling towers reduced the core damage frequency by 16 %. [2][12]

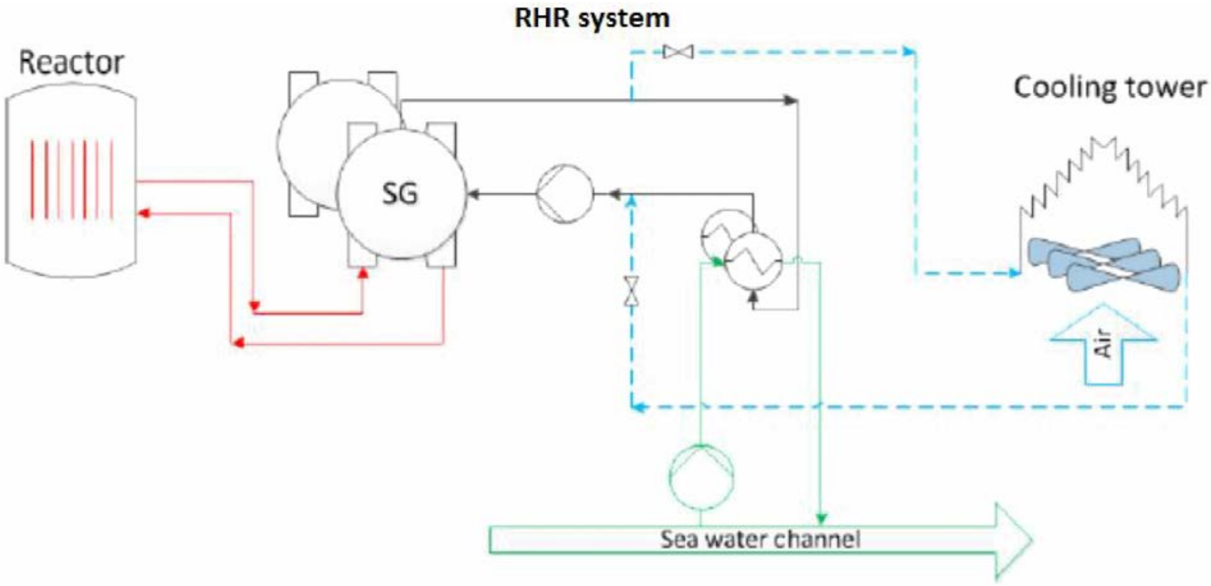

Figure 4. New cooling tower in Loviisa for removing decay heat from the reactor if seawater cannot be used for cooling. [12]

Figure 5. Construction of a new cooling tower in Loviisa. [12]

For Loviisa, STUK has approved modifications concerning water injection to the spent fuel pools with mobile pumps. [2][7]

3.4 Severe Accident Management

Severe accident management was not included in the original design of Olkiluoto 1&2 and Loviisa 1&2. Development of the severe accident strategies started after the Three Mile Island accident. In Olkiluoto 1&2, the management of severe accidents is based on cooling of the core debris in a deep water pool in the containment, while in Loviisa it is based on in-vessel melt retention by cooling the outer surface of the reactor pressure vessel. These backfitting measures were implemented in the 1980's and 1990's. In Olkiluoto, a containment filtered venting system was installed at the end of the 1980's. In Loviisa, an external spray system of the steel containment was installed in 1991, and primary system depressurization valves were installed in 1996. The valves are dedicated for severe accidents, and they are independent from the pressurizer safety valves. Passive autocatalytic recombiners were installed in 2003 in Loviisa. [6]

Prior to the Fukushima accident, Finland already had strict requirements for safety-classified severe accident management systems that are single-failure tolerant and independent from the safety systems that are designed for postulated accidents. Therefore major changes were not necessary as a result of the Fukushima accident.

At Olkiluoto 1&2, all spent fuel pools have been equipped with a temperature and level measurement system, which enables measuring the water level from the normal level down to the top of the fuel assemblies. The measuring system is visible outside the containment and independent from the power supply. The system was completed in 2016. [2][7]

Hydrogen combustion in the reactor buildings of Olkiluoto 1&2 will be prevented by installing hatches that allow venting of the hydrogen out of the building. The steam that is generated by boiling spent fuel pools can be exhausted through this route as well. The design process for the venting hatches was started in 2013, and they will be implemented in 2019. [7][11][6]

The reactor water level measurement system in Olkiluoto 1&2 consists of four parallel subsystems that are based on a differential pressure measurement. The plant operator plans to supplement the system with another system that is based on a different measuring principle. The installation of the new devices has been postponed beyond 2019. [2][8]

Emergency control rooms were constructed for Olkiluoto 1&2 because of tightened requirements concerning the need of supplementary control rooms. They were commissioned in 2015 at unit 2 and in 2016 at unit 1. In situations in which operations in the main control room are not possible, such as a fire, the reactor can be shut down and cooled from the emergency control room. Cooling of the spent fuel pool is also possible from the emergency control room. [2]

All Finnish reactors had full-scope PRAs already before the Fukushima accident. The PRAs cover power operation, start-ups and shutdowns, as well as refueling outages. They include internal initiating events, internal floods, extreme weather events, oil spills in the sea, fires, and seismic events. The PRAs are updated always when plant modifications are performed or when new information becomes available for example from research or operating experience. No need for improving the PRAs was found in the stress tests. The spent fuel storage is not included in the PRAs. STUK and the licensees are discussing about the need for spent fuel storage PRAs. [7]

Comprehensive severe accident management guidelines (SAMGs) were developed in Finland already before the Fukushima accident. Both in Loviisa and in Olkiluoto, the SAM procedures will be improved to support heat removal from the spent fuel pools. In Loviisa, new emergency operating procedures for shutdown states were developed in 2012. They cover the recovery of SAM systems that may be unavailable during an outage. [7]

As a lesson learned from the Fukushima accident, Finnish NPP operators have given more consideration to training exercises with multi-unit accidents and long-duration events, as well as to exercises related to the spent fuel pools. [7]

3.5 Emergency Preparedness and Response

Finland's Government Decree on Emergency Response Arrangements at Nuclear Power Plants was renewed in 2013. In the revised decree there is a requirement to take into account the possibility of several reactor units' simultaneous accident in the emergency planning. The emergency plans were updated both in Olkiluoto and in Loviisa in 2013. An emergency exercise in Loviisa in 2014 contained a two-unit scenario. [7]

Reliability of communication systems during emergencies will be improved by mobile generators that supply electricity to the network base stations at Finnish NPPs. [7]

As a lesson learned from the Fukushima accident, STUK requested the Finnish licensees to provide plans for access control and radiation monitoring of the staff and decontamination measures for personnel, vehicles and materials in case the normal provisions at the plant site are not available (e.g. in extreme natural hazards or fallout). Emergency plans have been updated and the new procedures have been exercised during the annual exercises. [7]

After the Fukushima accident, STUK requested the Finnish licensees to provide plans to restore the access routes to the site in the case of an extreme natural hazard. The plans were finalized in 2013. [7]

A new radiation monitoring network was constructed around the Loviisa NPP. It consists of 28 dose rate measuring stations, and it has been in operational use since 2015. The design basis of the new measuring stations is at least three months autonomic operation with long-term batteries. In addition, the weather monitoring system in Loviisa was renewed. A new on-site weather mast and an additional measurement point in the marine environment were constructed. The additional weather measuring point gives more precise data about the sea breeze and land breeze phenomena, which may strongly affect the dispersion of releases. [2]

3.6 Quantification of Safety Improvements

STUK requires full-scope level 1 and 2 PRAs that include internal initiating events, fires, floods, seismic events, harsh weather, and other external events. The PRAs shall cover power operation and low-power and shutdown states. The PRAs shall be updated continuously to reflect plant and procedure modifications and changes in reliability data. [2] Therefore results of the PRA can be used to quantify the effects of safety improvements.

In Olkiluoto 1, the core damage frequency has decreased from 1.33×10^{-5} /year in 2011 to 6.4×10^{-6} /year in 2018. In Loviisa 1, the core damage frequency has decreased from 4.3×10^{-5} /year in 2011 to 1.2×10^{-5} /year in 2018. [13][6] These values can be compared with the requirement in YVL Guide A.7, which applies to new reactors: the core damage frequency shall be less than 10^{-5} /year. Thus, the Loviisa plant does not fulfill the requirement of new reactors, but it is acceptable for old reactors. In Olkiluoto 3 EPR, the core damage frequency is 3.0×10^{-6} /year [6].

The limit of a large radioactive release in Finnish regulations is 100 TBq of Cs-137. In Olkiluoto, the large release frequency has decreased from 3.5×10^{-6} /year in 2012 to 2.1×10^{-6} /year in 2018. In Loviisa, the large release frequency has decreased from 1.4×10^{-5} /year in 2012 to 7.8×10^{-6} /year in 2018. [15][6] These values can be compared with the requirement in YVL Guide A.7, which applies to new reactors: the large release frequency shall be less than 5×10^{-7} /year. It is practically impossible for the old reactors to fulfill the requirement of new reactors. In Olkiluoto 3 EPR, the large release frequency is 1.0×10^{-7} /year [6].

4 Digital Instrumentation and Control in Finland

4.1 Loviisa 1 and 2

In the beginning of the 2000s, Fortum decided to renew the I&C systems of Loviisa 1 and 2 because, after 20 years of operation, it was getting difficult to obtain spare parts to the old analog I&C systems. The I&C was replaced with digital systems in two projects: LARA and ELSA. [16]

The agreement of the first project, LARA, was signed with Framatome and Siemens in 2004. The scope of the project included renewal of all automation systems and control rooms of both units, and the training simulator. The scale of the project caused enormous complexity. A lot of temporary solutions were needed because the new systems were first coupled to old systems, and later to new systems. Due to delays in design, the project scope had to be changed several times because of other plant modifications. This led to additional delays. The LARA project was terminated in 2014 because it turned out that the large scope and the complexity would lead to even further delays. [16][17]

A new project, ELSA, was started with Rolls-Royce in 2014. As lessons-learned from the LARA project, the scope of the ELSA project was reduced. The new project concentrated on the most important I&C systems (reactor protection, reactor control, and reactor power limitation) and a limited number of the most critical safety improvements. A hardwired backup for accident management, reactor trip, and engineered safety features actuations systems was included in the project. The systems already installed in the LARA project (including waste water process control system, control of the 110 kV / 20 kV switchyard, control rod control, and preventive protection safety functions) were retained and interfaced with the new Rolls-Royce systems. [16][17]

Improved project management practices were applied in the ELSA project. The requirements of the new systems were defined hierarchically, starting from the plant level and progressing to safety functions, redundancy, diversity and separation requirements, and interfaces with old systems. Big issues were solved in the early phase of the project. In the first phase of the project, STUK acceptance was obtained for an extended basic design. During the detailed design it was not necessary to change the upper level requirements. [16]

During the project, the plant safety analyses were renewed using the Apros code, which is developed by VTT Technical Research Centre of Finland and Fortum. The analyses showed that the new systems will work as designed when interfaced with old systems. The Apros model was connected to the automation cabinets in the project test field. The dynamic testing was started immediately when the first new system was ready. [16]

The ELSA project included renewing the control room to a hybrid that includes analog and digital systems of different ages. The operators of the power plant participated in the control room design for several years. VTT validated the new control room systems. In the end, the final design of the control room was validated with the training simulator. [16]

The new I&C was taken into use at both units in three phases in 2016, 2017 and 2018 during the normal annual maintenance outages. The cables and cabinets were installed and tested during power operation. During the outages the new systems were connected to the plant components and to the control room. The project was completed in the planned schedule and budget. [16]

4.2 Olkiluoto 1 and 2

Olkiluoto 1 and 2 are still using analog I&C systems, based on the 1970s technology. TVO is not going to move to digital I&C. Instead, they are going to replace the old devices with new similar analog devices. [18]

4.3 Olkiluoto 3

Design of the digital I&C systems for Olkiluoto 3 was very challenging. STUK found that the general architecture of the design, described in the construction license application in 2004, did not fulfill Finnish requirements. Most connections between the systems were bidirectional, which violated the independency and separation requirements. At the request of STUK, the plant supplier changed the connections that are related to the protection system to one-directional, and separated the severe accident mitigation systems from other I&C systems. The plant supplier continued modifying the system design until the year 2015. [19]

Several methods were utilized for showing that the I&C systems fulfill the requirements. The systems were tested in a test site, at the plant site, and in the simulator. STUK monitored many of the tests. Various reliability analyses were conducted, taking into account possible faults and simultaneous maintenance. STUK requested many changes to the analysis methods before it could conclude that the requirements are fulfilled. In addition to the tests and reliability analyses, also the design processes of the I&C systems were analyzed. VTT performed independent third party analyses for STUK using a formal mathematical model checking method for finding faults. [19]

Reviewing the I&C systems was challenging also for STUK, which hired almost ten experts of digital I&C systems during the 15 years’ project. [20]

5 Discussion and Conclusions

The Finnish legal framework for nuclear safety was briefly presented in this report, and its development since 2011 was described. Safety upgrades performed in Finnish reactors since 2011 were listed.

The Finnish classification of accidents differs from many other countries. Finland does not use the term *design basis accident*, but the term *postulated accident* has about the same meaning. Design extension conditions (DECs) are accidents that have a smaller probability than postulated accidents, but which the reactor must withstand without fuel damage. The term *severe accident* is defined as an accident with fuel damage. In the defense-in-depth principle, both postulated accidents and DECs are on the third level of defense, while severe accidents are on the fourth (and fifth) level. Possible weakening of the fourth and fifth levels of defense for Small Modular Reactors is being discussed, but no decisions have been made.

Finland's regulatory guides, called YVL Guides, are continuously re-evaluated for updating, considering operating experience, research, and advances in technology. The updated regulations apply as such to new reactors, but application to existing reactors is decided on a case-by-case basis. This can be considered as a good practice, since modern technology enables a higher safety level for new reactors, but it would be unreasonable to require the same safety level for old reactors by back-fitting. This practice follows the guiding principle of the Nuclear Energy Act: "The safety of nuclear energy use shall be maintained at as high a level as practically possible."

Finland's regulator STUK started a project of completely rewriting the YVL Guides in 2008. The project was ongoing when the Fukushima accident took place. Lessons learned from the accident were taken into account in the new YVL Guides. Prior to the Fukushima accident, Finland already had strict requirements for safety-classified severe accident management systems with redundancy and independency from the safety systems designed for postulated accidents. Thus, major changes to the regulations were not needed for severe accident management. The new guides require that nuclear facilities have to withstand more severe natural phenomena and power failures.

In the stress tests that were performed after the Fukushima accident, hazards or deficiencies requiring immediate actions were not identified, but many opportunities for safety improvements were found. Safety improvements have been implemented, following the Regulation STUK Y/1/2018: "Opportunities for improvements in technical and organisational safety, identified from operating experience, safety research and technical developments shall be assessed and implemented to the extent regarded as justified".

The most significant plant modifications are related to a loss of electric power. In Olkiluoto 1&2, two new systems for pumping water to the reactor during a loss of AC power were constructed. The low pressure system works with pumps that are independent from the plant electric systems, and the high pressure system pumps operate with steam turbines, similarly to the HPCI and RCIC systems in Fukushima. New mobile generators and pumps have been acquired. They can be used if all AC power is lost. New cooling towers were constructed in Loviisa in order to cope with a loss of ultimate heat sink. For Olkiluoto 3, no major opportunities for safety improvements were found in the stress tests. Only the possibility to pump cooling water to the steam generators and to the spent fuel pool with mobile pumps is implemented.

A potential challenge related to the construction of new safety systems is the increased complexity. It becomes more challenging for the operators to understand the behavior of all the plants systems. Increasing the number of systems also increases challenges in plant maintenance. A leak or malfunction in some of the numerous safety systems could even become an initiating event for an accident in some cases, even though there is no evidence of such events actually occurring.

Severe accident mitigation systems were constructed at all reactors in Finland in the 1980s and 1990s. Therefore only minor improvement possibilities for severe accident mitigation were found in the stress tests.

An important part of the stress tests was a reassessment of natural hazards, including earthquakes, flooding, and extreme weather phenomena. Probability estimates of these threats have been verified, and the effects of external hazards exceeding the design bases have been assessed. Only minor improvements were found beneficial, such as improving the flood protection of certain buildings and anchoring some components to the floors in order to improve their earthquake resistance. As regards emergency preparedness, organizations have been improved so that they can operate during a multi-unit accident.

The effects of safety improvements can be quantified by PRAs. From 2011 to 2018, the core damage frequency has decreased by 52 % in Olkiluoto 1, and by 72 % in Loviisa 1. From 2012 to 2018, the large release frequency has decreased by 40 % in Olkiluoto 1, and by 44 % in Loviisa 1.

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Implementation in Practice of Articles 8a-8c of Directive 2014/87/EURATOM

Safety Upgrades in German Nuclear Power Plants

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Content

1 Introduction

In the aftermath of the accident at the Fukushima Daiichi NPP site, many initiatives have been started to improve nuclear safety word wide. On the global level IAEA has revised its Safety Standards. In particular those safety standards dealing with the design and operation of NPPs, siting and safety assessment have been revised. The contracting parties of the Convention on Nuclear Safety (CNS) have approved the Vienna Declaration on Nuclear Safety (VDNS). On the European level, WENRA has updated its Safety Reference Levels for existing nuclear power plants. Furthermore, the European council amended its directive 2009/71/Euratom by adding a more detailed technical nuclear safety objective as expressed in Articles 8a to 8c.

In practice, safety of existing nuclear power plants has been reassessed in the light of the accident at the Fukushima Daichi NPP site. Directly after the accident in Fukushima, the German regulator BMU has tasked its advisory body, the Reactor Safety Commission (RSK) to conduct a national stress test. Later, the ENSREG stress test has been conducted for all NPPs in Europe. In Germany, the findings of both stress tests, the RSK stress test as well ENSREG stress test, have been included in the National Action Plan for German NPPs and have led to significant safety improvements.

The continuous improvement of nuclear safety has always been an important feature of the German regulatory environment. Long before the accident at the Fukushima Dai-ichi NPP site occurred, safety improvements of German NPPs have been performed since the beginning of the usage of nuclear energy in Germany on a continuous basis. These back-fittings are technically based on insights from lessons learned, operational experience, safety analyses as well as insights from research and development. Major safety upgrades have been implemented after the accident at the Chernobyl NPP. Especially those related to the management of beyond-design-basis accidents have been discussed within the German Reactor Safety Commission (RSK). The licensees have then implemented the discussed safety improvements on a voluntary basis to enhance nuclear safety within their plants. Here, voluntary basis means, that the licensees have committed themselves to follow the RSK recommendations related to the implementation of accident management measures. At the time the recommendations have been made neither a regulatory request nor any national regulation exits. In practice, there were only rare exceptions where accident management measures have not been implemented:

- The recommendation was not “not applicable” for a specific plant;
- The plant has been shut-down prior to the expected completion of the implementation;
- The recommended improvement was already included in the original plants design.

This report describes the legal and regulatory framework for the continuous improvement of nuclear safety in Germany. Implemented safety improvements of German NPPs before as well as after the accident at the Fukushima Daiichi NPP site will be presented. The focus is laid on those safety improvements enhancing accident management measures in the preventive and mitigatory domain.

2 German legal and regulatory framework for nuclear safety

Due to the German legislation the state-of-the-art in science and technology is the applicable standard for all decisions made by the regulator during licensing with respect to nuclear safety. The fundamental safety objective applicable to both licensing and oversight, to protect life, health and real assets against the hazards of nuclear energy and the harmful effects of ionising radiation is laid down under § 1, No. 2 of the Atomic Energy Act (Atomgesetz – AtG) [1]. When granting a licence, one prerequisite is to demonstrate that the applicant / licensee has taken the necessary precautions in the light of the state-of-the-art in science and technology to prevent damage resulting from the erection and operation of the installation. This is regulated in § 7 AtG in Germany.

In addition, § 7d AtG requires that, after a license has been issued, the licensee is obliged to implement the necessary safety improvements according to the ongoing state-of-the-art of science and technology, which are developed, suitable and adequate for providing not only an insignificant contribution to further precaution against risks for the public.

In Germany, the high-level requirements in the mandatory legislation (i.e. AtG) is further detailed in regulations published by BMU, most importantly the “Safety Requirements for Nuclear Power Plants” (SiAnf) [2]. In § 104 of the Radiation Protection Ordinance (Strahlenschutzverordnung – StrlSchV [3]) a rebuttable presumption is established that the necessary precaution in line with the state of the art in science and technology is fulfilled, if the installation meets the requirements contained in the SiAnf and the SiAnf Interpretations:

> “In particular, the licensing authority may consider these precautions to have been taken when the applicant for the design of the nuclear power plant has used as a basis those design basis accidents that must determine the design of a nuclear power plant in accordance with the published safety requirements for nuclear power plants and interpretations of the safety requirements for nuclear power plants.”

In particular, after the accident at the Fukushima Daiichi NPP site the following requirements have been implemented in the SiAnf:

- Requirement for a diverse ultimate heat sink;
- Requirements for connecting mobile equipment;

- Protection concept against internal and external hazards;
- Requirements for plant internal accident management.

Through the SiAnf, the licensee is obliged to give safety highest priority. The licensee is responsible to assure plant safety by giving preference to meeting the safety objective over other plant operational objectives. In addition, the prime objectives of the integrated management system (IMS) of the licensees are specified as:

  - the guarantee of safety,
  - the continuous improvement of safety, and
  - the promotion of safety culture.

The national nuclear regulations in Germany have been constantly developed and continuously adapted to the progressing state-of-the-art in science and technology. The German regulator BMU keeps continuously up to date with the developments in the area of nuclear safety by taking an active role in the work of international committees and working groups (IAEA, OECD/NEA, committees resulting from bi- and multilateral agreements and treaties, etc.). The results of the work of these committees and working groups as well as of the research programs and research and development projects sponsored by the Federal Government on a national as well as on an international level influence the constant improvement of the requirements for the safety of nuclear installations, namely by updating the SiAnf, in accordance with the progressing state-of-the-art in science and technology.

IAEA Safety Standards are considered and are consulted during the revision process of the national nuclear regulations in Germany. In addition, the WENRA safety reference levels for existing reactors [4] and the WENRA safety objectives for new NPP designs [5] are consulted too, as they represent the consensus of all competent European authorities regulating NPPs. The BMU also requests its advisory commissions RSK, ESK and SSK to comment on selected developments and events in the area of nuclear safety and to make recommendations.

The German TSO GRS is contracted by BMU to continuously monitor the ongoing developments in the field of nuclear safety and to support BMU to implement new requirements in the national regulatory framework. Figure 1 summaries the various international sources impacting the continuous improvement of the German regulatory framework.

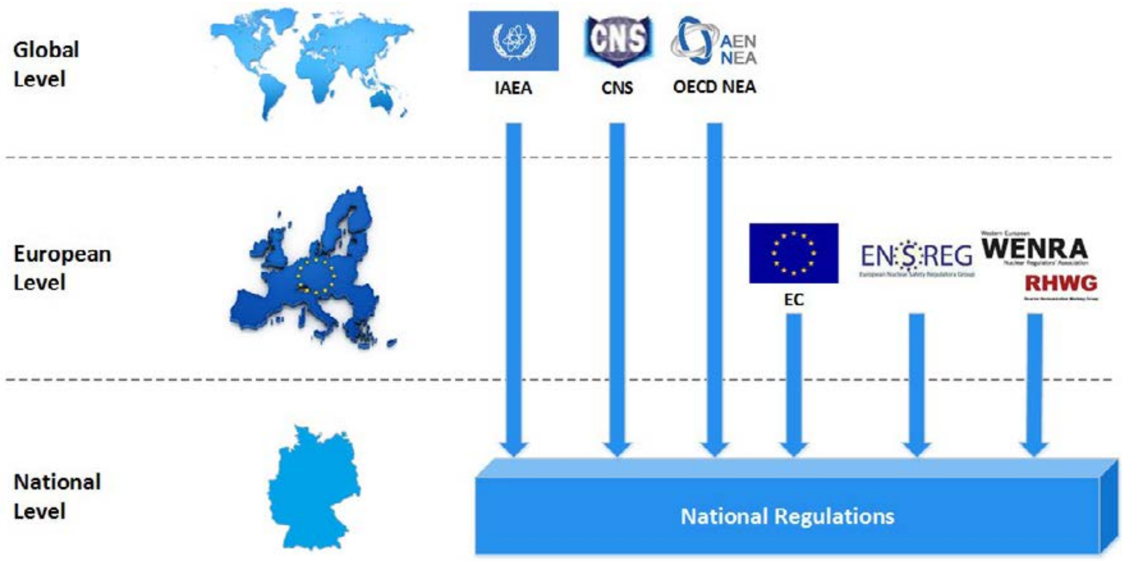

Figure 1 International impact on German Regulations for continuous improvement of the regulatory framework in Germany

If, during continuous regulatory supervision (§ 19 AtG), there are any new safety-related findings, their applicability to other nuclear installations and the need for any possible back-fitting measures will be examined. Events that have occurred in Germany as well as in foreign nuclear installations are evaluated with regard to their safety significance and applicability to other installations. When indicated, recommendations are provided in the form of information notices (WLNs) provided by the expert organization GRS. The indication is based on safety significance and transferability to German NPPs.

The BMU is regularly advised by the commissions: Reactor Safety Commission (RSK), Radiation Protection Commission (SSK) and Nuclear Waste Management Commission (ESK). The RSK provides advice in matters of nuclear safety including matters with respect to the physical protection of nuclear installations. The SSK provides advice in matters of protection against ionising and non-ionising radiation, and the ESK in matters of nuclear waste management. Independence, qualification and reflection of the technical-scientific range of opinions is to be ensured in the commissions. The members are obliged by statutes to express their opinion in a neutral and scientifically sound manner and are appointed by the BMU. The BMU requests its commissions (RSK, SSK, or ESK) for advice on important issues related to licensing and supervisory procedures, the development of rules and regulations, or safety research. In addition, the commissions may also give advice on their own initiative. Generic findings from the consultations of the RSK are considered by BMU when developing the regulatory guidelines further.

Insights from the different processes are used to determine the most recent state-of-the-art in science and technology in the field of nuclear safety. These findings are continuously benchmarked against the national regulatory framework to identify potential needs to improve the existing regulations and to update the German regulations accordingly.

- Germany has a well-established system for operating experience feedback. Every licensee is obliged to report events which occurred in his plant to the authority. Criteria for reporting are established in the Nuclear Safety Officer and Reporting Ordinance (AtSMV). Lists of reported events are regularly published.

- Germany takes active part in various peer review missions. Findings are carefully assessed and potential improvements for the regulatory system as well as for the NPPs will be discussed and, if necessary, timely implemented to further improve nuclear safety. In addition, Germany takes part in further self-assessments and benchmarking processes, like the RHWG-Benchmark on implementation the updated WENRA Reference Level published 2014.

- Germany is actively engaged in and continuously follows the development of international safety standards by continuously performing the following tasks:

    - Active involvement in all IAEA safety standards committees (CSS, EPReSSC, NUSSC, RASSC, WASSC, TRANSSC);

    - Secondment of technical experts for the development and revision of IAEA safety standards;

    - Formal public participation in the process of providing comments on IAEA safety standards by the member states. For this purpose, the relevant drafts are published in the Federal Gazette with an invitation to submit comments;

    - Preparation of annual summary reports on the work of the IAEA on safety standards;

    - Participation in the development and revision of the “WENRA Safety Reference Levels” and Safety Objectives for new nuclear power plants.

Aside from measures taken in the framework of continuous supervision (§ 19 AtG) and § 7d AtG, in accordance with § 19a AtG the licensees are additionally required to conduct and to evaluate a periodic safety review (PSR) of the installation in their responsibility every 10 years, and, on this basis, to continuously improve the nuclear safety of the installation as well. In Germany, the periodic safety review consists of a deterministic safety status analysis, a probabilistic safety analysis and a deterministic analysis of the physical protection of the installation.

By the processes anchored in the German regulatory system as described above, numerous safety improvements have been implemented in German NPPs. In particular, safety improvements have been identified due to an extensive analysis of operational experience of national NPPs and abroad. In Germany, safety improvements have been implemented both after major accidents in NPPs (such as TMI, Chernobyl, or Fukushima), but also on a continuously basis when safety improvements are indicated.

To summarize, Germany made very good experiences with the approach of continuous improvement of its NPPs due to continual but also due to complementary periodic safety reviews. These processes in place ensure that German NPPs have achieved a high level of safety commensurate with the necessary precautions in the light of the state-of-the-art in science and technology to prevent damage.

3 Background on Safety Upgrades

Continuous improvement of nuclear safety is a concept followed in Germany since the late 1970ies. The idea of continuous improvement is implemented in the German Atomic Energy Act and nuclear safety has to follow the state-of-the-art in science and technology. This is one of the prerequisites defined in the Atomic Energy Act which has to be fulfilled before the authority can issue a license (→ § 6 and § 7 para. 2 No. 3 Atomic Energy Act). Furthermore, after a license has been issued the licensee is obliged to implement the necessary safety improvements, if this measure will have not only an insignificant effect on the further precaution against risks (→ § 7d Atomic Energy Act).

Safety of German NPPs relies on the consequent application of the defence in depth concept as required in the SiAnf, which is in line with the requirements of Art. 8b of Council Directive 2009/71/EURATOM amend by Council Directive 2014/87/EURATOM.

In the following, the levels of defence in depth as defined in the SiAnf are described. The German safety philosophy relies on the contribution of each level of defence in depth to meet the nuclear safety objective as required in Art. 8a. To demonstrate that the nuclear safety objective will be met a holistic review and assessment of all measures on all levels of defence in depth is performed.

Level of defence 1: The objective of level of defence 1 is to ensure normal operation (undisturbed, specified normal operation) and to avoid abnormal operation. (compare with Art. 8b No. 1 (b) "*abnormal operation and failures are prevented*")

Level of defence 2: The objective of level of defence 2 is the control of operational occurrences and the avoidance of abnormal operation. The level of defence is characterised by the disturbed, specified normal operation. (compare with Art. 8b No. 1 (c) "*abnormal operation is controlled and failures are detected*")

At the second level of defence, particular importance is attached to the limitation systems that precede the reactor protection system. There are three types of limitation systems that are classified according to task and requirement. In case of anticipated operational occurrences, the limitations shall automatically limit the process variables to defined values in order to increase the availability of the installation (operational limitations) and to maintain initial conditions for the accidents to be considered (limitations of process variables). Furthermore, safety variables are brought back to values at which continuation of specified normal operation is permissible (protective limitations).

The overall objective is to reach a high degree of automation for relief of man from short-term measures and comprehensive preventive measures to counteract the development of anticipated operational occurrences into accidents and a high tolerance against human failures.

Level of defence 3: The objective of level of defence 3 is the control of design basis accidents and the prevention of multiple failure of engineered safety features safety. For this purpose, highly reliable safety systems and the reactor protection system are used. (compare with Art. 8b No. 1 (c) "*accidents within the design basis are controlled*")

In Germany, level 4 is split into three sublevels to control and mitigate accident conditions more severe than design basis accidents (compare with Art. 8b No. 1 (d) "*severe conditions are controlled, including prevention of accidents progression and mitigation of the consequences of severe accidents*")

Level of defence 4a: The objective of level of defence 4a is the control of events with postulated failure of the reactor scram system (ATWS).

Level of defence 4b: The objective of level of defence 4b is the control of events with multiple failure of safety systems to prevent accidents with severe core damage.

Here, preventive measures of accident management (level of defence 4b) are used which are to maintain or restore core cooling and transfer the installation into a safe state.

Level of defence 4c: Subsection 2.1 (3b) of the "Safety Requirements for Nuclear Power Plants" stipulates that on level of defence 4c "mitigative measures of the internal accident management shall be provided for accidents involving severe fuel assembly damages for the purpose of maintaining – by using all available measures and equipment – the integrity of the containment for as long as possible, excluding or limiting releases of radioactive materials into the environment according to Subsection 2.5 (1), and achieving a long-term controllable plant state."

The mitigative measures of level of defence 4c are provided in order to practically exclude events that could lead to

- any releases of radioactive materials caused by the early failure of the containment or
- any releases of radioactive materials requiring wide-area and long-lasting measures of off-site emergency preparedness,

by using all available measures and equipment, or to limit their radiological consequences to such an extent that off-site emergency preparedness measures will only be required to a limited spatial and temporal extent. For the nuclear installations in operation, the practical exclusion of events with early or large releases is proven by the interaction of plant operation, high reliability of the safety system and a comprehensive accident management.

In addition, a protection concept is required to protect the NPPs against internal and external hazards (compare with Art. 8b No. 1 (a) "*the impact of extreme external natural and unintended man-made hazards is minimised*").

All German NPPs are designed to control design basis accidents (i.e. level of defence-in-depth 3) and anticipated transients without scram which belong to sublevel 4a. For accident conditions like events with multiple failure of safety systems (in Germany level of defence-in-depth 4b) or severe accidents with core melt (in Germany level of defence-in-depth 4c), which are internationally called design extension conditions, preventive and mitigative accident management measures have been implemented.

Priority is given to preventive measures in order to avoid an escalation of an event progression to a severe accident with severe fuel degradation. In case of a severe accident, several measures and procedures are in place to control certain conditions to mitigate the consequences of such a severe accident. These measures and procedures are mainly phenomenon based. The main objective of the mitigative accident management is to ensure the integrity of the containment as the last barrier as long as possible.

In the following two sections, the main safety improvements to deal with accident conditions more severe than design basis implemented in German NPPs before and after the accident at the Fukushima Dai-ichi NPP site are described.

3.1 Information notices and RSK recommendations

In Germany, two important drivers for safety upgrades of German NPPs are the Information Notices (WLN) and the RSK statements and recommendations. Annex 1 and Annex 2 lists all Information Notices and RSK statements and recommendations, respectively.

One driver for safety upgrades of German NPPs is operating experience feedback resulting in Information Notices (WLN) of the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH. In the following, this process is described in more detail.

The national Incident Registration Centre is organised at the BfE[1]. The BfE carries out an evaluation of the events reported from the German nuclear installations, including the classification of the events according to the AtSMV[2], lists all information in a database and reports to the BMU in monthly reports. The database of reportable events is accessible to the nuclear licensing and supervisory authorities of the Länder, the BMU and GRS. The current reportable events are discussed in the committees of the RSK on the basis of the monthly reports of the BfE.

On behalf of the BMU, the technical safety organisation GRS evaluates the national and international operating experience on a holistic basis, partly involving further independent experts (Öko-Institut e.V. and Physikerbüro Bremen). In particular, the international events reported within the IRS of the IAEA and in the Working Group on Operating Experience (WGOE) of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) are systematically evaluated with regard to their applicability to German nuclear installations.

If the analysis of the events with safety significance reported by German or foreign nuclear installations reveals an applicability to German nuclear installations, GRS prepares Information Notices (WLN) on behalf of the BMU. These are released by the BMU and transmitted by GRS to the nuclear licensing and supervisory authorities of all Länder with

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1 BfE: Bundesamt für Kerntechnische Entsorgungssicherheit / Federal Office for the Safety of Nuclear Waste Management

2 AtSMV: Nuclear Safety Officer and Reporting Ordinance

nuclear installations, the expert organisations, the licence holders of the nuclear installations, the manufacturers and other specialised institutions.

An Information Notice (WLN) includes the following:

- description of the event;
- a root cause analysis;
- assessment of the safety significance;
- measures taken or planned by the licence holder;
- recommendations on investigations and, where appropriate, corrective measures to be taken at other nuclear installations as an essential element of a WLN.

Upon receipt of an Information Notice (WLN), each licence holder of a nuclear installation then prepares a statement for the competent nuclear licensing and supervisory authority of the Land. The focus of this statement is mainly on the implementation of the recommendations of the respective WLN. The plant-specific results of this information feedback are then reported to the BMU by the respective nuclear licensing and supervisory authority of the Land, including information about the implementation of the recommendations made. The information feedback is evaluated by GRS and made available to all recipients of the WLNs.

The procedures for recording, processing, evaluating and passing on safety-relevant operating experience from German and foreign nuclear installations have proved themselves over the years. The process is anchored in Supervision Manual of the Federation and Länder and is regularly reviewed and further developed. This is to ensure that new sources of knowledge can be identified and included in the feedback of experience.

The second driver for safety improvements are recommendations and statements of the advisory committee RSK (reactor safety commission). The recommendations result from discussions within the RSK either requested by BMU or by RSK itself. The BMU consults the RSK on important topics concerning licensing and supervisory procedures, development of regulations or safety research. The members of RSK are independent, qualified for the tasks and reflect the range of technical-scientific opinions. According to the commissions' statutes, the members are committed to express neutral and scientifically plausible opinions. The members of the RSK are appointed by the BMU. The focus of their activities is primarily on providing support and advice on issues of fundamental importance and initiating further safety-related developments. The results of the consultations are published in the form of general recommendations and statements concerning individual issues.

In addition, the results from the RSK are also made available to the licensing and supervisory authorities of the Länder. Following approval by the BMU, the minutes of the RSK and its committees as well as the RSK statements and recommendations are distributed by the RSK Secretariat to the competent licensing and supervisory authorities of the Länder. If the BMU has comments on the decisions (statements and recommendations) of the RSK, these will be communicated to the Länder upon transmission. Unless otherwise stated in the transmission text of the RSK Secretariat, the resolutions also reflect the BMU's view of the specific topic, so that it will base this on any assessments on the context of federal oversight. The licensing and supervisory authorities of the Länder evaluate the minutes and assess the decisions (recommendations and statements) of the RSK. The statements and recommendations of the RSK always take into account the current state of the art in science and technology and may therefore contain new findings. Generic findings from the RSK discussions are incorporated in the BMU's further development of the regulations.

3.2 Safety improvement in German NPPs before Fukushima

In Germany, most important safety improvements to deal with beyond-design-basis events have been implemented in the late 1980s and 1990s. As a response to the severe accidents at Three Miles Island and especially after the Chernobyl accident in 1986, the severe accident management (SAM) concept for German NPPs has been discussed by the German Reactor Safety Commission (RSK) with inclusion of both the vendor and utilities. The progression of the discussions and the requirements for the SAM concept published by RSK are described in detail inside chapter 6 of the ENSREG country report of Germany [6].

First requirements for a Severe Accident Management (SAM) program regarding beyond-design-basis events starting from power operation only were published in autumn 1988 after intensive discussions within the RSK [7]. The concept was called “Anlagen-interner Notfallschutz” (plant internal accident management). This plant internal accident management relies on fixed installed equipment, portable equipment, and SAMG. The primary intention was the prevention of severe accidents starting at power operation.

Some selected mitigative measures for dominating phenomena were proposed as well. For both necessary hardware modifications have been considered. The filtered containment venting system was one of the systems which was recommended and installed very early, in the late 1980s [8] [9]. In the following, reference is made to the major relevant RSK decisions relating to accident management:

- Containment isolation, RSK Recommendation, 218th meeting 17.12.1986 [7]
- Filtered venting of PWR containment, 218th meeting, 17.12.1986 [7]
- Filtered venting of BWR containment, 222nd meeting, 24.06.1987 [7]
- N_2 inertisation of BWR containment, 218th meeting 17.12.1986 [7]−Start of detailed discussions about accident management 1987/88
- development of an Accident Management Manual, 226th meeting, 21.10.1987
- Additional RPV injection or refilling options (BWR), 226nd meeting, 21.10.1987
- Electrical power supply, 226nd meeting, 21.10.1987
- Secondary-side and primary-side bleed and feed (PWR), 233rd meeting, 22.06.1988
- Diverse RPV pressure limitation for BWR, from 1989 onwards
- RSK Position Paper on accident management (273rd meeting), 1992 [10]
- Hydrogen recombination, RSK Position Paper, 314th meeting, 17.12.1997 [11] (Discussions since around 1987 regarding igniters or passive autocatalytic recombiners or dual concept)

Additional information was compiled by the Nuclear Safety Standards Commission (KTA) in 1996 [12].

In the context of the legally required decennial Periodic Safety Reviews (PSR) the defence in depth concept and the fundamental safety functions have to be reassessed using current site conditions and impacts conceivable at the plant site. These regular safety reviews address enhanced protection against hazards as well as the implementation of on-site or plant internal preventive and mitigative accident management measures. A PSR guideline specifies a set of beyond-design-basis scenarios to be analysed and covered by the accident management manual.

In parallel to the discussions of the RSK mentioned above several severe accident analyses have been performed by different institutions. For example, the basis design concept of the passive autocatalytic recombiners has been supported by containment analyses done at GRS.

3.3 Safety improvements in German NPPs after Fukushima

After Fukushima the robustness of the German NPPs against Fukushima like conditions (external hazards, station blackout, loss of service water cooling chain) has been reassessed in the frame of both the German stress test as well as the European stress test. Recommendations for the optimization of the SAM concept of German NPPs have been issued by the German Reactor Safety Commission.

The identification of potential safety improvements was based on a systematic analysis to ensure vital safety functions. This systematic analysis was performed in mainly perfromed in four steps [13]:

1. Identification of potential improvements to increase the robustness in case of hazards exceeding the design basis events.Unprobable, but not yet practically excluded, scenarios shall be taken into account.
2. Assessment of margins of existing safety features or measures for plant internal accident management to determine if in case of impacts exceeding the design basis will challenge the needed safety functions. These assessments will be based on engineering judgement.
3. Based on insights from step 2 it shall be assessed whether an increase of the robustness is possible either by improving existing safety features or by using existing or implementing additional safety features to compensate a potential loss of safety functions.
4. In the last step the necessary support and auxiliary systems should be analysed to ensure vital safety functions (e.g. electric power supply, essential cooling water supply)

The main recommendations are:

- long-term energy supply (e.g. mobile generator, bunkered supply connections),
- long-term heat removal from reactor core and spent fuel pool (second ultimate heat sink, which means a diverse heat sink like e.g. water/air heat exchanger, groundwater well etc.),
- long-term heat removal from wetwell for a BWR,
- safe release of the off-gas containing combustible gas species by the filtered containment venting system,
- availability of the measures under conditions of long-term Station Blackout,
- identification of available safety margins,
- SAM measures for the protection of the building structures surrounding SFP of a BWR against hydrogen combustions (e.g. passive autocatalytic recombiners),
- optimization of existing measures, and
- need of a SAMG Concept.

RSK performed a review on how and whether the RSK recommendations to strengthen the robustness of the plant have been adapted by the licensees [14]. One result was, that the systematic approach to derive additional safety demonstrations and additional safety features are in accordance with the RSK recommendations [13].

The complete list of recommendations can be found also in the German Action Plan [15] given on the BMU web side (https://www.bmu.de/en/download/national-action-plan-implementing-lessons-learned-from-fukushima-for-german-nuclear-power-plants/).

4 Technical Implementation in German NPPs

In the past, there have been several drivers to identify potential safety improvements. In case a certain technical solution has been selected to be implemented, the efficiency and effectivity of the solution must be demonstrated by safety analysis, typically by applying best estimate methods using realistic assumptions and boundary conditions.

4.1 Examples of Safety Improvements before Fukushima

In response to the severe accidents at Three Mile Island and especially after the Chernobyl accident in 1986, the German Reactor Safety Commission (RSK) was asked to check whether any measures to enhance the NPPs safety and to cope with severe accidents are possible and if so, what these measures could be. The results of the German Risk Study for Nuclear Power Plants „Deutsche Risikostudie Kernkraftwerke - Phase B" (1981-1989) [16], the first large comprehensive study including deterministic and probabilistic results of severe accidents based on a PWR reference plant, significantly influenced the development regarding severe accident management in Germany.

The primary intention of the SAM concept was the prevention of severe accidents starting at power operation. Some selected mitigative measures for dominating phenomena were proposed as well. For both prevention and mitigation respectively, necessary hardware modifications had been considered. The filtered containment venting system was one of the systems which was recommended and installed very early, in the late 1980s. The final RSK recommendation regarding a Severe Accident Management Program was published in 1992 [8] and provided all details for SAM concepts to be developed and implemented by the licensees to deal with severe accidents starting from full power operation.

Example 1: Implementation of passive autocatalytic recombiners (PAR)

Exemplarily, the process for the development of the hydrogen management during severe accident scenarios for German PWR shall be summarized here. The discussions regarding igniters or passive autocatalytic recombiners or a dual concept have been conducted since around 1987. The basis design has been examined by GRS between 1995 and 1998. A representative spectrum of severe accident sequences has been selected for the assessment of a PAR concept. Finally, the following severe accident sequences have been considered in these analyses:

- Large break loss of coolant accident (LB LOCA) due to guillotine break of surge line,
- Small break LOCA (SB LOCA) in the hot leg without secondary side cooldown,
- Station Blackout (SBO) with primary side depressurization and flooding of a partially damaged core after recovering of the electrical power supply,
- Total loss of feed water with primary side depressurization, and
- Small break LOCA (SB LOCA) in the hot leg with secondary side cooldown.

As for the SBO case no relevant hydrogen generation occurred in the calculation due to the reflooding of the core by recovering of the ECCS[3], that case has been replaced by the SB LOCA with secondary cooldown (5th case of the list above).

The examination of the basic design of the PAR system led to the RSK Position Paper, 314th meeting, 17-12-1997 [11] regarding the hydrogen recombination, which is the main accident measure for the hydrogen management during severe accident sequences. Passive autocatalytic recombiners were installed in all German PWRs and BWRs in the 1990ties on a voluntary basis by the utilities. For two German PWRs, KKS and KWO, no passive autocatalytic recombiners (PARs) have been installed because both plants have been permanently shut down shortly after the recommendation was made. As shown in Table 1 KKS made an application for a implementing a PAR system, but this was not any more realised due to its permanent shut down.

The possibility that recombiners could act as igniters during severe accidents sequences leading to an endangerment of the containment has been discussed by the RSK after 2000. That discussion ended in the RSK recommendation 419th Meeting, 03-09-2009 which depicts a summarization of actual safety relevant issues regarding the PAR concept. One of these recommendations was the extension of the deterministic analyses for the re-assessment of the PAR concepts. This should cover dry conditions in the containment by the consideration of the injection of cold water from the ECCS. Generally, the effectiveness of the PAR concept realized in the German NPPs was successfully demonstrated. Table 1 and Table 2 summarize the safety improvements implemented in German PWRs and BWRs before the accident at the NPP Fukushima Dai-ichi occurred, respectively.

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[3] ECCS: emergency core cooling system

Table 1 Overview on implemented safety improvements in German PWRs

No.	Measure	KWO[9)]	KKS[1)]	KWB A	GKN 1	KWB B	KKU	KKG	KWG	KKP 2	KBR	KKI 2	KKE	GKN 2
1	Accident management manual	R/1989-92	R/1992	R/1990	R/1988	R/1990	R/1989	R/1993	R/1992	R/1990	R/1987	R/1991	R/1994	R/1988
2	Assurance of core cooling													
2.1	Secondary-side bleed	R/1991	R/1991-95	R/2002	R/1992-94	R/2003	R/1992	R/1995	R/1993	R/1992	R[10)]	R/1995	D	D
2.2	Secondary-side feed	R/1991	R/1992-95	R/2002	R/1991	R/2003	R/1992	R/1990	R/1993	R/1992	R/1994	R/1995	R/1990	R/1991
2.3	Primary-side bleed	R/1992	R/1991[5)]	R/1990	R/1993	R/1991	R/1991	R/1999	R/1999	R/1993	R/2003	R/1995	R/1996	R/1993
2.4	Primary-side feed	R/1991	R/1991	R/1990	R/1993	R/1990	R/1991	R/1995	R/1999	design	R/1989	R/1995	D	D
3	Activity retention/assurance of containment integrity													
3.1	Filtered containment venting	R/1991	R/1994	R/2002	R/1992	R/2003	R/1992	R/1993	R/1993	R/1990	R/2003	R/1991	R/1991	R/1990
3.2	Hydrogen management by passive autocatalytic recom-biners	[2)]	A[3)]	R/2010	R/2001	R/2003	R/2000	R/2000	R/2000	R/2001	R/2003	R/2000	R/1999	R/1999
3.3	Assured containment isolation	R/1991	R/1988	R/1991	R/1990	R/1991	R/1991	R/1991	design	R/1990	R[10)]	R[10)]	D	D
3.4	Control room supply air filter-ing	R/1990	R/1992	R/1989	R/1991	R/1989	R/1989	R/1992	R/1990	R/1990	R/1998	R/1989	D	R/1988
3.5	Containment sampling system	R/2001	-	[6)]	R/1999	A/2008	R/2001	R/2003	R/2000	R/2001	R/2007	R/2002	R/2000	R/2002
4	Assurance of emergency power supply													
4.1	Neighbouring unit	n. a.	n. a.	R[10)]	R/1990	R	n. a.	n. a.	n. a.	R/1984	n. a.	[7)]	n. a.	R/1988
4.2	Increased battery capacity	R/1989	D	R/1991-92	R/1989-93	R/1991	D	R/1995	D	D[4)]	R[10)]	R/1989	R/1988-90	R/1988
4.3	Restoration of grid supply	D	R/1990	R/1990	R/1989	R/1990	R/1989	R/1990	R/1990	R/1989	R/1995	R[10)]	R/1996	D

No.	Measure	KWO[9)]	KKS[1)]	KWB A	GKN 1	KWB B	KKU	KKG	KWG	KKP 2	KBR	KKI 2	KKE	GKN 2
4.4	Additional grid supply via underground cable	R/1989	R/1992	R/1985	R/1989	R/1985	R/1992	R/1995	R/1993	R/1992	R/1995	R/1992[8)]	R/1993	R1988

A – application, L – licencing, R – realisation, D – included in original design, n. a. – not applicable

1) KKS was finally shut down on 14/11/2003, the first licence for decommissioning and dismantling was granted on 07/09/2005.
2) On 13/03/2003, the utility withdrew the application of 16/06/1999 due to the planned cessation of power operation by November 2005 at the latest.
3) The installation of catalytic recombines that had originally planned for the 2001 overall maintenance and refuelling outage was not carried out after all since the decommissioning of the plant was applied for on 23/07/2001.
4) In 2001, additional increase of the battery capacity in connection with the use of digital I&C and computer replacement.
5) Feasibility was confirmed but was not part of the operating licence.
6) An application for KWB A was only planned after conclusion of the corresponding licensing procedure for KWB B from 2008, but KWB A/B were permanently closed in 2011.
7) Not planned.
8) Utilisation of a 20-kV connection to the Isar hydroelectric power plant chain.
9) Final shutdown KWO on 1/05/2005, the first licence for decommissioning and dismantling was granted on 28/08/2008.
10) Safety improvement was implemented, but exact year not known by the authors

Table 2 Overview of implemented safety improvements in German BWRs.

No.	Measure	KKB	KKI	KKP 1	KKK	KRB B	KRB C
1	Accident management manual	R	R/1991	R/1989	R/1988	R/1991	R/1991
2	Assurance of core cooling and containment integrity						
2.1	Self-sufficient feed system (TJ system for BWRs model year 69)	R	R	R/1991	R/1989	n. a.	n. a.
2.2	Diverse containment pressure limitation	R/1991	R/1990	R/1990	R/1991	R/1992-93	R/1993
2.3	Additional RPV injection and make-up feeding	R	R	R/1990	R/1988	R/1995[1)]	R/1995[1)]
3	Activity retention/assurance of containment integrity						
3.1	Filtered containment venting	R/1988	R	R/1989	R/1988	R/1990	R/1990
3.2	Hydrogen management by inertisation, passive autocatalytic recombiners	R/1988	R	R/1989	R/1988	R/1990[6)]	R/1990[6)]
3.3	Assured containment isolation	R/1988	R	R/1989	R/1988	D	D
3.4	Control room supply air filtering	R/1988	R	R/1989	R/1988	R/1990	R/1990
3.5	Containment sampling system	[4)]	R/2007	R/2001	[5)]	R/2009	R/2009
4	Assurance of emergency power supply						
4.1	Neighbouring unit	n. a.	[7)]	R/1984-85	n. a.	R	R
4.2	Increased battery capacity	R	D	R/1987-88	R/1990	D	D
4.3	Restoration of grid supply	R	R	D	R/1989	R	R
4.4	Additional grid supply via underground cable	R/1990[2)]	R[3)]	R/1992	R/1990	R/1991	R/1991

A – application, R – realisation, D – included in original design, n. a. – not applicable

1) Realisation of an additional and independent residual-heat removal system.
2) Additional gas turbine with black start-up capability for supply emergency power system II (UNS).
3) Utilisation of a 6-kV connection to the Isar hydroelectric power plant chain.
4) KKB was permanently shut down in 2011.
5) KKK was permanently shut down in 2011. Thus, application to implement a containment sampling system was stopped.
6) Hydrogen recombiners (R/1999-2000) in connection with inertisation in the pressure suppression pool.
7) Not planned

4.2 Examples of Safety Improvements after Fukushima

As shown in section 4.1, German NPPs have got a comprehensive severe accident management concept regarding SAM measures for prevention and mitigation very early. Nevertheless, after Fukushima the robustness of the German NPPs against Fukushima like conditions (external hazards, Station Blackout, loss of service water cooling chain) has been re-assessed on behalf of the German government in the frame of both the German stress test as well as the European stress test. Recommendations for further optimization of the SAM concept of German NPPs have been issued by the German Reactor Safety Commission on behalf of the Federal Ministry BMU. The main recommendations resulting from the stress tests are:

- Addressing long term aspects of severe accidents:
  - long-term energy supply (e.g. mobile generator, bunkered supply connections);
  - long-term heat removal from reactor core and spent fuel pool (second ultimate heat sink, which means a diverse heat sink like e.g. water/air heat exchanger, groundwater well etc.);
  - long-term heat removal from wetwell for a BWR;
  - availability of the measures under conditions of long-term Station Blackout;
- Enhancing protection against hazards due to combustible gases:
  - safe release of the off-gas containing combustible gas species by the filtered containment venting system;
  - SAM measures for the protection of the building structures surrounding SFP of a BWR against hydrogen combustions (e.g. passive autocatalytic recombiners);
- Further optimization of plant robustness:
  - identification of available safety margins;
  - optimization of existing measures;
  - need of a SAMG concept.

The complete list of recommendations can be found also in the German Action Plan [15] given on the BMU web site (https://www.bmu.de/en/download/national-action-plan-implementing-lessons-learned-from-fukushima-for-german-nuclear-power-plants/)

Implementation of additional mobile equipment

For handling the long-lasting SBO e.g. in the PWR plants, in addition to the existing eight emergency diesel generators two new mobile emergency diesel generators have been added to the PWR plants in the aftermath of the Fukushima accident. In co-operation with the vendor of the plant, the licensees designed the system of the additional mobile emergency diesel generators and their bunkered supply connections located on the plant site. In case of a long-lasting SBO, the two mobile emergency diesel generators which are deposited on the plant site can be connected to the auxiliary power supply of the plant. One mobile emergency diesel generator recovers the instrumentation of the plant and all redundancies of the extra borating system allowing the injection of borated water with a mass flow of 8 kg/s in total. The second mobile diesel generator provides one bunkered redundancy of the emergency core cooling system. That allows the injection of coolant from the flooding tank by the SFP cooling pump. The main goal of the application of the mobile diesel generators is the recovering of core cooling and finally the transition to the closed circulation cooling mode of ECCS. With these measures the plant can be brought in a stable and safe state in case of a long-lasting SBO event.

Example 3: Implementation of SAMGs

After the Fukushima severe accident, a SAMG concept with additional mitigative SAM measures have been implemented in all German NPPs. The new SAMG concept for the German NPPs has been developed by Framatome (former AREVA NP) on behalf of the utilities. After finishing the concept, the SAMG concept has been checked for each plant by the Federal Ministry BMU, the responsible state authority and its TSO.

The SAMG strategies and their appendant procedures come into play for severe accidents with core damage. Such severe accident scenarios are very unlikely for German NPPs due to their design and realized EOPs. They are only possible from initiating events (transients and LOCAs) in combination with multiple failures of safety systems (more than the demanded failures from the single failure concept), the finalization of the preventive measures of the “Emergency Operating Manual (NHB)” of the plant and the absence of successful recovery actions of failed systems/components. For such cases, the new SAMG concept is documented in the “Handbook of Mitigative Severe Accident Management Measures (HMN)”, which is part of the plant documentation of all German NPPs.

After reaching the criterion for transition from the “Emergency Operating Manual (NHB)” into the HMN, the crisis team must perform a periodically diagnosis of both the status of the reactor pressure vessel and the status of the containment respectively. Dependent of the plant state evaluated from the diagnosis, a strategy and appendant processes for mitigating the SA progression will be selected from HMN by the crisis team. The procedures cover e.g. recovery actions of failed systems/components, additional injection of water from operational systems, and the usage of systems/components to minimize the release of radionuclides into environment. The main goal of the application of the SAMG strategies/procedures is to mitigate the severe accident progression, to minimize the release of radionuclides, and to get the plant in a secured and safe state in the long term. These goals are demanded in the German nuclear regulatory framework (Interpretation I-7 (Requirements for accident management) of the “Safety Requirements for Nuclear Power Plants”). They are:

- termination of the core melt sequence;
- protection of intact barriers for retention of radionuclides;
- limitation of the release of radionuclides;
- achievement of a controllable plant state in the long-term.

5 Conclusion

In Germany, improving nuclear safety was a task from the onset of nuclear energy and will be a task until the phase out from nuclear energy has been completed. A thorough analysis of operating experience feedback and insights from research and development in nuclear safety is the main driver for identifying safety improvements. Thus, not only major accidents like those occurred in the NPPs TMI, Chernobyl or Fukushima Daiichi triggered several safety improvements as described in chapters 3 and 4. It can be concluded that following the state-of–the-art in science and technology in nuclear safety, identifying and implementing safety improvements is a living practice in Germany.

To ensure that nuclear safety is continuously improved in the operating NPPs in Germany, a comprehensive programme to monitor the progressing state-of-the-art in nuclear safety has been established as described in chapter 2. This is regarded as mandatory to firstly identify potential safety improvements and to secondly assess the necessity to implement a safety improvement based on the safety significance and the transferability of findings to German NPPs.

As shown in general and by the examples in chapter 4, the interaction of research in nuclear safety, continuous evaluation of operating experiences of German NPPs and NPPs abroad, analysing lessons learned from major accidents and deriving actions to further optimize nuclear safety as well as insights from deterministic and probabilistic safety analyses as well as the concept of continuous supervision are considered an effective set of instruments to identify and implement safety improvements and continuous optimization of nuclear safety.

German NPPs in operation have been designed according to safety standards of the 1970s and 1980s. The practical elimination of events leading to early or large releases at German NPPs is demonstrated by the interaction of plant operation, high reliability of the safety system and a comprehensive accident management. It can be illustrated by five tiers:

(1) The first tier forms the design of systems, structures and components of high reliability and quality. One example is the application of the concept of basic safety developed in the late 1970s years to prevent catastrophic failure of those components. It is characterised by the following principles:

- safety high-quality materials, especially with respect to fracture toughness;
- conservative stress limits;
- avoidance of peak stresses by optimisation of the design;
- ensuring application of optimised manufacturing and test technologies;
- knowledge and evaluation of existing flaws;
- accounting for the operating medium.

Later, this concept was developed further to the integrity concept. Until now, the integrity concept has been proven in practice and presents an important contribution in terms of damage precaution. The technical basis for it is nuclear safety standard KTA 3206 "Verification Analysis for Rupture Preclusion for Pressure Retaining Components in Nuclear Power Plants" [10]. By thorough application of these deterministic approaches, the frequency of e.g. loss of coolant accidents could be reduced.

(2) The second tier is characterized by the highly reliable safety systems. In case of postulated design basis accidents, e.g. large break loss of coolant accident, it has to be demonstrated that the capacity of the emergency core cooling systems can cope with a guillotine break of a main coolant line. In addition, safety analyses for different leak sizes and locations have to be performed to prove conformance with established acceptance criteria, like the emergency core cooling criteria.

(3) The third tier is represented by the measures of the preventive plant internal accident management. The licensee has to retain pre-planned measures to re-establish the fundamental safety functions. The effectiveness of these measures has to be demonstrated by safety analyses. For such safety analyses, realistic models and boundary conditions can be applied.

(4) Mitigative plant internal accident management can be considered as the fourth tier to achieve practical elimination of large and early releases. Again, the effectiveness of these measures has to be demonstrated by safety analyses applying realistic models and boundary conditions.

(5) The above mentioned four tiers are based on deterministic approaches. Complementary probabilistic safety analysis (PSA) can be considered as the fifth tier. By PSA level 1 and PSA level 2 the achievement of practical elimination can be substantiated. It can be demonstrated, that the implemented design features, periodic testing and in-service inspections, together with preventive and mitigative measures of the plant internal accident management will lead to very low values of large early release frequencies (LERF).

The continuous improvement of German NPPs resulted in a safe fleet of NPPs with highly reliable safety systems and additional safety features for plant internal accident management. Confidence that today's radiological objective for nuclear power plants allowing only limited off-site countermeasures in area and time can be met is achieved by the existing NPPs due to the interaction of safe plant operation, highly reliable safety systems and a comprehensive plant internal accident management programme.

As the responsibility for nuclear safety rests with the licensee by law, a system of independent review of proposed safety improvements by the competent authority consulted with authorized experts, in particular TSOs, is seen as an effective approach to increase trust in the reliability and efficiency of the proposed measures to improve safety. In addition, experts of the RSK provides independent recommendations for further improving nuclear safety in Germany.

6 References

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Annex 1 List of Information Notices (WLN) in the period 1981-2019

WLN-No.	Title
1981/01A	Systematische Schäden an Niveaugebern vom Typ FTC 280 der Firma Endress+Hauser, Maulberg
1981/01B	Systematische Schäden an Niveaugebern vom Typ FTC 280 der Firma Endress+Hauser, Maulberg
1981/02	Schrumpfen von Spulenkörper an Gleichstromschützen
1981/03	Versagen von motorgetriebenen Plattenschiebern, wenn diese gegen Differenzdruck schließen müssen
1982/01	Folge eines Containmentabschlusses zu einem Leck im Volumenregelsystem
1982/02	Defekte am Notstromdiesel (SACM)
1982/03	Versehentliche Unterbrechung der Batteriespeisungen
1982/04	Versagen von Vorsteuerventilen von FD-Sicherheitsventilen
1982/05	Ausfall von Schützen nach Langzeitumschaltung
1982/06A	Störfall in Ginna
1982/06B	Störfall in Ginna - NRC Report (englisch)
1982/07	Feststellung eines Steuerstabes bei Einfahranregung mit nachfolgender RESA und Durchdringungsabschluß
1982/08	Blitzeinschlag in den 220-kV-Hauptnetzanschluß
1982/09	Netzausfall in Nord- und Mittelbelgien
1982/10	Leck in einer HD-Turbinenanzapfleitung
1982/11	Abriß einer Leitung in Not- und Nachkühlsystem
1982/12	Schäden an Steuerleitungen der Frischdampf-Sicherheitsventile
1982/13	Defekte Grenzsignalgeber der Notstromdieselinstrumentierung
1982/14	Kontaktfehler im Relaissockel einer Reaktorschutzzeitstufe
1982/15	Fehlende Löschdioden bei Schützen des 6-Kontaktsystems
1983/01	Leck im Volumenregelsystem
1983/02	Schäden an den Kolben-Rückschlagventile in den Saugleitungen des Nachkühlsystems
1983/03A	Ausfall der automatischen Schnellabschaltung
1983/03B	Ausfall der automatischen Schnellabschaltung - Untersuchung möglicher Konsequenzen auf deutsche Anlagen
1983/04	Pleuellagerschaden an einem Notstromdiesel
1983/05	Ausfall eines Notstromdiesels wegen zu hoher Kühlwassertemperatur
1983/06	Lagerschaden an einer Nachkühlpumpe
1983/07	Heißrisse an austenitischen basissicheren Rohrbögen
1983/08	Unfall mit Todesfolge am Ra-2 Reaktor im argentinischen Forschungsreaktor Constistuyentes
1983/09	Brand in einem Aktivkohlebehälter der Abgasanlage
1983/10	Leckage an einer Vorsteuerleitung eines Druckhalter-Sicherheitsventils
1983/11	Reaktorschnellabschaltung nach Ausfall einer Hauptkühlmittelpumpe
1984/01	Offenbleiben eines Sicherheits-Entlastungsventils
1984/02	Nichtöffnen von Frischdampf-Sicherheitsventilen
1984/02A	Nichtöffnen von Frischdampf-Sicherheitsventilen
1984/02B	Probleme mit den Vorsteuerarmaturen des Sicherheitsventils des Druckhalters
1984/03	Ausfall von Vorsteuerarmaturen aufgrund von Verklammerungs- oder Verklebungseffekten
1984/04	Funktionsstörung am Frischdampfschieber
1984/05	Frischdampf-Sicherheitsventile: Nichtöffnen bzw. Nichtschließen von Vorsteuerventilen aufgrund korrosiver Beläge
1984/06	Bruch von Kolbenringen an den Führungsbuchsen von kombinierten Impulssteuerventilen
1984/07	Risse an Schrauben der Brennelementkastenbefestigung
1984/08	Abgebrochene Schaltnocken an Reservestellantrieben
1984/09	Schäden an Pleuellagerschalen eines Notstromdiesels (SACM)
1984/09A	Schäden an Pleuellagerschalen eines Notstromdiesels (SACM)
1984/10	Abriß eines Ventils im Zwischenüberhitzer-Kondensatsystems
1985/01	Absturz eines Brennelementes während der Handhabung mit dem BE-Einfachgreifer
1985/02	Schäden an Batteriegefäßen im Kernkraftwerken der Bundesrepublik Deutschland und in amerikanischen Anlagen
1985/03	Ausfall von Leistungsverteilungsdetektoren
1985/04	Trennung und Blockierung des Laufrades einer Hauptkühlmittelpumpe
1985/04A	Ergänzungsbericht - Trennung und Blockierung des Laufrades einer Hauptkühlmittelpumpe
1985/05	Wanddickenschwächung der Dampferzeuger-Heizrohre und Schäden an den Halteverbänden
1985/06	Defekte Spaltkammerverstärker in der Neutronenflußinstrumentierung
1985/07	Ausfall der Haupt- und Notspeisewasserversorgung
1985/08	Bruch einer Speisewasserleitung
1985/09	Versagen von 220-V-Gleichstromhilfsschützen
1985/10	Versagen der Einschaltbereitschaft von Sicherungsautomaten
1985/10A	Berichtigung - Versagen der Einschaltbereitschaft von Sicherungsautomaten
1985/11	Ausfall einer Sicherheitseinspeisepumpe bei einem Pumpenprobelauf
1985/12	Unbeabsichtigtes Öffnen der Frischdampf-Umleitstation und Gemischaustrag in die Frischdampfleitungen

1986/01	Ausfall des integrierten Blockregelsystems
1986/02	Versagen von Vorsteuerventilen von Sicherheits-/Entlastungsventilen in finnischen und schwedischen SWR-Anlagen
1986/02A	Probleme mit Vorsteuerventilen des Druckentlastungssystems in neueren schwedischen Siedewasserreaktoren
1986/02B	Nichtöffnen von Vorsteuerventilen im Druckentlastungssystem
1986/03	Unterdimensionierte Anschlußleitungen in Motoren der HD-Förderpumpen
1986/04	Fehlerhaftes Absicherungskonzept von Stellungsgebern an den Vorsteuerarmaturen der Sicherheits- und Entlastungventile (06.06.1986)
1986/05	Unvollständige Füllung der Saugeleitung des TH-Systems
1986/06	Schrauben der Kernumfassung defekt
1986/07	Hauptkühlmittelpumpen-Wellenbruch
1987/01	Bruch einer Hauptkühlmittelpumpenwelle
1987/02	Schäden an AEG-Leistungsschaltern
1987/03	Bruch der Speisewasserleitung
1987/03A	Erosion der Speisewasserleitung
1987/04	Offenbleiben eines Sicherheits- und Entlastungsventils
1987/05	Verminderung des Nebenkühlwasserdurchsatzes
1987/06	Spannungsrisskorrosion an Vorsteuerleitung S/E-Ventile
1987/07	Lose Distanzstücke im Generatorläufer eines Notstromdieselaggregates
1987/08	Störungen an ISKAMATIC-Vorrangbaugruppen
1987/09	Verformung an Ventileinbauten der SI/E-Ventile
1987/10	Unscharfschaltung automt. Notstromdieselstart
1987/11	Messung der Abschaltsicherh. in Oskarshamn 3
1987/12	Borsäurekorrosion in Arkansas Nuclear 1
1987/13	Verbacken von Sicherheitsventilen im Sekundärkreis
1987/14	Funktionsstörung Notstromschaltanlage
1987/15	Ausfall eines Stellantriebes
1987/16	Störung der Vorsteuerventile der FSA-Station
1987/17	Verlängerte Schließzeit Lüftungsklappen
1987/18	Verformung der Ventileinbauten von SI/E-Ventile
1988/01	Schrumpfen von Schütz-Spulenkörpern
1988/02	Defektes Steuerventil für TH-Iso-Armatur
1988/03	Fehlerhafte Taktgeberbaugruppe
1988/04	Offene Erstabsperrung im TH-System
1988/05	Risse in Hauptkühlmittelpumpenwelle
1988/06	Leistungsschwankungen bei Naturumlauf
1988/06A	Leistungsschwankungen bei Naturumlauf
1988/07	Durchgehender Riss in einer nicht absperrbaren Rohrleitung des Not-und Nachkühlsystems
1988/08	Ausfall Kondensator Reaktorschutzsystem
1988/09	Unzureichende Selektivität Notstromversorgung
1988/10	Defekte Schrauben an FD Isolationsventilen
1988/11	Defekter Endschalter am Frischdampfschieber
1988/12	Schaden an Kupplung Notspeisedieselaggregat
1988/13	Systematische Fehler Druckmessumformer Teleperm
1988/14	Fehlende Teile Kupplung Notspeisedieselaggregate
1989/01	Fehlerhafte leittechnische Schrankeinspeisung
1989/02	Riss in Welle der Hauptkühlmittelpumpe
1989/02A	Ergänzung zur WL 1989/02 bezüglich frequenzselektiver Überwachung von DWR-Pumpenwellen
1989/03	Instabiles Verhalten der Sicherheitsventile
1989/04	Nichtgekuppelter Steuerstabantrieb
1989/05	Spannungskorrosion an Motorgehäusen
1989/06	Sicherheitsventil und Abblaseventil, Vorsteuerung
1989/07	Ausfall Notstromversorgung, Nachwärmeabfuhr
1989/08	Absturz Brennelement im Lagerbecken
1989/09	Verlängerte Schliesszeit von Radialschiebern
1989/10	Fehlerhafte NH-Sicherung, während Revision 1989
1989/11	Störung am Manipulatormast der Brennelementbühne
1989/12	Überlastete Sicherungen im THTR-300 und KWB-A
1989/13	Startversagen eines Notstromdiesels bei einer Funktionsprüfung
1989/14	Mangelhafte Lötverbindungen auf Iskamatic-A-Baugruppen ASS11 im KKU und KBR
1990/01	Brennelementschäden durch Siedeübergang (dryout)
1990/02	Befunde an Kernbehälterschrauben

1990/03	Defekte Aktivitätsmesstellen im GKN-2 und KMK
1990/04	Ausfall einer gesicherten Drehstromschiene durch Öffnen des Leistungsschützes
1990/05	Ungeplantes Ansprechen eines Druckhalter-Abblaseventils bei einer Wiederkehrenden Prüfung
1990/06	Klemmen eines Betätigungsmagneten der Sicherheits- und Entlastungsventile bei Prüfung im Block B und im Block C
1990/07	Kontaktprobleme an FT-Baugruppen
1990/08	Bruch einer Speisewasserleitung
1990/08A	Bruch einer Speisewasserleitung
1990/09	Funktionsstörungen an Armaturen mit Stellantrieben durch ungewolltes Ansprechen von Drehmomentendschaltern
1990/10	Schäden an Steuerstabsführungsrohren
1990/11	Zusammenfassung von 2 Vorkommnissen, die durch elektromagnetische Störbeeinflussung hervorgerufen wurden
1990/12	Befunde an Reaktordruckbehälterdeckeln von zwei Siedewasserreaktoren in den USA
1990/12A	Befunde an Reaktordruckbehälterdeckeln von zwei Siedewasserreaktoren in den USA
1990/13	Gebrochene Schrauben der Deckplattenbefestigung der Tragplatte des oberen Kerngerüstes
1990/14-1	Stellkraftreserven von Absperrschiebern - Ergebnisse von amerikanischen Großversuchen
1990/14-2	Stellkraftreserven von Absperrschiebern - Literaturverweise
1990/14A	Funktionsstörungen an Absperrschiebern im Nachkühlsystem
1990/15	Ausfall von zwei Gleichrichterpaaren für die Versorgung einer 24-V-Schaltanlage während der Revision
1990/15A	Ausfall von zwei Gleichrichterpaaren für die Versorgung einer 24-V-Schaltanlage während der Revision
1990/16	Defekte Abgaskompensatoren an Notstromdieseln (KKB und KKK)
1991/01	Korrosionsbefunde am Sicherheitsbehälter im Einspannbereich
1991/02	Nichtöffnen einer Rückschlagklappe in einem Nebenkühlwasserstrang
1991/03	Einwirkungen auf sicherheitsrelevante Einrichtungen aufgrund von Interferenzen durch Funksprechgeräte
1991/04	Mögliches Fehlumschalten der Auswahlschaltung in der Einspeiseleitung der Sicherheitshochdruckeinspeisepumpen auf die kalte Einspeisung
1991/05	Ergebnisse neuerer Untersuchungen zu Primärkreisleckagen über Niederdrucksysteme außerhalb des Sicherheitsbehälters
1991/06	Wasserstoffbrand nach Leitungsabriß im Reaktorgebäuderingraum
1991/07	Schaden an einem Absperrventil im Dampferzeugerabschlämmsystem
1991/08	Defekte Befestigungsschrauben an Paßplatten der RDB-Tangentialabstützung
1991/09	Sitzleckagen an den Vorsteuerventilen der Primärkreis-Sicherheitsventile
1992/01	Fehleinfahrten von Steuerstäben in den unbeladenen Reaktor
1992/02	Schäden an Getriebezahnrädern von Zwischengetrieben für Absperrklappen
1992/03	Ausfall eines 24 V Gleichrichters im GKN I und KKB
1992/04	Befunde an Schweißnähten der Saugleitung des Nachkühlsystems
1992/04A	Befunde an der Schweißnaht ZR 112 WN 3 der Treibwasserschleife I
1992/04B	Befunde an Schweißnähten von austenitischen Rohrleitungen
1992/05	Ausfall der Hauptwärmesenke durch Fehlansprechen eines Lastsprungrelais
1992/06	Befunde an Elektrolytkondensatoren auf Reaktorschutz-Grenzwertmelderbaugruppen RG 11
1992/07	Leckage an den Durchführungen der Kerninstrumentierungslanzen
1992/08	Nichtöffnen von Armaturen des Feuerlöschwassersystems
1992/09	Fehler an Schutzrelais von Leistungsschaltern in den Notstromanlagen
1992/10	Defekte Arretierungshilfen der Steuerabführungrohre
1992/11	Rißbefund an je einem Zylinderkopf von zwei Notstromdieselmotoren
1992/12	Fehlfunktion einer Bypaßklappe im nuklearen Zwischenkühlsystem
1992/13	Erkenntnissen über mögliche Überlastungen des Abblasebehälters beim Ansprechen der Druckhalterarmaturen
1992/14	Verstopfen der Sumpfansaugöffnungen der Notkühlsysteme infolge Fehlöffnens eines Sicherheitsventils
1992/14A	Verstopfen der Sumpfansaugöffnungen der Notkühlsysteme infolge Fehlöffnens eines Sicherheitsventils
1992/14B	Verstopfen der Sumpfansaugöffnungen der Notkühlsysteme infolge Fehlöffnens eines Sicherheitsventils
1992/14C	Verstopfen der Sumpfansaugöffnungen der Notkühlsysteme infolge Fehlöffnens eines Sicherheitsventils im Kernkraftwerk Barsebäck-2 (Schweden) am 28.07.1992
1992/15	Fehlsignale in Reaktorschutz- und Begrenzungseinrichtungen durch Einkopplung von Störimpulsen beim Einsatz eines Kabelmeßwagens
1992/16	Abschalten eines 24-V-Gleichrichters infolge einer Netzstörung
1993/01	Schäden an Frischdampf-Isolationsventilen (SWR) infolge Öffnungsvorgängen mit hoher Öffnungsgeschwindigkeit im KKB und KKP-1
1993/02	Defekte Tellerfedern in Radialschiebern des Lagerdruckwassersystems und der Reaktorwasserreinigung
1993/03	Störung in der zentralen Gasversorgung
1993/04	Defekte Keramik-Kondensatoren auf Trennverstärkerbaugruppen und Grenzwertmeldern
1993/05	Unkorrekter Einsatz von Analog-Trennwandlerbaugruppen
1993/06	Nichtöffnen eines Hauptspeisewasserabsperrschiebers
1993/07	Befund am Stellantrieb der Abdampfarmatur des Hochdruck-Einspeisesystem
1993/08	Funktionsstörungen an Armaturen des nachgerüsteten diversitären Druckbegrenzungssystems bei einer wiederkehrenden Prüfung im KKI-1 und KKP-1

1994/01	Funktionsstörung in der Mechanik der Schmelzlotauslösung von Brandschutzklappen im KKP-2 und KKI-2
1994/01	Malfunction of the Thermal Actuation Mechanism of Fire Dampers
1994/01A	Funktionsstörungen in der Mechanik der Schmelzlotauslösung von Brandschutzklappen im KKP-2 und KKI-2
1994/01B	Funktionsstörungen in der Mechanik der Schmelzlotauslösung von Brandschutzklappen im KKP-2 und KKI-2
1994/01C	Funktionsstörungen an den Fernbedienungen von Brandschutzklappen im KKP-2
1994/01D	Funktionsstörungen an den Fernbedienungen von Brandschutzklappen im KKP-2
1994/01E	Mängel an neu eingebauten Brandschutzklappen
1994/02	Fälschliches Ansprechen des Kurzschlußschutzes beim Einschalten von Niederdruck-Nachkühlpumpen
1994/02	False actuation of the short-circuit protection during switch-on of low-pressure safety injection pumps
1994/03	Ausfall eines 0,4-kV-Leistungsschalters bei der wiederkehrenden Prüfung eines Notstromdieselaggregates
1994/04	Ausfall von Druckspeicher-Füllstandsmessungen im KKE und GKN-2
1994/05	Ausfall einer Verstärker-Baugruppe des Reaktorschutzsystems
1994/06	Fehlöffnen eines 10-kV-Leistungsschalters bei einer Wiederkehrenden Prüfung
1994/07	Eintrag verdünnter Schwefelsäure in den Wasser-Dampf-Kreislauf
1994/08	Störung am Entregungsschalter eines Notstromdiesels
1994/08A	Störung am Entregungsschalter eines Notstromdiesels und Nichtzuschalten eines Entregungsschalters bei einer Notstromprüfung
1994/09	Unvollständige Dichtheit von zwei Absperrventilen im Not- und Nachkühlsystem bei einer wiederkehrenden Prüfung
1994/09A	Funktionsstörungen an absperrbaren Druckspeicher-Rückschlagventilen im GKN-2 und KKE
1994/10	Temperaturtransiente im Reaktordruckbehälter nach einem Ausfall der Zwangsumwälzpumpen
1994/11	Befunde an Kernmänteln von Siedewasserreaktoren
1994/12	Nicht vollständiger Einfall eines Steuerstabes nach einer Reaktorschnellabschaltung
1994/13	Eintrag von nicht kondensierbaren Gasen in Siede- und Druckwasserreaktoren
1995/01	Kühlmittelleckage an einer Einspeiseleitung des Volumenregelsystems
1995/01A	Kühlmittelleckage an einer Einspeiseleitung des Volumenregelsystems
1995/01B	Kühlmittelleckage an einer Einspeiseleitung des Volumenregelsystems
1995/02	Schaden an einer Wirkdruckleitung zur Erfassung des Sicherheitsbehälterdruckes
1995/03	Zu hoher Ansprechdruck von Frischdampf-Sicherheitsventilen
1995/04	Vertauschung der Stellungsrückmeldung an den Einspeiseschaltern der doppelt versorgten 380-V-Notstromschienen
1995/05	Schaden an einem Turbolader eines Notstromdiesels
1995/06	Rißbefunde in der Stellitierung der Kegellaufbuchsen von Frischdampf-Abblaseregelventilen
1995/07	Lagerschaden an einer HD-Förderpumpe (KBA33 AP001)
1995/08	Defekte Hüllrohre an bestimmten Brennelementen
1995/09	Tropfleckage am Armaturengehäuse 12TH22 S006
1995/10	Batteriezellenschädigung durch Plattenkorrosion
1995/10A	Kapazitätsminderungen an 220-Volt- und 24-Volt-Batterien in den Jahren 1997/1998
1995/11	Nicht erfolgtes automatisches Wiederzuschalten von Gleichrichtern im Notstromsystem
1996/01	Reaktorschutz-Signalauslösung nach Ansprechen einer Sicherung während einer Wiederkehrenden Prüfung
1996/02	Nichtöffnen von Absperrschiebern bei wiederkehrenden Prüfungen infolge eingeschlossenen Mediums im KKB und KWG
1996/03	Fehlöffnen des Druckhalter-Abblaseventils
1996/04	Gebrochene Kolbenringe in den Magnet-Vorsteuerventilen der Frischdampf-Isolationsventile im KKI-1 und KKB
1996/05	Ausfall eines 24-V-Gleichrichters mit nachfolgenden Auswirkungen in der Notstromversorgung
1996/06	Störung an einem 10-kV-Leistungsschalter bei einer WKP
1996/07	Leckage an einer Referenzsäule der Reaktordruckbehälter-Füllstandsmessung
1996/08	Abschaltung eines Notstromdiesels bei einer WKP durch Ansprechen des Aggregateschutzes
1996/09	Ausfall einer Aktivitätsmeßstelle
1996/10	Verzögertes Ansprechen der Druckhalter-Abblasesteuerventile bei einer wiederkehrenden Prüfung
1996/11	Verlängerte hydraulische Totzeit an Magnetvorsteuerventilen durch Korrosionsvorgänge aufgrund des Eintrags von Schmiermitteln
1997/01	Gelöste Schraubensicherungen in den Magnet-Vorsteuerventilen der Speisewasser-Rückschlagventile
1997/02	Ausfall von Baugruppen zur Stromversorgung von Meßumformern
1997/03	Nichtöffnen eines Absperrschiebers im Einspeisesystem
1997/04	Turboladerschaden an einem Notspeisenotstromdiesel bei wiederkehrender Prüfung
1997/05	Störung an einem Einspeiseschieber im Nachkühlsystem
1997/06	Fehlöffnen eines Sicherheits- und Entlastungsventils
1997/07	Vertauschung von Unterspannungsrelais
1997/08	Kontaktbeläge in Schaltern der Steuerstabversorgung
1997/08A	Kontaktbeläge an Paketschaltern
1997/09	Funktionsbeeinträchtigung von Reibbremsen an federbelasteten Sicherheitsventilen
1997/10	Fehlmontage einer Trennklemme am Leistungsschalter einer Beckenkühlpumpe
1997/11	Unvollständiges Öffnen einer Armatur durch große Schalthysterese der Drehmomentüberwachung
1998/01	Korrosionsangriff auf den Außenoberflächen von sicherheitsrelevanten austenitischen Rohrleitungen

1998/02	Schäden an Steuerstäben in den japanischen Kernkraftwerken Tsuruga-1 und Fukushima-1 (Siedewasserreaktoranlagen)
1998/03	Nichtverfügbarkeit von 1 von 4 Frischdampf-Sicherheitsarmaturen-Stationen (FSA) bei Anforderung im Kernkraftwerk Unterweser (KKU) am 06.06.1998
1998/03A	Nichtverfügbarkeit von 1 von 4 Frischdampf-Sicherheitsarmaturen-Stationen (FSA) bei Anforderung im Kernkraftwerk Unterweser (KKU) am 06.06.1998
1998/04	Überprüfung der Software mit sicherheitstechnischer Bedeutung im Hinblick auf die Datumsumstellung zur Jahrtausendwende
1998/04A	Überprüfung und gegebenenfalls Ertüchtigung der Software mit sicherheitstechnischer Bedeutung auf Störungsfreiheit bei der Datumsumstellung zur Jahrtausendwende und anderen kritischen Zeitumstellungen
1998/05	Unvollständiges Schließen einer Armatur durch überhöhte Stopfbuchsreibung bei wiederkehrender Prüfung
1998/06	Kühlmittelverlust durch eine Leckage im Nachkühlsystem
1998/07	Lose Sicherheitsmuttern an Steuerstabantriebsgehäuserohren in der Revision im Juni 1998
1998/08	Leck in einem Dampferzeuger-Heizrohr in der Anlage Grafenrheinfeld und Anzeigen an Dampferzeuger-Heizrohren in der Anlage Unterweser
1998/09	Fehlfunktion von Armaturen aufgrund des Einsatzes der Vorrangbaugruppe AV17
1998/10	Abgerissene Stiftschrauben am Drehzahlreglerantrieb von MTU-Notstromdieselmotoren
1998/11	Schaden an einem Aktivkohlefilter des nuklearen Abluftsystems
1998/12	Einschaltversagen eines 10-kV-Leistungsschalters
1999/01	Nichtverfügbarkeit eines Teil-Reaktorschutzsystems aufgrund des Ausfalls der 24-V-Gleichstromversorgung
1999/02	Einschaltversagen eines Leistungsschalters aufgrund eines defekten Hilfsschützes
1999/02A	Einschaltversagen eines Leistungsschalters aufgrund eines defekten Hilfsschützes
1999/02B	Wiederholtes Einschaltversagen von Leistungsschaltern aufgrund defekter Hilfsschütze in der Zeit vom 25.11.1998 bis 12.08.1999
1999/02C	Einschaltversagen eines Leistungsschalters aufgrund eines defekten Hilfsschützes im Kernkraftwerk Brunsbüttel am 17.10.2009
1999/03	Ventilfehlstellungen an den Meßumformern der Füllstandsmessung zweier Druckspeicher
1999/04	Kleinstleckage eines Dampferzeuger-Heizrohres
1999/05	Ausfall eines Notstromdiesels aufgrund eines defekten Öldruckwächters
1999/06	Ausfall der Spannungsregelung eines Notstromdieselgenerators
1999/07	Reaktorschnellabschaltung nach einem geplanten Lastabwurf auf Eigenbedarf
1999/07A	Reaktorschnellabschaltung nach einem geplanten Lastabwurf auf Eigenbedarf
2000/01	Mechanische Verformung der Antriebsstange eines Steuerelementes
2000/02	Schäden in der Stellungsüberwachung von Rückschlagklappen
2000/03	Verlängerte Schließzeit einer Gebäudeabschlußklappe
2000/04	Explosion in einer Sauerstoff-Versorgungsleitung
2000/05	Nichtschließen einer Primärkreisabschlußarmatur bei einer wiederkehrenden Prüfung
2000/06	Fertigungsfehler bei Manometer-Prüfventilen
2000/07	Bruch einer Steuerleitung im Turbinenschutz-Niederdruck-Bypass-System
2000/08	Fehlfunktion von Meßumformer-Versorgungsbaugruppen
2000/09	Defekte Profilhülsen in der Kupplung von Nebenkühlwasserpumpen
2000/10	Nicht versandt (Blayais)
2000/11	"Kurzzeitige Spannungslosigkeit einer Notstromschiene" im Kernkraftwerk Obrigheim (KWO) am 24.06.2000
2000/12	"Partieller Ausfall eines Reaktorschutz-Kettengliedes" im Kernkraftwerk Philippsburg 2 (KKP-2) am 30.09.1999
2000/13	"Fehlerbedingte sekundärseitige Lastabsenkung und nicht erfolgter Stabeinwurf" im Gemeinschaftskernkraftwerk Neckar, Block 1, am 10.05.2000
2000/14	"Korrosionsbeläge an Vorsteuerventilen der Druckhalter-Sicherheitsventil im Kernkraftwerk Grafenrheinfeld, erkannt am 30.06.2000
2001/01	Rissbefunde in einer Mischnaht am Stutzen einer Nachkühlleitung im Kernkraftwerk Biblis, Block A, entdeckt am 12.10.2000
2001/02	Verunreinigte Messleitungen im Feuerlöschsystem im Kernkraftwerk Gundremmingen II, Block B, am 3.11.2000
2001/03	Nicht spezifikationsgerechte Schraubenverbindungen an Sicherheitsventilflanschen in den Kernkraftwerken Philippsburg, Block 1, am 18.12.1999 und Block 2, am 16.02.2000
2001/04	Rissbefunde am Austrittsstutzen der Nachkühl-Saugearmatur TH02 S001 (Erstabsperrarmatur) und in dem anschließenden Rohrleitungsteilstück im Kernkraftwerk Stade am 09. März 2001
2001/05	Schäden an Mischnähten der Reaktordruckbehälterstutzen in den Kernkraftwerken Virgil C. Summer (USA) und Ringhals 4 (Schweden) entdeckt im Herbst 2000
2001/06	Versagen von Ventilen im Feuerlöschsystem in den Kernkraftwerken Biblis, Block A, am 29.11.2000 und Gundremmingen, Block B, am 15.03.2001
2001/07	Absturz eines Brennelementes im Kernkraftwerk Krümmel am 06.04.2001
2001/08	Unterschreitung des Sollfüllstands in vier Flutbehältern beim Anfahren am 10.08.2001 und während des nachfolgenden Leistungsbetriebs am 27.08.2001 erkannte zu geringe Borkonzentration in drei Flutbehältern im Kernkraftwerk Phillipsburg, Block 2
2001/09	Tiefentladung einer Batterie der unterbrechungslosen Gleichstromversorgung im Kernkraftwerk Philippsburg 2 (KKP-2) am 01.08.2001
2001/10	Befund an einer Rohrleitungshalterung im Schnellabschaltsystem im Kernkraftwerk Philippsburg,Block 1, am 26.11.2000
2002/01	Whiskerbildung an leittechnischen Baugruppen in deutschen Kernkraftwerken
2002/01A	"Whiskerbildung an leittechnischen Baugruppen in deutschen Kernkraftwerken"
2002/02	"Absturz eines Brennelementes beim Beladen des Transportbehälters" im Kernkraftwerk Biblis,Block B, am 06.08.2001,

2002/03	"Reaktorschnellabschaltung der Anlage durch Neutronenfluss LD > 120%" im Kernkraftwerk Philippsburg-1 (KKP-1) am 23.11.2001
2002/04	"Bruch der Deckelsprühleitung" im Kernkraftwerk Brunsbüttel am 14.12.2001
2002/05	"Lastabsturz infolge Versagen des Hubwerks des Rundlaufkrans im Sicherheitsbehälter in der kompakten natriumgekühlten Kernreaktoranlage Karlsruhe (KNK-II)" am 16.11.2001
2002/06	"Ausfall der Blockeinspeisung" im Kernkraftwerk Grafenrheinfeld (KKG) am 02.04.2002
2002/07	"Fehlöffnen von Schaltern an einer 400-V-Notstromschiene bei einer wiederkehrenden Prüfung und Anzugsversagen von Hilfsschützen" im Kernkraftwerk Philippsburg, Block 1, am 21.05.2002
2002/08	"Mängel an störfallfesten Stellantrieben" im Kernkraftwerk Biblis, Block B, vorgefunden am 30.08.2002, im Kernkraftwerk Emsland, vorgefunden am 02.10.2002 und im Kernkraftwerk Neckar, Block 1, am 11.10.2002
2003/01	Erhöhter Ansprechdruck von Gehäusebruchsicherungen an Gebäudeabschlussarmaturen der Ölversorgung der Hauptkühl-mittelpumpen im Kernkraftwerk Stade, vorgefunden am 23. bis 27. Mai 2002
2003/02	Große Korrosionsmulde im Reaktordruckbehälter-Deckel des Kernkraftwerkes Davis Besse (USA)
2003/03	Kontamination innerhalb des Überwachungsbereiches im Kernkraftwerk Philippsburg, Block 1, am 24.09.2002
2003/04	Lösen eines aufblasbaren Dichtstopfens in einer Frischdampfleitung durch zu hohen Druck in einem Dampferzeuger im Kernkraftwerk Gentilly-2, Kanada, am 12. April 2001
2003/05	Abriss eines Wärmeschutzrohres am Stutzen des nuklearen Nachwärmeabfuhrsystems im Kernkraftwerk GKN-2, entdeckt am 24.08.2002
2003/06	Nichtschließen eines Absperrventils in einer Frischdampf-Anwärmleitung bei wiederkehrender Prüfung im Kernkraftwerk Brokdorf am 21.05.2001
2003/07	Befunde an Speisewasserstutzen der Dampferzeuger im Kernkraftwerk Unterweser, entdeckt am 13.11.2002
2003/08	Fehler in der Steuerung der Notstromversorgung und der Not- und Nachkühlsysteme im Kernkraftwerk Brunsbüttel, erkannt am 17.07.2002, 27.08.2002 und am 12.09.2002
2003/09	Schäden an Gehäuseschrauben von Freilaufrückschlagventilen im Wasserabscheider-Kondensatsystem im Kernkraftwerk Krümmel, am 12.11.2001
2003/10	Brand in einer 500-V-Schaltanlage des UNS-Systems im Kernkraftwerk Stade (KKS) am 11.08.2002
2003/11	Aushärtung von Dämpfungsmasse in Viskosedämpfern vorgefunden im Kernkraftwerk Emsland am 7.11.2001 und weiteren Kernkraftwerken
2003/12	Funktionsstörung einer Grenzwertmelderbaugruppe im Kernkraftwerk Brunsbüttel (KKB) am 18.12.2002
2003/13	Ausfall der Brandmeldeanlage Reaktorgebäude aufgrund einer Störung in der Spannungsversorgung im Kernkraftwerk Krümmel (KKK) am 12.03.2002
2003/14	Durch Netzstörungen verursachte Abschaltungen von Gleichrichtern im Kernkraftwerk Brokdorf (KBR) am 23.02.2002
2004/01	"Korrosion an Komponenten des Kühlwasserkreislaufs von Dieselaggregaten" in den Kernkraftwerken Grohnde (KWG) und Neckar II
2004/02	"Aufgetretene Fehler bei Schaltvorgängen von Leistungsschaltern in Kernkraftwerken der Bundesrepublik Deutschland"
2004/02a	„Fehlende Betriebsbereitschaft von Leistungsschaltern in Kernkraftwerken der Bundesrepublik Deutschland", Ergänzung zur WLN 2004/02
2004/03	"Ausfall der Eigenbedarfs- und Notstromversorgung für 2 Stunden" im taiwanesischen Kernkraftwerk Maanshan-1 am 18. März 2001
2004/04	"Befunde an nuklearen Zwischenkühlern" im Kernkraftwerk Unterweser, entdeckt am 20.11.2002
2004/05	"Lagerschaden an der Sperrwasserpumpe 24RY42 D001 infolge eines niedrigen Ölfüllstands" im Kernkraftwerk Biblis, Block B, vorgefunden am 03.04.2003
2004/06	"Dampfleckage an einer Messleitung an einem Dampferzeuger" im Kernkraftwerk Biblis, Block A" am 08.02.2004
2004/07	"Befund an einem Sicherheitsventil im Volumenregelsystem in der Revision" im Kernkraftwerk Philippsburg-2 am 28.07.2002
2004/08	"Abweichungen von den Vorprüfunterlagen an Absperrarmaturen Speisewassersystem" im Kernkraftwerk Philippsburg, Block 2, vorgefunden am 25.07.2003
2004/09	"Potenzieller Einfluss des Chloridgehalts auf das Spannungsrisskorrosionsverhalten ferritischer Stähle unter Heißwasserbedingungen
2004/10	"Schäden an Schalldämpfern der Dieselmotoren der Notstromversorgung in den Kernkraftwerken Emsland, Brokdorf und Stade"
2004/11	"Sporadische Funktionsstörung in der Leittechnik des Masthubwerks" im Kernkraftwerk Brunsbüttel am 19.07.2002
2004/11A	"Abweichungen im Betrieb der Brennelementwechselbühne" im Kernkraftwerk Brunsbüttel
2004/12	"Mängel an ölarmen 10-kV-Leistungsschaltern in mehreren Kernkraftwerken der Bundesrepublik Deutschland"
2005/01	"Schaltversagen von Koppelschützen im Kernkraftwerk Brokdorf am 08.12.2003 und im Kernkraftwerk Krümmel am 11.03.2004"
2005/02	"Kleinstleckagen an den Gehäusedeckeldichtungen der Druckhalter-Handabsperrventile" im Kernkraftwerk Brokdorf am 19.07.2003
2005/02A	"Kleinstleckagen an den Gehäusedeckeldichtungen der Druckhalter-Handabsperrventile" im Kernkraftwerk Brokdorf am 19.07.2003
2005/03	"Fehlende Erdbebenverstiftung an sicherheitstechnisch wichtigen Komponenten" in mehreren Kernkraftwerken und Mängel in der Qualitätssicherung
2005/04	"Fehlende Überbrückung der Drehmomentabschaltung an Stellantrieben" im Kernkraftwerk Biblis, Block B, vorgefunden am 09.05.2004
2005/05	"Schäden an den Reaktordruckbehälter-Deckelentlüftungsleitungen in den Kernkraftwerken Neckar II und Brokdorf"
2005/06	"Mikrobiologisch induzierte Korrosion an Komponenten in Nebenkühlwassersystemen von Kernkraftwerken"
2005/07	"Wirbelstromanzeigen an Heizrohren im Rohrbodenbereich von Dampferzeugern" im Kernkraftwerk Biblis, Block A am 10.04.2005
2005/08	"Leckagen an den Einspritzleitungen der Notstromdieselmotoren" in den Kernkraftwerken Grohnde am 03.11.2003 und Biblis A am 27.04.2004
2005/09	"Ausfall eines Notstromtransformators bei Langzeitumschaltung im Rahmen einer wiederkehrenden Prüfung" im Kernkraftwerk Krümmel am 20.09.2004

2005/09A	"Ausfall eines Notstromtransformators bei Langzeitumschaltung im Rahmen einer wiederkehrenden Prüfung" im Kernkraftwerk Krümmel am 18.09.2004
2005/10	"Funktionsstörungen an 10-kV-Leistungsschaltern" im Kernkraftwerk Gundremmingen II, Block C (KRB-II-C) am 24.04.2002 und 26.07.2004
2005/11	"Kurzschluss in einem 10-kV-Kabel der Eigenbedarfsversorgung" im Kernkraftwerk Brunsbüttel, aufgetreten am 23.08.2004
2005/12	"RESA über Frischdampfdruck > max. bei 17 % Reaktorleistung" im Kernkraftwerk Isar, Block 2, am 18. Juli 2004
2005/13	"Möglichkeit des Fehlfahrens von zwei Steuerstäben aufgrund vertauschter Leistungs- und Rückmeldekabel der Antriebe" im Kernkraftwerk Krümmel am 20. August 2004
2005/14	"Rissanzeigen an Komponenten des Notspeisesystems" im Kernkraftwerk Grafenrheinfeld und weitere Schäden in anderen Anlagen infolge chloridinduzierter transkristalliner Spannungsrisskorrosion
2005/15	"Auslegungsfehler in den Funktionsgruppensteuerungen des Nuklearen Nachkühlsystems TH bei der Wiederaufnahme des Nachkühlbetriebs im Notstromfall" im Gemeinschaftskernkraftwerk Neckar, Block 1, am 19.02.2005
2006/01	"Auslösung der CO2-Löschanlage für einen Rechnerraum mit Beschädigung einer Brandschutztür" im Kernkraftwerk Brunsbüttel am 29.07.2005
2006/02	"Schäden am Mantelrohr von stillgelegten Druckhalter-Heizstäben" im Kernkraftwerk Biblis, Block B, festgestellt am 03.10.2005
2006/03	"Nicht vollständiges Schließen von Gebäudeabschlussarmaturen der Druckluftversorgung bei wiederkehrender Prüfung" im Kernkraftwerk Emsland (KKE) am 04.07.2003
2006/04	"Aktivitätsübertritt vom Dekontsystem für Primärkreiskomponenten in das Deionatsystem mit anschließender Freisetzung" im Kernkraftwerk Neckarwestheim, Block 2, 26.08.2004
2006/05	"Temporäre Störung von Symphony-Baugruppen" im Kernkraftwerk Isar 1 am 26.01.2005
2006/06	"Fehlerhaft montierte Dübel im Kernkraftwerk Biblis, Block A"
2006/06A	"Fehlerhaft montierte Dübel" im Kernkraftwerk Biblis, Block A (KWB-A)
2006/07	Ereignis im schwedischen Kernkraftwerk Forsmark, Block 1 am 25.07.2006: "Nichtzuschalten von zwei Notstromdieseln nach Ausfall der 400-kV-Netzanbindung"
2006/08	"Ausfälle von Drehstromschützen in den Kernkraftwerken Isar 1 und Brunsbüttel"
2007/01	"Befunde an Kernbehälter- und Kernumfassungsschrauben" in den Kernkraftwerken Neckar I und Biblis B
2007/02	"Schäden an Rohrleitungen in Nebenkühlwassersystemen für sicherheitstechnisch wichtige Kühlstellen"
2007/03	"Harzeintrag in den Reaktorkühlkreislauf im französischen Kernkraftwerk Fessenheim-1", am 24. Januar 2004
2007/04	"Ansprechen von Sicherheitsventilen bei der Durchführung der RDB-Druckprüfung mit der Folge des Anrisses einer Impulsleitung" im Kernkraftwerk Krümmel, am 31.08.2005
2007/05	"Ausfall eines Zuluftventilators" im Kernkraftwerk Biblis B am 05.07.2005
2007/06	"Gebrochene Ventilspindeln an Belüftungsventilen hinter einem Sicherheits- und Entlastungsventil sowie eine kurzzeitige Leckage innerhalb des Sicherheitsbehälters" im Kernkraftwerk Philippsburg, Block 1
2008/01	"Funktionseinschränkung an sicherheitstechnisch wichtigen Armaturen aufgrund unvollständiger Spezifikation" im Kernkraftwerk Krümmel, erkannt am 14.08.2006
2008/02	"Kurzzeitige Unverfügbarkeit zweier Sicherheitsteileinrichtungen" im Kernkraftwerk Emsland am 21.11.2006
2008/03	"Risse in austenitischen Armaturengehäusen infolge chloridinduzierter transkristalliner Spannungsrisskorrosion" im Kernkraftwerk Krümmel
2008/04	"Befunde an Dampferzeuger-Heizrohren" im Kernkraftwerk Unterweser
2008/05	"Störung an einer Druckspeicherarmatur" im Kernkraftwerk Unterweser am 25.04.2006
2008/06	"Verzögertes Hochlaufen von Notstromdieseln bei Funktionsprüfungen" im Kernkraftwerk Philippsburg II, am 31. Januar 2006
2008/07	"Eindringen von Brandgasen in die Warte des Kernkraftwerks Krümmel beim Brand eines Maschinentransformators am 28.06.2007"
2008/08	"Mängel in Organisation und Betriebsführung" in mehreren deutschen Kernkraftwerken
2009/01	Reaktorschnellabschaltung durch kurzzeitigen Ausfall der Eigenbedarfsversorgung aufgrund eines Kurzschlusses in einem Maschinentransformator im Kernkraftwerk Krümmel am 28.06.2007
2009/02	"Leckagen infolge transkristalliner Spannungsrisskorrosion an den Außenoberflächen der Zuleitungen zu den Steuerstabantrieben" im Kernkraftwerk Cofrentes (Spanien)
2009/03	"Vertauschen von Meldebaugruppen in der Geamatic-Steuerung" im Kernkraftwerk Brunsbüttel, erkannt am 19. September 2005
2009/04	"Versagen eines Hebezeuges" im Kernkraftwerk Gundremmingen Block C am 03. Oktober 2008
2009/05	Ereignis im schweizerischen Kernkraftwerk Leibstadt am 06.03.2007: "Fehlerhafte Auslösung des automatischen Druckentlastungssystems ADS"
2009/06	Ausfall einer Zeitüberwachungsbaugruppe im dynamischen Logikteil des Reaktorschutzes im Kernkraftwerk Krümmel am 15.10.2007
2010/01	"Anrisse im Dichtungsgehäuse der Hauptkühlmittelpumpe" im Kernkraftwerk Biblis, Block B
2010/02	Meldepflichtiges Ereignis im Kernkrafterk Brunsbüttel am 07.08.2009 "Schaden an einem Notstromdieselaggregat nach Wartung"
2010/03	Reaktorschnellabschaltung über niedrigen Dampferzeugerfüllstand aus Teillast im Kernkraftwerk Emsland am 24.07.2009
2010/04	"Nichtschließen eines Sicherheits- und Entlastungsventils aufgrund einer Schwergängigkeit im zugehörigen Vorsteuerventil" im Kernkraftwerk Krümmel am 28.08.2004 (endgültige Meldung 08.09.2009)
2010/05	"Rückstände von Formierpapier in verschiedenen Systemen" in den Kernkraftwerken Flamanville-1 (Frankreich) und Philippsburg-2
2010/06	Kontaktprobleme an Simulierschaltern auf Baugruppen des Typs XKU im Kernkraftwerk Isar-1 am 14.03.2008
2010/07	"Malware auf speicherprogrammierbaren Steuerungen unter SIMATIC WinCC und SIMATIC PCS7"
2010/07A	„Malware auf speicherprogrammierbaren Steuerungen unter SIMATIC WinCC und SIMATIC PCS 7“
2011/01	"Rissbefunde an den Reaktorwasserreinigungspumpen" in den Kernkrafterken Brunsbüttel und Isar-1
2011/02	"Kleinstleckage an einer Entleerungsleitung am Dampferzeuger" im Kernkraftwerk Neckarwestheim-2 am 28.09.2010

2011/03	„Fehler am Generatorleistungsschalter des UNS-Notstromdiesels EY60“ im Kernkraftwerk Brunsbüttel
2011/04	Meldepflichtiges Ereignis im Kernkraftwerk Grohnde am 15.04.2010 „Fehlfunktion der Startwiederholung am Notspeise-Notstromdiesel GY50“
2011/05	„Anzeigen im Vorschuhende des Stutzens der Hauptkühlmittelleitung zur Volumenausgleichsleitung“ im Kernkraftwerk Grafenrheinfeld
2011/06	Meldepflichtiges Ereignis im Forschungsreaktor München II am 01.04.2009 „Funktionsstörung an der Krananlage SMA10 durch eine defekte Elektronikbaugruppe des Funksteuerempfängers“
2011/07	„Befunde an einer Zwischenkühlpumpe des Sicherheitskomponentensystems“ im Kernkraftwerk Neckarwestheim, Block 2, am 08.10.2008
2011/08	„Blockierte Federhänger/-stützen an Rohrleitungen“ im Kernkraftwerk Emsland, vorgefunden am 20.06.2010
2012/01	Einsatz nicht spezifikationsgerechter Feinsicherungen auf leittechnischen Baugruppen in deutschen Kernkraftwerken
2012/02	Auswirkungen des Tohoku-Erdbebens an den japanischen Kernkraftwerksstandorten Fukushima Dai-ichi (I) und Dai-ni (II) am 11.03.2011 und des Niigataken Chuetsu-Oki-Erdbebens am japanischen Kernkraftwerksstandort Kashiwazaki-Kariwa am 16.07.2007
2012/03	Regenwassereintrag in das Kernkraftwerk Brunsbüttel am 04.09.2011
2012/03A	"Regenwassereintrag im Bereich des Feststofflagers" im Kernkraftwerk Brunsbüttel, gemeldet am 28.06.2017
2012/04	Bruch von Niederhaltefedern von Brennelementen mit Stahlführungsrohren“ in den Kernkraftwerken Brokdorf und Grafenrheinfeld
2012/04A	"Bruch von Niederhaltefedern an Westinghouse-Brennelementen" in den Kernkraftwerken Emsland (KKE) und Grohnde (KWG), gemeldet am 24.05.2017 (KKE) bzw. 28.09.2017 (KWG)
2012/05	„Befunde an Messwerken der Füllstandsonden des Typs AVL200“ im Kernkraftwerk Grohnde
2012/06	"Schäden an Schiebern der saugseitigen Absperrung der nuklearen Nebenkühlwasserpumpen" im Kernkraftwerk Unterweser
2013/01	Ausfall einer gesicherten Drehstromschiene bei fehlerhafter Anregung der Drehzahlüberwachung aller rotierenden Umformer im Kernkraftwerk Grohnde
2013/02	Befunde an bautechnischen Brandschutzmaßnahmen im Kernkraftwerk Philippsburg 2 am 10.04.2012 sowie nachfolgend
2013/02A	Befunde an bautechnischen Brandschutzmaßnahmen im Kernkraftwerk Philippsburg 2 am 10.04.2012 sowie nachfolgend
2013/03	Sicherstellung der erforderlichen Mindestlast für Notstromaggregate während des Nichtleistungsbetriebes der Anlagen sowie in der Nachbetriebs- und Stilllegungsphase
2013/04	Nichtstarten eines Notspeisenotstromdiesels wegen defekter Startluftverteilerscheibe im Kernkraftwerk Grafenrheinfeld am 10.04.2009
2013/05	„Unzureichend detektierte Ausfälle einzelner Phasen der Fremd- bzw. Reservenetzanbindung in mehreren ausländischen Anlagen“
2013/06	Spanbildung in Leittechnikschränken in Kernkraftwerken der Bundesrepublik Deutschland
2013/07	Anregung der Notstromsignale in den Redundanzen 1 und 4 nach Ausfall des 380-kV-Hauptnetzanschlusses im Kernkraftwerk Biblis, Block A
2013/08	Freischaltfehler im Kernkraftwerk Philippsburg, Block 2
2013/09	„Abweichungen vom spezifizierten Zustand in jeweils einem Kanal der Flutbehälterfüllstandsmessungen“ im Kernkraftwerk Philippsburg-2
2014/01	Schäden an Brennelement-Zentrierstiften im Kernkraftwerk Philippsburg 2
2014/02	Nicht auslegungsgemäße Brunnenwasserversorgung des Notstandsnebenkühlwassersystems“ im belgischen Kernkraftwerk Tihange
2014/03	Schäden an Komponenten infolge Primärkreisdekontamination im Kernkraftwerk Biblis, Block A
2014/04	Fehlöffnen von Magnetvorsteuerventilen in den FSA-Stationen in den Kernkraftwerken Neckarwestheim-2 (GKN-2), Emsland (KKE) und Isar 2 (KKI-2)
2014/05	„Befunde an Druckfedern von Drosselkörpern“ im Kernkraftwerk Grohnde vorgefunden während der Revision 2014
2014/06	„Vorrangbaugruppen mit schadhaften Kondensatoren“ im Kernkraftwerk Philippsburg-2
2014/07	Ausfall eines Messkreises der Drehzahlerfassung an der Hauptkühlmittelpumpe in GKN-2
2014/08	Ausfall des Drehzahlgebers der Laufbrücke der BE-Lademaschine im Kernkraftwerk Emsland
2014/09	„Rohrleckage im Zwischenkühler eines Nachkühlstranges“ im Kernkraftwerk Brunsbüttel
2014/10	Versagen der automatischen Zuschaltung einer Umluftanlage im Notspeisegebäude im Kernkraftwerk Brokdorf am 01.04.2013
2014/11	„Ausfall eines UNS-Diesels aufgrund Generatorschaden“ im Kernkraftwerk Brunsbüttel
2014/12	„Ansprechen der Notkühlkriterien während des Abfahrens im Kernkraftwerk Neckarwestheim 2“
2014/13	Schaden an Batterien der unterbrechungsfreien Spannungsversorgung im Zwischenlager Brunsbüttel
2014/14	Absturz eines 20'-Containers durch Versagen einer Sicherungseinrichtung beim Krantransport im Zwischenlager Nord
2015/01	„Nicht ordnungsgemäße Zuordnung von Stellantrieben an Frischdampfentwässerungsarmaturen“ im Kernkraftwerk Emsland, erkannt am 17.06.2012
2015/02	„Schwelbrand von Reststoffen in einem Abfallgebinde innerhalb der Trocknungsanlage“ im Kernkraftwerk Isar, Block 1, am 30.09.2011
2015/03	„Nichtschließen des Druckhalter-Abblaseabsperrventils“ im Kernkraftwerk Brokdorf (KBR) am 07.10.2013
2015/04	„Schaden an einer Entwässerungsleitung im Frischdampfsystem bei einer WKP während des Abfahrens der Anlage“ im Kernkraftwerk Brokdorf, vorgefunden am 30.05.2015
2015/05	„Auffälligkeit am Schienentisch in der Materialschleuse“ im Kernkraftwerk Unterweser, erkannt am 01.11.2015
2015/06	„Fehlansprechen von Reaktorschutzabschlussgliedern durch wiederkehrende Prüfungen“ in den Kernkraftwerken Neckarwestheim 2 (GKN-2) und Brokdorf (KBR)
2015/07	Mangelhafte Störfallfestigkeit verschiedener elektrotechnischer Komponenten in mehreren deutschen Kernkraftwerken
2015/08	„Erhöhte Ausfallrate von Widerstandsthermometern aufgrund eines Herstellungsmangels“ erkannt im Kernkraftwerk Krümmel am 03.12.2013

2015/09	„Anzeigen an Stiftschrauben des Rückschlagventilblocks der Frischdampfabschlussarmaturen im Zuge von zerstörungsfreien Prüfungen“ im Kernkraftwerk Emsland vorgefunden am 24.05.2013
2016/01	„Neue Erkenntnisse zur Wirksamkeit von Systemen zur gefilterten Druckentlastung des Sicherheitsbehälters“
2016/02	„Ausfall eines 6,3-/0,4-kV-Notstromtransformators über Buchholz-Auslösung“ im Kernkraftwerk Isar 1
2016/03	„Befund am Wärmeschutzrohr in einem Stutzen des nuklearen Nachwärmeabfuhrsystems“ im Kernkraftwerk KKI-2, vorgefunden am 12.07.2015
2016/04	„Wiederholtes Versagen von Unterspannungsauslösern des Typs 3AX 1103 -2F (Siemens)“ im Kernkraftwerk Neckarwestheim, Block 2
2016/05	„Lösen eines Brennstabbündels vom Brennelementkopf“
2016/06	„Nichtschließen des Ventils ‘Einspeisung heiß‘ im Not- und Nachkühlsystem bei betrieblicher Anforderung“ im Kernkraftwerk Brokdorf am 09.11.2011
2016/07	„Wanddickenschwächung einer Entlüftungsleitung im Frischdampfsystem“ im Kernkraftwerk Neckarwestheim, Block 2, festgestellt am 06.04.2011
2016/08	„Schadsoftwarefunde im Kernkraftwerk Gundremmingen, Block B“, gemeldet am 24.04.2016
2016/09	Fehlerhafte Auslösung von Brandschutzklappen im unabhängigen Sabotage- und Störfallschutzsystem (USUS) infolge einer Störung in der Brandmeldeanlage MF51" im Kernkraftwerk Philippsburg-1, gemeldet am 07.11.2013
2016/10	„Ablagerungen an Kühlwassertemperaturreglern der Notstromdiesel“ im Kernkraftwerk Grohnde, gemeldet am 16.10.2015
2016/11	„Gelöste Laufradmutter in einer Nachkühlpumpe“ im Kernkraftwerk Grohnde, gemeldet am 18.04.2016
2016/12	„Unregelmäßigkeiten bei wiederkehrenden Prüfungen in den Blöcken 1 und 2“ im Kernkraftwerk Philippsburg
2016/13	„Einbau ungeeigneter Ersatzkomponenten in mehreren deutschen Kernkraftwerken“
2016/14	"Baugruppenfehler in einer Brandmeldezentrale" im Kernkraftwerk Brunsbüttel, gemeldet am 23.08.2016
2017/01	„Beschädigte Verbindungsbolzen an Halterungen von Lüftungskanälen im Notspeisegebäude“ im Kernkraftwerk Philippsburg-2, gemeldet am 20.12.2016
2017/02	"Unzureichendes Schaltvermögen von Gleichstromschaltern nach Erweiterung der Batteriekapazität" in einer ausländischen Anlage
2017/03	"Ölfreisetzung an einer Hauptkühlmittelpumpe mit lokaler Flammbildung“ im Kernkraftwerk Emsland am 27. Oktober 2013
2017/04	"Erhöhte Oxidschichtdicke an Brennstabhüllrohren von Brennelementen" im Kernkraftwerk Brokdorf (KBR), gemeldet am 17.02.2017
2017/05	"Unerkannter Einsatz nicht qualifizierter Relais mit programmierbaren Bauelementen in mehreren ausländischen Anlagen"
2017/06	"Defekte Membranen in Armaturen in aktivitätsführenden Systemen in den Kraftwerken Philippsburg-2, Brokdorf und Biblis, Block A und B"
2017/07	"Ausfall von Drehzahlwächtern und Drehmessumformern der Firma Jaquet" in mehreren deutschen Kernkraftwerken
2018/01	"Ausfall der logarithmischen Mittelwertmesser der Impulskanäle im Kernkraftwerk Brokdorf", gemeldet am 08.06.2017
2018/02	"Nicht erfolgtes automatisches Wiederzuschalten mehrerer Gleichrichter in einer Redundanz bei einer Eigenbedarfsumschaltung" im Kernkraftwerk Isar (KKI-2), gemeldet am 27.09.2017
2018/03	"Verstellte Zentriermuttern an Stellungsanzeigen von Erstabsperrarmaturen" im Kernkraftwerk Emsland, erkannt am 23.05.2017
2018/04	"Schutzabschaltung eines Notstromdiesels bei 110 % Lastlauf" im Kernkraftwerk Philippsburg-2, gemeldet am 04.08.2016
2018/05	"Messumformer KLA85-CP871 entspricht nicht der Spezifikation" im Kernkraftwerk Neckarwestheim-2, gemeldet am 30.09.2015
2018/06	Anzeigen bei Wirbelstromprüfungen an Dampferzeugerheizrohren im Kernkraftwerk Neckarwestheim-2 (GKN-2) gemeldet am 25.09.2017 und 14.09.2018
2018/07	"Schäden durch Wasserhämmer im Bereich der FSA-Station im Kernkraftwerk Angra 2 (Brasilien)", aufgetreten am 02.03.2018
2019/01	"Anstieg des Unterdrucks im Reaktorsicherheitsbehälter (RSB) auf 80 mbar im Rahmen einer betrieblichen Schalthandlung" im Kernkraftwerk Brokdorf am 21.03.2017
2019/02	Ausfälle von leittechnischen Baugruppen durch defekte Tantal-Elektrolytkondensatoren im Kernkraftwerk Neckarwestheim-2 (GKN-2) und im Forschungsreaktor BER-II
2019/03	"Fehlende Seiten im Betriebshandbuch" im Kernkraftwerk Brokdorf, gemeldet am 18.06.2018
2019/04	"Befunde bei der Befestigung von Haltesegmenten der Klappenblattdichtung von Absperrklappen" im Kernkraftwerk Philippsburg, Block 2, gemeldet am 22.04.2016
2019/05	"Notfallmaßnahmen zur Beherrschung eines gemeinsam verursachten Ausfalls (GVA) der Frischdampf-Abblase-Regelventile bei unverfügbarer Hauptwärmesenke"

Annex 2 List of RSK statements and recommendations in the period 1973-2019

Datum	Titel
27.03.2019	Anforderungen bei einer passiven Kühlung der Brennelemente im Lagerbecken
06.02.2019	Aspekte der Qualitätssicherung bei wiederkehrenden Prüfungen und Instandhaltungsmaßnahmen sowie beim Einsatz von Fremdpersonal
23.05.2018	Bewertung der Sicherheitsnachweise für die Reaktordruckbehälter der belgischen Kernkraftwerke Doel-3 / Tihange-2 (korrigierte Fassung)
06.12.2017	Zusammenfassende Stellungnahme der RSK zu zivilisatorisch bedingten Einwirkungen, Flugzeugabsturz - Teilbericht: Festlegung der Lastannahmen und Bewertung der Konvoi-Anlagen
06.09.2017	Bewertung der Umsetzung von RSK-Empfehlungen im Nachgang zu Fukushima
22.03.2017	Randbedingungen der Nachweisführung zur Störfallbeherrschung
22.03.2017	Bewertung der Umsetzung der Empfehlungen der RSK aus der Sicherheitsüberprüfung deutscher Forschungsreaktoren
03.11.2016	Blitze mit Parametern oberhalb der genormten Blitzstromparameter
03.11.2016	Monitoring von Know-how- und Motivationsverlust und geeignete Maßnahmen zur Stärkung von Motivation und Know-how-Erhalt in der deutschen Kernenergiebranche
18.05.2016	Schäden an BE-Zentrierstiften und Kernbauteilen
13.04.2016	Vorläufige Kurzbewertung der Sicherheitsnachweise für die Reaktordruckbehälter der belgischen Kernkraftwerke Doel-3 / Tihange-2
10.02.2016	Aspekte der Ermittlung des standortspezifischen Bemessungshochwassers
09.12.2015	Anforderungen an die Brennelement-Lagerbeckenkühlung
24.06.2015	Nachweis einer Restduktilität/Restfestigkeit mittels einer ECR-Grenzkurve
15.04.2015	Anforderungen an die statistische Nachweisführung bei Kühlmittelverluststörfall-Analysen
15.04.2015	Wasserstofffreisetzung aus dem Sicherheitsbehälter
18.03.2015	Verformungen von Brennelementen in deutschen Druckwasserreaktoren (DWR)
14.01.2015	Scheibenübergreifende Unverfügbarkeiten aufgrund elektrischer Kopplungen zwischen redundanten Scheiben des Notstromsystems deutscher Kernkraftwerke
11.12.2014	Ausbildung und Auswirkungen eines Deionatpfropfens beim Dampferzeugerheizrohrleck
06.11.2014	Leitfaden für die Durchführung von ganzheitlichen Ereignisanalysen
04.09.2014	Rahmenempfehlungen für die Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken
26.06.2014	Ein- oder zweiphasiger Ausfall des Haupt-, Reserve- oder Notstromnetzanschlusses
22.05.2014	Der RSK-Leitfaden für die Durchführung von ganzheitlichen Ereignisanalysen im Vergleich zum VGB-Leitfaden Ganzheitliche Ereignisanalyse
06.11.2013	Einschätzung der Abdeckung extremer Wetterbedingungen durch die bestehende Auslegung
29.08.2013	RSK-Verständnis der Sicherheitsphilosophie (Veröffentlicht im Bundesanzeiger AT 05.12.2013 B4)
20.06.2013	Konkretisierung von Anforderungen im Zusammenhang mit der 10 h-Autarkie bei zivilisatorischen Einwirkungen von außen (Notstandsfälle)
11.04.2013	Mindestwert von 0,1g (ca. 1,0 m/s²) für die maximale horizontale Bodenbeschleunigung bei Erdbeben
11.04.2013	DWR-Neutronenflussschwankungen
21.02.2013	Druck- und Dichtheitsprüfungen an Bauteilen der Druckführenden Umschließung (DFU) und der Äußeren Systeme, insbesondere nach Reparaturen
17.01.2013	Ultraschallanzeigen am Reaktordruckbehälter des belgischen Kernkraftwerks Doel, Block 3 (Doel-3)

13.12.2012	Kriterien für die Alarmierung der Katastrophenschutzbehörde durch die Betreiber kerntechnischer Einrichtungen
13.12.2012	Netzstabilität: Rückwirkungen von Stabilitätsproblemen im deutschen Stromnetz auf elektrische und leittechnische Einrichtungen von Kernkraftwerken und Sicherstellung der notwendigen elektr. Energieversorgung dieser Anlagen aus dem Netz
01.12.2012	Prüf- und Überwachungsprogramm für Öl-Papier-isolierte Transformatoren und Trockentransformatoren in deutschen Kernkraftwerken
26.09.2012 - 27.09.2012	Redesign von leittechnischen Baugruppen und Komponenten in Kernkraftwerken
26.09.2012 - 27.09.2012	Betriebsunterlagen: Auflagen und Bedingungen des sicheren Betriebs
26.09.2012 - 27.09.2012	Empfehlungen der RSK zur Robustheit der deutschen Kernkraftwerke
12.07.2012	Drohende Gefährdung der kerntechnischen Sicherheit durch Know-How- und Motivationsverlust
03.05.2012	Anwendung des Betriebshandbuches im Fahrbetrieb deutscher Kernkraftwerke (Veröffentlicht im Bundesanzeiger AT 30.07.2012 B1)
03.05.2012	Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Forschungsreaktoren unter Berücksichtigung der Ereignisse in Fukushima-I (Japan)
03.05.2012	Zu unterstellende Leckagen an Dampferzeuger(DE)-Heizrohren: Mehrfachrohrbruch/Lecköffnung wanddickengeschwächter DE-Heizrohre (Veröffentlicht im Bundesanzeiger AT 26.07.2012 B3)
05.04.2012	Empfehlungen zur maximalen zulässigen kritischen Borkonzentration zur Sicherstellung der Unterkritikalität nach „Reflux-Condenser-Betrieb“ beim kleinen Leckstörfall
05.04.2012	Spannungsnachweis und Prüfbarkeit der Schweißnaht an der Verbindung zwischen Zylinder und unterer Bodenkalotte in Reaktordruckbehältern (RDB) von Kernkraftwerken mit Siedewasserreaktoren (SWR) der Baureihe 69 Kernkraftwerke Krümmel (KKK), Brunsbüttel (KKB), Philippsburg Block 1 (KKP-1) und Isar, Block 1 (KKI-1)
05.04.2012	Ausfall der Primären Wärmesenke (Veröffentlicht im Bundesanzeiger AT 03.08.2012 B5)
29.02.2012 - 01.03.2012	Stellungnahme zum Kerntechnischen Regelwerk Entwurfsfassung Rev. E
17.11.2011	RSK-Verständnis zur Robustheit im Zusammenhang mit dem EU-Stresstest
15./16.09.2011	Rechnerbasierte Sicherheitsleittechnik für den Einsatz in der höchsten Sicherheitskategorie in deutschen Kernkraftwerken
07.07.2011	Regelungen zu Anlagenzuständen nach Eintritt eines Störfalls
11.05.2011 - 14.05.2011	Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Kernkraftwerke unter Berücksichtigung der Ereignisse in Fukushima-I (Japan) [aus der 437. RSK-Sitzung vom 11. bis 14. Mai 2011]
30.03.2011	"Vorspann" und "Anforderungskatalog für anlagenbezogene Überprüfungen deutscher Kernkraftwerke unter Berücksichtigung der Ereignisse in Fukushima-I (Japan)"
16.12.2010	Bewertung des Drei-Säulen-Konzeptes zur Prüfung sicherheitstechnisch wichtiger Armaturen
14.10.2010	Rahmenempfehlungen für die Planung von Notfallschutzmaßnahmen durch Betreiber von Kernkraftwerken
12.04.2007	Antrag nach § 6 AtG auf Errichtung des Brennelementzwischenlagers Obrigheim
13.09.2006	RSK-Stellungnahme zum Synthesebericht des BfS "Konzeptionelle und sicherheitstechnische Fragen der Endlagerung radioaktiver Abfälle - Wirtsgesteine im Vergleich"
10.08.2006	Darstellung des Kenntnisstandes zum Einfluss der Gammastrahlung auf die betriebliche Zähigkeitsabnahme ferritischer Reaktordruckbehälterwerkstoffe
10.08.2006	Auswirkung fortgeschrittener Kernbeladungen auf das Reaktivitätsverhalten des Reaktorkerns und seiner Reaktivitätsstellglieder
09.03.2006	Stellungnahme zu Mängeln an Mittelspannungskabeln mit sicherheitstechnischer Bedeutung in deutschen Kernkraftwerken aus Anlass des Meldepflichtigen Ereignisses ME E 13.1/04 - Kernkraftwerk Brunsbüttel (KKB) "Störung in der Eigenbedarfsversorgung mit RESA" vom 23.08.2004
09.03.2006	Strahlenschäden im Steinsalz

15./16.12.2005	Beherrschung eines Kühlmittelverluststörfalls bei DWR unter Berücksichtigung von Totvolumina im Reaktorsicherheitsbehälter- Sicherhheitsmanagement-Aspekte
15./16.12.2005	Vorschlag für Anforderungen an die Stilllegung im kerntechnischen Regelwerk
15./16.12.2005	Derzeitige Vorgehensweise bei zerstörungsfreien Prüfungen, die im Rahmen der wiederkehrenden Prüfungen an ferritisch-austgenitischen Mischschweißnähten erfolgen (Stellungnahme zum Schwerpunktthema 1 des Untersuchungsvorhabens SR 2360 des BMU)
06.10.2005	Einstufung von "VO-Ereignissen in die Sicherheitsebenen des gestaffelten Sicherheitskonzepts und Konzept zur Neubestimmung von Vorosrgemaßnahmen (VM)
08.09.2005	Spezifikation der Fa. AREVA/COGEMA zu hochdruckkompaktierten radioaktiven Abfällen (CSD-C) aus der Wiederaufarbeitung von deutscchenLWR-Brennelementen
08.09.2005	Gestaffeltes Sicherheitskonzept
08.09.2005	Festlegung von Versagenspostulaten für Komponenten
07.07.2005	Stellungnahme der RSK zum Abschalten der Hauptkühlmittelpumpen bei ATWS-Ereignissen in Kernkraftwerken mit Druckwasserreaktor
07.07.2005	Wirbelstrombefunde an Steuerelementen- Meldepflichtige Ereignisse 02/2003 im Kernkraftwerk Brokdorf, 02/2004 im Kernkraftwerk Neckarwestsheim, Block 2 und 08/2004 im Kernkraftwerk Emsland
02.06.2005	Genehmigungsverfahren zur Stilllegung und zum Rückbau des Kernkraftwerks Stade
28.04.2005	Untersuchungsvorhaben SR 2392 des BMU "Einsatz von Thermoelementen zur Erfassung der Temperatur von Rohrleitungswandungen in Kenrkraftwerken im Rahmen der Ermüdungsanalyse" und Berücksichtigung des Mediumeinflusses bei Ernüdungsanalysen nach dem KTA-Regelwerk
24.02.2005	Sicherheitstechnische Aspekte konzeptioneller Fragestellungen zur Endlagerung von bestrahlten Brennstäben mittels Kokillen in Bohrlöchern anhand eines Vergleichs mit dem Konzept "Streckenlagerung von dickwandigen Behältern"
27.01.2005	Untersuchungsvorhaben SR 2318 des BMU "Bewertung von Aussagefähigkeit von Ultraschall- und Wirbelstromprüfung austenitischer Plattierungen von Reaktordruckbehältern"
27.01.2005	Sicherheitsaspekte des Einsatzes hochabgebrannter Brennelemente unter Reaktivitätsstörfall-Bedingungen
27.01.2005	Gase im Endlager
16.12.2004	Stellungnahme der RSK zur Erweiterung der Urananreicherungsanlage Gronau
16.12.2004	Meldepflichtiges Ereignis "Absturz eines Brennelementes beim Beladen des Transportbehälters" im Kernkraftwerk Biblis, Block B, am 06.08.2001
22.07.2004	Anforderungen an den Nachweis der Notkühlwirksamkeit bei Kühlmittelverluststörfällen mit Freisetzung von Isoliermaterial und anderen Stoffen
04.03.2004	Vorkommnis der INES-Kateggorie 3 im amerikanischen Kernkraftwerk Davis Besse vom 6. März 2002, "Borsäurekorrosion am Reaktordruckbehälterdeckel" und Schlussfolgerungen für deutsche Anlagen
27.02.2004	KTA-Regel 2201.1: "Auslegung von Kernkraftwerken gegen seismische Einwirkungen; Teil 1: Grundsätze"; Fassung 6/90 - Empfehlungen für die Überarbeitung der Regel
11.12.2003	Handhabungsfehler im Gemeinschaftskernkraftwerk Neckar beim Umsetzen eines Brennelementens (Block I, ME 03/2002) und einer Primärneutronenquelle (Block II, ME 02/2002)
13.11.2003	Meldepflichtiges Ereignis im Kernkraftwerk Unterweser (KKU), Nichtverfügbarkeit einer von vier Frischdampf-Sicherheitsaramturen-Stationen (FSA) bei Anforderung am 06.06.1998
16.10.2003	Schäden an Steuerelementen in SWR-Anlagen
18.09.2003	Erhöhung der thermischen Reaktorleistung des Kernkraftwerkes Grafenrheinfeld (KKG)
10.07.2003	Kernkraftwerk Unterweser (KKU), Meldepflichtiges Ereignis, Befundanzeigen an Anschlussstutzen des Speisewassersystems an die Dampferzeuger

10.04.2003	Umsetzung der Empfehlung der RSK "Sicherheitstechnische Leitlinien für die trockene Zwischenlagerung bestrahlter Brennelemente in Behältern" im BfS-Genehmigungsentwurf für das Standort-Zwischenlager in Gemmrigheim der Gemeinschaftskernkraftwerk Neckar Gmb
10.04.2003	Gemeinschaftskernkraftwerk Neckarwestheim, Block 2 (GKN-2), Abriss des Thermosleeves am Stutzen der kalten Einspeisung JNA 31
13.03.2003	RSK-Stellungnahme zum Fraktionsumlauf der KTA-Basisregeln
05.12.2002	Gemeinsame Stellungnahme der RSK und der SSK betreffend BMU-Fragen zur Fortschreibung der Endlager-Sicherheitskriterien
07.11.2002	Umsetzung der Grundsätze der Empfehlung der RSK "Sicherheitstechnische Leitlinien für die trockene Zwischenlagerung bestrahlter Brennelemente in Behältern" im BfS-Genehmigungsentwurf für das Standortzwischenlager Grafenrheinfeld
07.11.2002	Verfüll- und Verschließkonzept für das Endlager für radioaktive Abfälle Morsleben (ERAM)
10.10.2002	Qualitätsmanagement hinsichtlich sicherheitstechnischer Anforderungen an die Herstellung und Beschaffung von Brennelementen
05.09.2002	Umsetzung der Grundsätze der Empfehlung der RSK "Sicherheitstechnische Leitlinien für die trockene Zwischenlagerung bestrahlter Brennelemente in Behältern" im BfS-Genehmigungsentwurf für das Standortzwischenlager Lingen
11.07.2002	Sicherheit deutscher Zwischenlager für bestrahlte Brennelemente in Lagerbehältern bei gezieltem Absturz von Großflugzeugen
13.06.2002	Memorandum der RSK zur Gewährleistung einer angemessenen Sicherheitskultur
16.05.2002	RSK-Stellungnahme zum Arbeitsprogramm KTA-2000, Beschlussfassung zur Verabschiedung der Basisregeln als Entwürfe
16.05.2002	Folgerungen aus einer anlagenspezifischen probabilistischen Sicherheitsanalyse für den Nicht-Leistungsbetrieb bei Leichtwasserreaktoren
07.03.2002	Stellungnahme zu aktuellen Fragen zur Endlagerung in Salzgestein
10.01.2002	Grundsätze für das Vorgehen zur Beherrschung von Alterungsprozessen in Kernkraftwerken
13.12.2001	Weiterleitungsnachricht der GRS 2000/13, Fehlerbedingte sekundärseitige Lastabsenkung und nicht erfolgter Stabeinwurf im Gemeinschaftskernkraftwerk Neckar, Block 1 (GKN-1) am 10.05.2000
11.10.2001	KTA-Regeländerungsvorhaben, KTA 1404,Regelentwurf (Gründruck "Dokumentation beim Bau und Betrieb von Kernkraftwerken"
11.10.2001	Sicherheit deutscher Atomkraftwerke gegen gezielten Absturz von Großflugzeugen mit vollem Tankinhalt
13.09.2001	Schwerpunkte zukünftiger FuE-Arbeiten bei der Endlagerung radioaktiver Abfälle
13.09.2001	Fragen zur Zusammenarbei mit der russischen Föderation bei der Entsorgung von Plutonium aus der Abrüstung
07.06.2001	Kernkraftwerk Stade (KKS) Rissbefund am Austrittsstutzen der nachkühl-Saugearmatur TH 022 S1 (Erstabsperrung) und anschließende Rohrleitungen
03.05.2001	Bewertung von Membranspannungen
03.05.2001	ATWS-Ereignisse
03.05.2001	Kernkraftwerk Stade (KKS) Rissbefund am Austrittstutzen einer Nachkühl-Saugearmatur (Erstabsperrung) und anschließende Rohrleitungen
01.02.2001	Kernkraftwerk Biblis, Block A /(KWB-A), Vorkommnis VA 07/00, Risse in der Schweißnaht YA01 65W
01.02.2001	Nutzung von Freihaltepositionen im Brennelementlagerbecken des Kernkraftwerkes Stade
09.11.2000	Kernkraftwerk Biblis, Block A (KWB-A), Risse in einer Schweissnaht einer Anschlussleitung (TH-System) an die Hauptkühlmittelleitung
07.09.2000	Konsequenzen aus dem Kritikalitätsunfall in der JCO-Uranverarbeitungsanlage in Tokaimura, Japan, für deutsche Anlagen
08.06.2000	Beibehaltung der Anwendung der DIN/ISO 7503.1 zum Nachweis der Einhaltung der verkehrsrechtlichen Grenzwerte für Kontaminationen an Oberflächen von Versandstücken

06.04.2000	Meldepflichtiges Ereignis N07/98 und Vorkommnis vom 23.09.1999 im Kernkraftwerk Krümmel (KKK) "Lose Sicherheitsmutterna an Steuerstabantriebsgehäuserohren"
02.12.1999	Ergänzende Stellungnahme der RSK zur Jahr-2000-Problematik
11.11.1999	Jahr-2000-Problematik Überprüfunge von Software und softwarebasierten Systemen und Komponenten mit sicherheitstechnischer Bedeutung in deutschen Kernkraftwerken im Hinblick auf die Datumsumstellung
16.09.1998	Beratung von Erdbebenfragen in der RSK Auslegung deutscher Kernkraftwerke gegen Erdbeben Skalierung von Bemessungsspektren bei amax < 1 m/s2
16.09.1998	Beratung von Erdbebenfragen in der RSK Seismologische Standortbewertung für das Kernkraftwerk Krümmel (KKK) und für den Forschungsreaktor der GKSS in Geesthacht
16.09.1998	Wirksamkeit der Notkühlsysteme bei Freisetzung von Isoliermaterial bei KMV-Störfällen
16.09.1998	Einsatz von Brennelementen mit hohen Abbränden
16.09.1998	Erhöhung der Lagerkapazität von Brennelement-Lagerbecken in deutschen Kernkraftwerken
06.07.1998	Forschungsreaktor München II (FRM-II) Fortsetzung der Beratung zum Genehmigungsverfahren
06.07.1998	Verarbeitung von verschnittenem Uran in der Brennelementfertigung der Fa. Advanced Nuclear Fuels GmbH (ANF) in Lingen/Ems
20.05.1998	Kernkraftwerk Brokdorf (KBR) Beantragte Leistungserhöhung
20.05.1998	Beratung von Erdbebenfragen in der RSK Kerntechnische Anlagen in Baden-Württemberg Externes Brennelementlager am Kernkraftwerk Obrigheim (KWO)
10.09.1997	UF6-Freilager der Fa. Advanced Nuclear Fuel (ANF) GmbH in Lingen Kritikalitätsdetektierungs- und Warnsystem
02.07.1997	Der Beitrag der Kernenergie zu einer Nachhaltigen Entwicklung
21.05.1997	Zusammenarbeit RSK/GPR Sicherheitsanforderungen an zukünftige Kernkraftwerke mit Druckwasserreaktor
23.04.1997	RSK-Denkschrift zur Sicherheitskultur in der Kerntechnik
19.02.1997	Auslegung von deutschen Kernkraftwerken gegen Erdbeben
11.09.1996	Anwendung von Dämmschichtbildnern (DSB) als Brandschutzmaßnahme für Kabelanlagen in Kernkraftwerken
26.06.1996	Kernkraftwerk Brunsbüttel (KKB) Auslegung gegen Erdbeben
22.05.1996	Neues Notstandssystem für die Blöcke A und B des Kernkraftwerks Biblis
20.03.1996	Kernkraftwerk Unterweser (KKU) Probabilistische Sicherheitsanalyse der Frischdampf-Sicherheitsarmaturen-Station (FSA-Station)
20.03.1996	Kernkraftwerk Philippsburg 1 (KKP-1) Geplanter Austausch der Speisewasserleitungen in der Revision des Jahres 1997
27.09.1995	Erkundungsbergwerk Gorleben Schadensfälle an Schrauben im Fundament- und Stützringbereich der Schächte
27.09.1995	Reaktordruckbehälter von Kernkraftwerken älterer Bauart hier: Kernkraftwerk Obrigheim (KWO)
27.09.1995	Blitzschutz- und Erdungskonzepte in deutschen Kernkraftwerken Ergebnisse anlagenspezifischer Untersuchungen
10.08.1995	Fortsetzung der Beratungen aufgrund der RSK-Empfehlung vom 23.11.1998 Abschlußbericht über die Ergebnisse der Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland hier: Auswirkungen eines Brandes in einer Redundanz auf die Signalübe
21.06.1995	Einsatz von digitaler Sicherheitsleittechnik in deutschen Kernkraftwerken
17.05.1995	Wiederaufarbeitungsanlage Karlsruhe (WAK) HAWC-Verglasung auf dem Gelände des Forschungszentrums Karlsruhe (FZK)
12.04.1995	Weiterleitungsnachricht zu meldepflichtigen Ereignissen in Kernkraftwerken der Bundesrepublik Deutschland (WL 10/94)

12.04.1995	Besondere Vorkommnisse in ausländischen Kernkraftwerken Störfall im schwedischen Siedewasserreaktor Barsebäck 2 am 28.07.1992 Übertragbarkeit des Ereignisses auf deutsche Kernkraftwerke und von deutschen Betreibern getroffene Maßnahmen
12.04.1995	Kernkraftwerk Greifswald (KGR), Blöcke 1 bis 6 Stillegung der Anlage und Abbau von Anlagenteilen
15.02.1995	Auswirkungen von Funksignalen auf sicherheitstechnisch wichtige elektrische Einrichtungen
15.02.1995	Kernkraftwerk Rheinsberg (KKR) Stillegung und Teilabbau
14.12.1994	Kernkraftwerke mit Druckwasserreaktor Mögliches fehlerhaftes Umschalten der Sicherheitshochdruck-Einspeisung auf die kalte Einspeisung
12.07.1994	Spezifikation für die Abfallprodukte aus der Verglasung hochradioaktiver Abfallösungen aus der Wiederaufarbeitungsanlage Karlsruhe (WAK)
12.07.1994	Seismische Instrumentierung in deutschen Kernkraftwerken
12.07.1994	Seismotektonische Verhältnisse an den norddeutschen Kernkraftwerksstandorten hier: Kernkraftwerk Brunsbüttel (KKB)
12.07.1994	Technische Fragestellungen im Zusammenhang mit internen Zwangsumwälzpumpen der Kernkraftwerke mit Siedewasserreaktor (KKB, KKP-1, KKI-1)
12.07.1994	Siemens Brennelementewerk Hanau Betriebsteil MOX-Verarbeitung Brandschutzkonzept mit dem Einsatz von Halon als Löschmittel
12.07.1994	Produktionsstätte HOBEG der NUKEM GmbH Genehmigungsverfahren nach § 7 Abs. 3 AtG zur Stillegung und Änderung der Zweckbestimmung
18.05.1994	Kernkraftwerk Biblis, Block A und B (KWB-A und KWB-B) Maßnahmen zur Durchmischung des Wasserstoffs in der Atmosphäre des Sicherheitsbehälters
16.02.1994	Kernkraftwerk Brunsbüttel (KKB) Reparaturkonzept für die austenitischen Rohrleitungen
08.12.1993	Nuklearer Brennstoffkreislauf Beförderung von unbestrahlten SNR-Brennelementen auf dem Luftwege
10.11.1993	Rißbildungen in Rohrleitungen größer gleich NW 80 aus stabilisierten austenitischen St„hlen X 10 CrNiTi 18 9 (W-Nr. 1.4541) und X 10 CrNiNb 18 9 (W-Nr. 1.4550) in deutschen Leichtwasserreaktoren/Schadensbild und Reparaturkonzepte
10.11.1993	Betriebssicherheit nuklearer Einrichtungen Ergebnisse der Operational Safety Review Team (OSART)-Mission im Kernkraftwerk Grafenrheinfeld (KKG)
22.09.1993	Spezifikation der UKAEA für die zementierten Abfälle ("Intermediate Level Residue Specification Cemented Liquid Wastes") aus der Wiederaufarbeitung von bestrahlten Brennelementen aus deutschen Forschungsreaktoren
22.09.1993	Kernkraftwerke mit Druck- und Siedewasserreaktor Wiederkehrende Prüfungen Zerstörungsfreie Prüfung austenitischer Schweißverbindungen von Rohrleitungen
16.06.1993	Urananreicherungsanlage Gronau (UAG) Inbetriebnahme des zweiten Bauabschnitts und Betrieb der Gesamtanlage (5. TG)
16.06.1993	Spezifikationen der COGEMA für Abfälle aus der Wiederaufarbeitung von bestrahlten LWR-Brennelementen Stellungnahmen zu a) bis c) a) Hülsen und Strukturteile c) Konditionierte technologische Abfälle
16.06.1993	Besondere Vorkommnisse in deutschen Kernkraftwerken Vorkommnis vom 06.04.1992 im Kernkraftwerk Brunsbüttel (KKB) Schnelles Öffnen der Frischdampf-Isolationsventile
17.02.1993	Kernkraftwerk Brunsbüttel (KKB) Rißbildungen in Rohrleitungen aus titanstabilisiertem Stahl
09.12.1992	Vorbeugende Instandhaltung an Sicherheitssystemen während des Anlagenbetriebs
14.10.1992	Kernkraftwerke mit Siedewasserreaktoren Nuklear-thermohydraulische Stabilität
14.10.1992	Kernkraftwerke mit Druckwasserreaktoren und Siedewasserreaktoren Probenahme aus dem Sicherheitsbehälter bei auslegungsüberschreitenden Ereignissen
16.09.1992	Kernkraftwerk Obrigheim (KWO) Bedingungen für die 3. Wiederkehrende Druckprüfung am Primärkreis
16.09.1992	Kernkraftwerk Würgassen (KWW) Rißbefunde an der austenitischen Umfangsschweißnaht ZR 112 WN 3 (sog. Sulzer-Naht) der Treibwasserschleife I während der Jahresrevision 1992

16.09.1992	Spezifikationen der BNFL zu mittel- bzw. niedrigradioaktiven Abfallprodukten aus der Wiederaufbereitung von bestrahlten LWR-Brennelementen aus deutschen Kernkraftwerken 4 Stellungnahme zu a) bis d) a) Hülsen- und Strukturteile ("Hulls and Ends"), b) Feedk
16.09.1992	Siemens Brennelementewerk Hanau Betriebsteil MOX-Verarbeitung Weiterer Übergangsbetrieb nach Paragraph 9 AtG bis zur Inbetriebnahme der Anlage nach Paragraph 7 AtG
17.06.1992	Kernkraftwerk Würgassen (KWW) Gefilterte Druckentlastung des Sicherheitsbehälters (Containment-Venting) hier: Behandlung von Fragen aus dem Gutachten der Elektrowatt Ingenieurunternehmung GmbH zur Errichtung und Nutzung des Druckentlastungssystems
17.06.1992	Kernkraftwerk Biblis, Block A und Block B (KWB-A, KWB-B) Nachrüstung einer Absperrarmatur innerhalb des Reaktorsicherheitsbehälters in der Hauptkühlmitteleinspeisung der Volumenregelung
17.06.1992	Spezifikationen der BNFL für verglaste Abfälle aus der Wiederaufarbeitung von bestrahlten LWR-Brennelementen aus deutschen Kernkraftwerken
17.06.1992	RSK-Empfehlung vom 23.11.1988: "Abschlußbericht über die Ergebnisse der Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK" Anwendung der Leck-Postulate, Umrüstung der Haupt- und Hilfssprühleitungen, Gemeinschaftsker
17.06.1992	Erfüllungsgrad der RSK-"Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk" beim Endlager Morsleben (ERAM)
15.04.1992	Anlageninterner Notfallschutz Umrüstung der Druckhalter-Armaturenstation hier: Gemeinschaftskernkraftwerk Neckar, Block 1 (GKN-1)
15.04.1992	Gemeinschaftskernkraftwerk Neckar, Block 1 (GKN-1) Anwendbarkeit der Leckpostulate gemäß der geltenden Fassung der RSK-Leitlinien auf die Hauptkühlmittelleitung und die Volumenausgleichsleitung von GKN-1
15.04.1992	Anlageninterner Notfallschutz Einfluß von Maßnahmen zur primärseitigen Druckentlastung und Bespeisung auf das Verhalten von Konvoi-Anlagen bei Störfällen
18.03.1992	Kernkraftwerk Würgassen (KWW) Umfangsschweißnähte der Treibwasserschleifen 1 und 2 Grundwerkstoff X 10 CrNiNb 18 9 (Betriebsdauer: 100.000 h)
18.03.1992	Allgemeine Anforderungen an Krisenstabübungen
27.01.1992	Sicherheit des Kernkraftwerks Greifswald, Block 5, als Beispiel für den sowjetischen Reaktortyp WWER-440/W-213 und Verbesserungsmöglichkeiten
27.01.1992	Fortsetzung der Beratungen im Zusammenhang mit der RSK-Empfehlung vom 23.11.1988 "Abschlußbericht über die Ergebnisse der Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK"· Brandschutz ·· Kriterien- und Maßnahmenka
27.01.1992	·· Bereitstellung von Sicherheitsstatusberichten zum Brandschutzkonzept durch die Anlagenbetreiber Erforderlicher Umfang an Informationen im "Sicherheitsstatusbericht zum Brandschutz"
27.01.1992	· Erdbeben Bedeutung des Baseler Erdbebens von 1356 für die Beurteilung der seismotektonischen Verhältnisse an Standorten deutscher Kernkraftwerke, insbesondere im Oberrheingraben
10.09.1991	Siemens-Brennelementwerk Hanau, Betriebsteil MOX-Verarbeitung Besonderes Vorkommnis am 17.06.1991
10.09.1991	Siemens-Brennelementwerk Hanau, Betriebsteil MOX-Verarbeitung Gasentwicklung in Spaltstoffgebinden mit Folienumhüllungen
24.06.1991	Bewertung des Sicherheitsstatus im Rahmen der periodischen Sicherheitsüberprüfung Kernkraftwerk Obrigheim (KWO)
24.06.1991	Anlageninterner Notfallschutz Kernkraftwerk Obrigheim (KWO) Nachrüstung der Druckhalter-Armaturenstation im Hinblick auf die primärseitige Druckentlastung und Bespeisung
24.06.1991	Integritätsnachweis für langzeitig betriebene Reaktordruckbehälter von Leichtwasserreaktoren Kernkraftwerk Stade (KKS)
24.06.1991	Kernkraftwerke mit Druckwasserreaktor Mögliches fehlerhaftes Umschalten der Sicherheitshochdruckeinspeisung auf Kalteinspeisung
24.06.1991	Auswirkungen von Funksignalen auf sicherheitstechnisch wichtige elektrische Einrichtungen
24.06.1991	Kernkraftwerke mit Hochtemperaturreaktor Sicherheitseigenschaften, Betriebserfahrungen u. zukunftsorientierte Forschungsvorhaben

24.06.1991	Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK am 23.11.1988 Maßnahmen zur diversitären Druckbegrenzung Kernkraftwerk Krümmel (KKK)
24.06.1991	Kernkraftwerk Obrigheim (KWO) Verschiebung der nächsten wiederkehrenden Druckprüfung des Primärkreises, der Frischdampf- und Speisewasserleitung, der Dampferzeuger-Sekundärbehältermantel sowie der relevanten zerstörungsfreien Prüfungen
22.05.1991	Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland Kernkraftwerk Obrigheim (KWO) Austausch von Rohrleitung im Reaktorgebäude
22.05.1991	Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland Kernkraftwerk Unterweser (KKU) Umrüstung der Druckhalter-Abblasearmaturenstation
22.05.1991	Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland Anlagen•bergreifende Beratungsthemen · Schutz von Sicherheitseinrichtungen gegen Überflutung · Auslegung von Leitungen im Ringraum · Sicherheitsbehälterdurchdringungen
24.04.1991	Kernkraftwerk Mülheim-Kärlich Vergleichbarkeit des für die Komponenten Reaktordruckbehälter, Druckhalter, Dampferzeuger und Hauptkühlmittelpumpen eingesetzten Stahls 22 NiMoCr 37 mit dem Stahl 20MnMoNi55
24.04.1991	Kernkraftwerk Vandellos-1 (Spanien) Brand im Maschinenhaus am 19.10.1989
24.04.1991	Kernkraftwerk Loviisa (Finnland) Bruch einer Speisewasserleitung am 28.05.1990
20.02.1991	Stellungnahme/Entwurf Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK vom 23.11.1988 Erfüllung der Leck-Postulate für Lecks an Hauptkühlmittel-, Frischdampf- und Speisewasserleitungen innerhalb des Reaktorsicherhe
20.02.1991	Stellungnahme/Entwurf Siemens Brennelementwerk Hanau Sicherheitstechnische Bedeutung des besonderen Vorkommnisses vom 12.12.1990 im Betriebsteil Uranverarbeitung für das Genehmigungsverfahren nach § 7 AtG für den Betriebsteil MOX-Verarbeitung
20.02.1991	Stellungnahme Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK vom 23.11.1988 Notfallmaßnahmen zur diversitären Druckbegrenzung hier: Kernkraftwerk Brunsbüttel (KKB)
23.01.1991	Stellungnahme Anlageninterner Notfallschutz Kernkraftwerk Stade: Nachrüstung der Druckhalter-Abblasestation im Hinblick auf die primärseitige Druckentlastung und Bespeisung
19.12.1990	Stellungnahme/Entwurf zum Vorschlag der MPA-Stuttgart an den BMU zum Thema: "Flankierende Erhebungen und Untersuchungen zur Beurteilung von rißauslösenden Faktoren beim Betrieb vobn druckführenden Komponenten in Leichtwasser-Reaktoranlagen".
19.12.1990	Stellungnahme Kernkraftwerke mit Druckwasserreaktor Ermüdungsanalysen, Erschöpfungsgrade und Betriebsfestigkeitsnachweise von Anschlußrohrleitungen und Stutzen im Primärkreis
19.12.1990	Stellungnahme/Entwurf Anlageninterner Notfallschutz Aufgaben des Schichtleiters und des Krisenstabs bei schweren Störfällen und bei anlageninternen Notfällen
14.11.1990	Stellungnahme Anlageninterner Notfallschutz Kernkraftwerk Philippsburg 2: Änderungsmaßnahmen im Zusammenhang mit der sekundär- und primärseitigen Druckentlastung und Bespeisung
14.11.1990	Stellungnahme Anlageninterner Notfallschutz, Organisationsstruktur des Krisenstabs
14.11.1990	Stellungnahme Regelungen zum Schichtwechsel in deutschen Kernkraftwerken
19.09.1990	Stellungnahme Kernkraftwerk Biblis, Blöcke A und B Anlageninterner Notfallschutz Gefilterte Druckentlastung der Sicherheitsbehälter
19.09.1990	Stellungnahme Kernkraftwerk Biblis, Block B Umrüstung der Druckhalter-Abblasestation
19.09.1990	Stellungnahme Kernkraftwerk Biblis, Block A Besonderes Vorkommnis der Kategorie E am 06.06.1990: Ausfall der 24 V Gleichspannungsversorgung
19.09.1990	Stellungnahme der RSK zur Stellungnahme der atomrechtlichen Genehmigungsbehörde des Landes Nordrhein-Westfalen (MWMT) vom 30.11.1988 zum Bethe-Tait-Komplex
19.09.1990	Stellungnahme Siemens Brennelementwerk Hanau Betriebsteil MOX-Verarbeitung Abschließende Errichtungs- und Betriebsgenehmigung

20.06.1990	Stellungnahme Prüfumfang und Austauschkonzepte bei Schäden an Brennelement-zentrierstiften in Druckwasserreaktoren
20.06.1990	Stellungnahme Maßnahmen zur diversitären Druckbegrenzung für den Reaktor-druckbehälter bei den Kernkraftwerken KKI-1 und KRB-II
20.06.1990	Stellungnahme Meldekriterien für besondere Vorkommnisse in Kernkraftwerken, GRS vos/kur 81 122 vom 8.5.1990
20.06.1990	Stellungnahme der RSK zu speziellen Fragen des BMU zur weiteren Behandlung des Bethe-Tait-Störfalls im Genehmigungsverfahren des Kernkraftwerkes Kalkar (SNR-300)
30.05.1990	Stellungnahme Richtlinie für Programme zur Erhaltung der Fachkunde des verant-wortlichen Schichtpersonals in Kernkraftwerken
21.03.1990	Stellungnahme Langzeitsicherheit bei der Endlagerung radioaktiver Abfälle in der Schachtanlage Konrad
21.03.1990	Stellungnahme Regeländerungsentwurf KTA 3201.4 "Komponenten des Primärkrei-ses von Leichtwasserreaktoren, Teil 4: Wiederkehrende Prüfungen und Betriebs-überwachung", Fassung 6/89
21.02.1990	Stellungnahme Kernkraftwerk Biblis, Block A Ertüchtigung der primärseitigen Druck-absicherungseinrichtungen
21.02.1990	Stellungnahme zur Regelentwurfsvorlage KTA 3211.3, Fassung 4/89 Druck- und ak-tivitätsführende Komponenten von Systemen außerhalb des Primärkreises, Teil 3, Herstellung
24.01.1990	Stellungnahme Kernkraftwerk Biblis, Blöcke A und B Konzept des Notstandssystems
24.01.1990	Stellungnahme Wasserstoffbildung und -verbrennung nach schweren Unfällen in Kernkraftwerken mit Druckwasserreaktor
20.12.1989	Stellungnahme Auslegung von Kernkraftwerken gegen Flugzeugabsturz - Ergeb-nisse von Versuchen bei SANDIA/USA
20.12.1989	Stellungnahme Brennelementfertigungsanlage Lingen (BFL) Trockenkonversion
20.12.1989	Stellungnahme Siemens Brennelementwerk Hanau Betriebsteil MOX-Verarbeitung Auslegung der Prozeßanlagen hinsichtlich der Minimierung der Strahlenexposition
20.12.1989	Stellungnahme Filtersysteme für die Druckentlastung der Sicherheitsbehälter von Kernkraftwerken mit Druckwasserreaktor nach einem schweren Unfall
15.11.1989	Stellungnahme Teilaustausch von Frischdampfleitungen im Kernkraftwerk Philipps-burg 1 (KKP-1)
15.11.1989	Stellungnahme Kernkraftwerk Brunsbüttel · Bruchpostulate für Speisewasserleitun-gen außerhalb des Sicherheitsbehälters
15.11.1989	Stellungnahme Vorübergehender Ausfall der Nachwärmeabfuhr und der Notstrom-versorgung im Kernkraftwerk Trillo bei der Erdschlußsuche am 9.9.1988 und ver-gleichbare Vorkommnisse in deutschen Kernkraftwerken
15.11.1989	Stellungnahme Kompakte Natriumgekühlte Kernreaktoranlage II (KNK II/2) Schwer-gängigkeit an den Abschalteinrichtungen
15.11.1989	Stellungnahme Aussagekraft der Betriebsberichte zur Information der RSK
15.11.1989	Stellungnahme Qualifizierung neuer Informationssysteme für Kernkraftwerke
15.11.1989	Stellungnahme Wiederaufarbeitungsanlage Wackersdorf
15.11.1989	Stellungnahme Staatliche Verwahrung von Kernbrennstoffen gemäß § 5 AtG
18.10.1989	Stellungnahme Kernkraftwerk Philippsburg 1 Diversitäre Sicherheits- und Entlas-tungsventile
18.10.1989	Stellungnahme Kernkraftwerk Brunsbüttel · Bruchpostulate für Frischdampfleitungen außerhalb des Sicherheitsbehälters
18.10.1989	Stellungnahme Kernkraftwerk Grundremmingen, Blöcke B und C · Inertisierung der Kondesationskammern · Druckentlastung der Sicherheitsbehälter über Filtersysteme · Wartenzuluftfilterung
17.07.1989	Stellungnahme Kernkraftwerk Biblis A/B Einbindung der Füllstandsmessung im Re-aktordruckbehälter in den Reaktorschutz
17.07.1989	Stellungnahme Kernkraftwerk Neckarwestheim 1 · Einbindung der Füllstandsmes-sungen im Reaktordruckbeh„lter in den Reaktorschutz · Verbesseruung des Dampf-erzeuger-Heizrohrbruch-Konzeptes durch Nachrüsten automatischer Maßnahmen

17.07.1989	Kernkraftwerk Neckarwestheim 2 (GKN-2) - Druckentlastung des Sicherheitsbehälters über Filtersysteme bei Unfällen
17.07.1989	Stellungnahme Kernkraftwerk Obrigheim - Druckentlastung des Sicherheitsbehälters über Filtersysteme bei Unfällen
17.07.1989	Stellungnahme Kernkraftwerk Obrigheim - Filterung der Wartenzuluft
24.05.1989	Stellungnahme Bedeutung, Inhalt und Umfang von Notfallhandbüchern in Kernkraftwerken
24.05.1989	Stellungnahme Kernkraftwerk Philippsburg 1 Verschiebung der nächsten wiederkehrenden Druckprüfung des Reaktordruckbehälters
24.05.1989	Stellungnahme Vermischen von radioaktiven Abfällen
24.05.1989	Stellungnahme Tritiumerzeugung in Hochtemperaturreaktoren (HTR) Schreiben des e.V. Greenpeace am 8.12.1988 an den Niedersächsischen Umweltminister
26.04.1989	Stellungnahme Kernkraftwerk Hamm-Uentrop (THTR-300) Befunde in den Heißgaskanälen und ihre sicherheitstechnische Bewertung im Hinblick auf den weiteren Betrieb
15.03.1989	Stellungnahme Kernkraftwerk Biblis, Blöcke A und B Filterung der Wartenzuluft
15.03.1989	Stellungnahme Kernkraftwerk Biblis A, Füllstandsmessung im Reaktordruckbehälter
15.03.1989	Stellungnahme Überprüfung des Alterungsverhaltens elektronischer Baugruppen des Sicherheitssystems
15.02.1989	Stellungnahme Kernkraftwerk Kalkar (SNR-300) Einsatz von Kleinhebezeugen im Hinblick auf die KTA-Regel 3902
18.01.1989	Stellungnahme Übertragbarkeit des in der Deutschen Risikostudie Phase B für Biblis B untersuchten Ereignisablaufs Dampferzeuger-Heizrohrbruch mit zusätzlich unterstellten Systemfehlern auf andere Druckwasserreaktoren
18.01.1989	Stellungnahme Anlagen- und sicherheitstechnische Aspekte des Gutachtens "Die Gefahr von Strahlenschäden durch Plutonium" von Prof. Dr. Horst Kuni, Marburg
18.01.1989	Stellungnahme Kennzeichnung von Endlagern in der Nachbetriebsphase
23.11.1988	Stellungnahme Anlageninterner Notfallschutz · Druckentlastung des Sicherheitsbehälters von Druckwasserreaktoren unter Einsatz von Jodsorptionsfiltern mit Molekularsieben
23.11.1988	Stellungnahme Anlageninterner Notfallschutz Gemeinschaftskernkraftwerk Neckar 1 (GKN-1) · Druckentlastung des Sicherheitsbehälters über Filtersysteme
23.11.1988	Stellungnahme Anlageninterner Notfallschutz Gemeinschaftskernkraftwerk Neckar (GKN-1) · Filterung der Wartenzuluft
19.10.1988	Stellungnahme Kernkraftwerk Kalkar (SNR-300) · Elektrische Energieversorgung für die sicherheitstechnisch wichtige Leittechnik
21.09.1988	·Stellungnahme Kernkraftwerk Grafenrheinfeld (KKG) · Druckentlastung des Sicherheitsbehälters bei Unfällen Wartenzuluftfilterung
21.09.1988	Stellungnahme Kernkaftwerk Philippsburg 2 (KKP-2) · Druckentlastung des Sicherheitsbehälters bei Unfällen
21.09.1988	Stellungnahme Kompakte Natriumgekühlte Kernreaktoranlage II (KNK-II) Erfahrungen aus dem bisherigen Brennelement-Betriebsverhalten und Konsequenzen für die beantragte Standzeitverlängerung
21.09.1988	Stellungnahme Kernkraftwerk Hamm-Uentrop (THTR-300) Auswirkungen der Errichtung und des Betriebs einer Ammoniak-Versorgungsanlage für die Kohlekraftwerksblöcke auf den THTR-300
22.06.1988	Gemeinsame Stellungnahme der Reaktor-Sicherheitskommission (RSK) und der Strahlenschutzkommission (SSK), Stand: 30.6.1988 Zeitrahmen für die Beurteilung der Langzeitsicherheit eines Endlagers für radioaktive Abfälle
22.06.1988	Stellungnahme Brennelementfabrik der Firma ALKEM Sicherheitszustand der gegenwärtig betriebenen Pu-Nitrat-Lagerung und Pu-Konversion
22.06.1988	Stellungnahme Kernkraftwerk Mülheim-Kärlich (KMK) Anlageninterner Notfallschutz - Druckentlastung des Sicherheitsbehälters bei Unfällen - Filterung der Wartenzuluft
22.06.1988	Stellungnahme Kernkraftwerk Würgassen (KWW) Anlageninterner Notfallschutz · Inertisierung des Sicherheitsbehälters · Druckentlastung des Sicherheitsbehälters · Wartenzuluftfilterung

18.05.1988	Stellungnahme Wasserstoffbildung und -verbrennung nach hypothetischen Kernschmelzunfällen in Leichtwasserreaktoren
18.05.1988	Stellungnahme Erkundungsbergwerk Gorleben, Schachtausbaukonzept gegen Wasser- und Laugenzutritt aus Deckgebirgshorizonten mit ungleichförmiger Verteilung der horizontalen Gebirgsspannungen
18.05.1988	Stellungnahme Kernkraftwerk Biblis, Block A Verwendung von Brennelementen mit stärker oxidierten Brennstabhüllrohren für die nächsten Betreibszyklen
18.05.1988	Gemeinschaftskernkraftwerk Neckar, Block 1 (GKN 1) Antrag auf Verschiebung der wiederkehrenden Ultraschallprüfung am Reaktordruckbehälter
18.05.1988	Stellungnahme Sicherheitstechnische Bewertung der Tiefversenkung tritiumhaltiger Wässer aus der Wiederaufarbeitung ausgedienter Brennelemente im Vergleich zur Zementierung und Verbringung in ein begehbares Endlagerbergwerk als alternatives Verfahren der E
20.04.1988	Stellungahme Spezifikation für Filtersysteme in den Druckentlastungsstrecken des Sicherheitsbehälters von DWR und SWR
16.03.1988	Stellungnahme Überprüfung der Sicherheit der Kernkraftwerke in der Bundesrepublik Deutschland Kernkraftwerk Philippsburg 1 (KKP-1) · Druckentlastung des Sicherheitsbehälters über ein Filtersystem · Inertisierung des Sicherheitsbehälters · Zuluftfilte
16.03.1988	Stellungnahme Überprüfung der Sicherheit der Kernkraftwerke in der Bundesrepublik Deutschland Kernkraftwerk Philippsburg 2 (KKP-2) Druckentlastung des Sicherheitsbehälters über ein Filtersystem, Inertisierung des Sicherheitsbehälters, Zuluftfilterung der
16.03.1988	Stellungnahme Kernkraftwerk Philippsburg 1 (KKP-1) Stellungnahme zu Presseberichten über mögliche Belastungen bei Rohrleitungsbrüchen
16.03.1988	Stellungnahme Kernkraftwerk Philippsburg 2 (KKP-2) Stellungnahme zu Presseberichten über Zweifel an der ordnungsgemäßen Durchf•hrung sicherheitstechnischer Überprüfungen
16.03.1988	Stellungnahme Spezifikation der COGEMA für verglaste Abfälle aus der Wiederaufarbeitung von abgebrannten LWR-Brennelementen aus deutschen Kernkraftwerken
20.01.1988	Stellungnahme Stellungnahme zur Gasbildung in Abfallfässern
20.01.1988	Stellungnahme Kompakte Natriumgekühlte Kernreaktoranlage 2 (KNK 2) - Verlängerung der Standzeit des Kerns
16.12.1987	Stellungnahme Überprüfung der Sicherheit der Kernkraftwerke in der Bundesrepublik Deutschland Kernkraftwerk Isar 1 (KKI-1) - Druckentlastung des Sicherheitsbehälters über ein Filtersystem - Inertisierung des Sicherheitsbehälters - Zuluftfilterung der Wart
16.12.1987	Stellungnahme Neubau des Forschungsreaktors der Technischen Universität München
16.12.1987	Stellungnahme Urananreicherungsanlage Gronau (UAG), Errichtung des 2. Bauabschnitts (600 t UTA/a)
25.11.1987	Stellungnahme Kernkraftwerk Kalkar (SNR-300), Beratungen zur Inbetriebnahme
25.11.1987	Stellungnahme Überprüfung der Sicherheit der Kernkraftwerke mit Leichtwasserreaktoren in der Bundesrepublik Deutschland Anlageninterne Notfallschutzmaßnahmen für die Konvoi-Anlagen gemäß der RSK-Empfehlung vom 17.12.1986
25.11.1987	Stellungnahme Kernkraftwerke Stade (KKS), Unterweser (KKU), Grohnde (KWG) · Reaktorsicherheitsbehälterabschluß · Ausstattung der Warte · Druckentlastung des Sicherheitsbehälters
25.11.1987	Stellungnahme Kernkraftwerk Stade (KKS) Strahlenbeeinflussung des Reaktordruckbehälters
25.11.1987	Stellungnahme Kernkraftwerk Obrigheim (KWO) Strahlenbeeinflussung des Reaktordruckbehälters und zur Qualität der Hauptkühlmittelleitungen
21.10.1987	Stellungnahme Änderungen an Brennelementen von Druckwasserreaktoren
21.10.1987	Stellungnahme Regeländerungsvorlage KTA 3201.3, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil 3: Herstellung, Fassung September 1987
21.10.1987	Stellungnahme Regeländerungsentwurfsvorschlag KTA 3201.4, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Wiederkehrende Prüfungen und Betriebsüberwachung, Fassung Juni 1987
30.07.1987	Stellungnahme Brennelementfabrik NUKEM (alt)

30.07.1987	Stellungnahme Kernkraftwerk Brunsbüttel (KKB) Inertisierung des Sicherheitsbehälters
30.07.1987	Entwurf/Stellungsnahme der RSK/SSK Zeitrahmen für die Beurteilung der Langzeitsicherheit eines Endlagers für radioaktive Abfälle
24.06.1987	Stellungnahme Überprüfung der Sicherheit der Kernkraftwerke mit Leichtwasserreaktor in der Bundesrepublik Deutschland
24.06.1987	Stellungnahme Zur Veröffentlichung von E. Grimmel, Hamburg und E. Koch, Büchelberg, Naturwissenschaftliche Rundschau, 35. Jahrg., 10/1982 Sicherheit von Kernkraftwerken bei Erdbeben aus geowissenschaftlicher Sicht
24.06.1987	Stellungnahme Kernkraftwerk Biblis, Block A, Verwendung stärker oxidierter Brennstabhüllrohre in den nächsten Betriebszyklen
15.04.1987	Stellungnahme Oxidschichtbildung bei Brennstabhüllrohren von Druckwasserreaktoren
15.04.1987	Stellungnahme Blitzschutzmaßnahmen bei den in Betrieb befindlichen Kernkraftwerken
15.04.1987	Stellungnahme Bewertung des Unfalls im Kernkraftwerk Tschernobyl im Hinblick auf das Kernkraftwerk Kalkar (SNR-300)
15.04.1987	Stellungnahme Grundsätze der Sicherheitsanforderungen an die trockene Lagerung bestrahlter Brennelemente in Brennelementbehältern
18.02.1987	Stellungnahme Vorhaben auf dem Gebiet der Sicherheit von Kernkraftwerken - Kinetische Grenztragfähigkeit von Stahlbetonumschließungen im Kernkraftwerksbau
18.02.1987	Stellungnahme Schallemissionsanalyse der druckführenden Komponenten von Kernkraftwerken
18.02.1987	Stellungnahme Kernkraftwerk Hamm-Uentrop (THTR-300), Stellungnahme zu den Ergebnissen und Erfahrungen bei der Inbetriebnahme und zum weiteren Betrieb
18.02.1987	Stellungnahme Schäden an Schrauben der Kernumfassung aus dem Werkstoff Inconel X 750 bei einigen Kernkraftwerken mit Druckwasserreaktor
18.02.1987	Stellungnahme Schäden an Zentrierstiften der oberen Gitterplatte aus dem Werkstoff Inconel X 750 am Kernkraftwerk Neckarwestheim 1
17.12.1986	Stellungnahme "Untersuchungen zu den Ereignisabläufen mit Kernschmelzen und Aktivitätsfreisetzungen in den DWR-Anlagen KKS und KBR sowie in den SWR-Anlagen KKB und KKK", erstellt vom TÜV Norddeutschland; Entwurf vom Dezember 1985
17.12.1986	Stellungnahme Kernkraftwerk Grafenrheinfeld (KKG), Bruch einer Hauptkühlmittelpumpenwelle, weiteres Vorgehen und Konsequenzen für vergleichbare Wellen
12.11.1986	Zweiter Bericht der Reaktor-Sicherheitskommission (RSK) zur Bewertung des Unfalls im Kernkraftwerk Tschernobyl im Hinblick auf Kernkraftwerke mit Leichtwasserreaktor in der Bundesrepublik Deutschland
12.11.1986	Stellungnahme Kernkraftwerk Biblis (Blöcke A und B) Maßnahmen zur Überwachung und Durchmischung des Wasserstoffs in der Reaktorsicherheitsbehälteratmosphäre nach einem Kühlmittelverlust-Störfall
15.10.1986	Stellungnahme Bewertung der Ergebnisse der Expertentagung über den Unfall im Kernkraftwerk Tschernobyl und weiter RSK-Beratungen
17.09.1986	Stellungnahme Brennelementfabrik der Firma ALKEM Verlegung des Chemiebetriebs in den Bunkerbereich
17.09.1986	Stellungnahme Kernkraftwerk Brokdorf (KBR) Überprüfung der Sicherheit im Zusammenhang mit dem Unfall im Kernkraftwerk Tschernobyl
17.09.1986	Stellungnahme Gemeinschaftskernkraftwerk Neckar 1 (GKN 1) Verbesserung des Leckageüberwachungssystems der druckführenden Umschließung
17.09.1986	Stellungnahme Filterung der Abluft bei der Druckentlastung von Sicherheitsbehältern nach einem Kernschmelzunfall
25.06.1986	Stellungnahme Fragen der Genehmigungsbehörde im Zusammenhang mit den Genehmigungsverfahren für den Forschungsreaktor BER II
06.06.1986	Zwischenbericht der Reaktor-Sicherheitskommission (RSK) zur vorläufigen Bewertung des Unfalls im Kernkraftwerk Tschernobyl im Hinblick auf Kernkraftwerke in der Bundesrepublik Deutschland
14.05.1986	Stellungnahme Einsatz von Uran/Plutonium-Mischoxid MOX-Brennelementen im Kernkraftwerk Brokdorf
14.05.1986	Stellungnahme der RSK zum Unfall im Kernkraftwerk Tschernobyl in der UdSSR

19.03.1986	Stellungnahme Brennelementfabrik der Firma ALKEM - Sicherheitstechnische Fragen zum gegenwärtigen Betrieb
19.03.1986	Stellungnahme BMI-Vorhaben SR 350, Übertragbarkeit von Ergebnissen aus Bestrahlungsüberwachungsprogrammen auf den Werkstoffzustand der Reaktordruckbehälter von Leichtwasserreaktoren
19.02.1986	Stellungnahme Konvoi-Anlagen Verbindungsmöglichkeiten im Notspeisesystem und Stellungsüberwachung sicherheitstechnisch relevanter Handarmaturen
22.01.1986	Stellungnahme Bewertung neuerer wissenschaftlicher Erkenntnisse zum Kernzerlegungsstörfall im Hinblick auf den SNR-300
18.12.1985	Stellungnahme Optimierung von Brandschutzmaßnahmen in Kernkraftwerken (SR 144)
18.12.1985	Stellungnahme Verfahren zur Zulassung von Dübeln in kerntechnischen Anlagen
13.11.1985	Stellungnahme Leckratenprüfung an Sicherheitsbehältern von Kernkraftwerken
13.11.1985	Stellungnahme Kernkraftwerk Philippsburg 2, Konzept für wiederkehrende zerstörungsfreie Prüfungen
13.11.1985	Kernkraftwerk Grohnde, Konzept für wiederkehrende zerstörungsfreie Prüfungen
13.11.1985	Stellungnahme KTA-Regelvorlage 1502.1, Überwachung der Radioaktivität in der Raumluft von Kernkraftwerken, Teil 1: Fassung 6/85 Kernkraftwerke mit Leichtwasserreaktoren
13.11.1985	Stellungnahme Regeländerungsvorlage KTA 1201 Anforderungen an das Betriebshandbuch, Fassung 12/85
13.11.1985	Stellungnahme Regelvorlage KTA 2201.4, Auslegung von Kernkraftwerken gegen seeismische Einwirkungen, Teil 4: Maschinen- und elektrotechnische Anlagenteile, Fassung 10/85
13.11.1985	Stellungnahme Regelentwurfsvorlage KTA 2206, Auslegung von Kernkraftwerken gegen Blitzeinwirkung, Fassung 9/85
13.11.1985	Stellungnahme Regelvorlage KTA 2101.1, Brandschutz in Kernkraftwerken, Teil 1: Grundsätze des Brandschutzes, Fassung August 1985 mit Änderungen aus Sitzungen des KTA-UA ANLAGEN- UND BAUTECHNIK am 1.10. und 28.10.1985
18.09.1985	Stellungnahme Kernkraftwerk Kalkar (SNR-300) Entfallen des Schockabsorbers im Hilfsanlagengebäude und der Halon-Löschanlage im Gaslager-Kühlsystem
18.09.1985	Stellungnahme Regelentwurfsvorlage KTA 2206, Auslegung von Kernkraftwerken gegen Blitzeinwirkungen, Fassung 3/85
19.06.1985	Stellungnahme Wiederaufarbeitungsanlage Wackersdorf - Einfluß der Revision der Konzeptunterlagen und der Ergebnisse des Gutachtens zur Bau- und Anlagentechnik auf die Empfehlung der RSK aus der 198. Sitzung (17.10.1984)
19.06.1985	Stellungnahme Untersuchungen zur Beanspruchung von Gebäuden für den Lastfall Flugzeugabsturz - Vorhaben des Instituts für Massivbau und Baustofftechnologie, Universität Karlsruhe
19.06.1985	Stellungnahme Bewertung des im Rahmen von SR 209 entwickelten Konzeptes einer Kernrückhaltevorrichtung in Leichtwasserreaktoren
19.06.1985	Stellungnahme Beurteilung der Erdbebenlastannahmen für den Standort Hanau
19.06.1985	Stellungnahme Gemeinschaftskernkraftwerk Neckar 1 (GKN 1) Schließsicherheit der Frischdampf-Schnellschlußschieber
19.06.1985	Stellungnahme Erfahrungen mit eigenmediumgesteuerten Sicherheits- und Entlastungsventilen
24.04.1985	Stellungnahme Forschungsreaktor BER II in Berlin - Umbau und Betrieb des Reaktors mit einer Leistung von 10 MW
24.04.1985	Stellungnahme Regelvorlagen KTA 1408.1 bis 1408.3 Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- und aktivitätsführende Komponenten in Kernkraftwerken Teil 1: Einungsprüfung, Teil 2: Herstellung, Teil 3: Verarbeitung, KTA Dok.-Nr. 140
24.04.1985	Stellungnahme Regeländerungsvorlage KTA 3401.2, Reaktorsicherheitsbehälter aus Stahl, Teil 2: Auslegung, Konstruktion und Berechnung, Fassung 2/85
24.04.1985	Stellungnahme Regeländerungsvorlage KTA 3401.3, Reaktorsicherheitsbehälter aus Stahl, Teil 3: Herstellung, Fassung 2/85
24.04.1985	Stellungnahme Regelentwurfsvorlage KTA 2103 Explosionsschutz in Kernkraftwerken, Fassung 8/84

24.04.1985	Stellungnahme Regelentwurfsvorlage KTA 3705.1, Schaltanlagen, Transformatoren und Verteilungsnetze zur elektrischen Energieversorgung des Sicherheitssystemes in Kernkraftwerken,Teil 1: Übergeonrdnete Anforderung an Auslegung und Berechnung, Fassung 11/84
24.04.1985	Stellungnahme Regelentwurfsvorlage KTA 3904 Warte, Notsteuerstelle und örtliche Leitstände in Kernkraftwerken, Fassung 3/1985
23.01.1985	Stellungnahme Erfahrungen mit eigenmediumgesteuerten Sicherheits- und Entlastungsventilen in Kernkraftwerken im Hinblick auf Ausfälle mit gemeinsamer Ursache
19.12.1984	Stellungnahme der RSK zur Systemstudie Andere Entsorgungstechniken Entsorgungstechniken, PKA, WA 700, Langzeitsicherheit
19.12.1984	Stellungnahme Meldekriterien für Besondere Vorkommnisse in Kernkraftwerken (Stand: Oktober 1984)
14.11.1984	Stellungnahme Antrag der Firma ALKEM auf Erteilung eines Vorbescheides nach § 7a AtG zu konzeptionellen Einzelfragen der nach § 7 AtG beantragten Gesamtanlage am Standort Hanau-Wolfgang
14.11.1984	Stellungnahme Geänderte Abschnitte 5.4 und 6 der Regeländerungsentwurfsvorlage KTA-1201 (Anforderungen an das Betriebshandbuch), Fassung 1201/84/6
14.11.1984	Stellungnahme KTA-1301/1, Berücksichtigung des Strahlenschutzes der Arbeitskräfte bei Auslegung und Betrieb von Kernkraftwerken, Teil 1: Auslegung (Fassung Juni 1984)
17.10.1984	Stellungnahme Wiederaufarbeitungsanlage Bayern/Niedersachsen, Anlagensicherung
17.10.1984	Stellungnahme Optimierung der Bemessung von Gebäuden für den Lastfall Flugzeugabsturz
17.10.1984	Stellungnahme Regeländerungsvorlage KTA 3501, Reaktorschutzsystem und Überwachungseinrichtungen des Sicherheitssystems, Fassung 9/84
17.10.1984	Stellungnahme Regelentwurfsvorlage KTA 3504, Elektrische Antriebe des Sicherheitssystems von Kernkraftwerken, Fassung 9/84 und vorangegangene Fassung KTA Dok. Nr. 3504/84/3
17.10.1984	Stellungnahme Regelvorlage KTA 3505, Typprüfung von Meßwertgebern und Meßumformern des Rektorschutzsystems, Fassung 9/84
17.10.1984	Stellungnahme Regelvorlage KTA 3506, Systemprüfung der leittechnischen Einrichtungen des Sicherheitssystems, Fassung 5/84
17.10.1984	Stellungnahme Regelvorlage KTA 3301, Nachwärmeabfuhrsystem von Leichtwasserreaktoren Fassung Juni 1984 mit dokumentierten Änderungen vom Oktober 1984
17.10.1984	Stellungnahme Überspannungsschutz für sicherheitstechnisch wichtige Gleichstromverbraucher in Kernkraftwerken
19.09.1984	Stellungnahme Kernkraftwerk Grundremmingen II (KRB II), Block C, Ergebnisse der Basisprüfung des Reaktordruckbehälters
19.09.1984	Stellungnahme Interpretation der RSK-Leitlinien Kap. 15 (6) Handhabung eines Brennelement-Transportbehälters
19.09.1984	Stellungnahme Regelvorlage KTA 3301: Nachwärmeabfuhrsystem von Leichtwasserreaktoren
26./27.06.1984	Versuchsanlage beim ehemaligen Heißdampfreaktor (HDR), Rohrleitungsabriß während einer Versuchsserie, mögliche Schlußfolgerungen für Kernkraftwerke
26./27.06.1984	Stellungnahme zur Auslegung eines zukünftigen Natriumgekühlten Reaktors vom Pool-Typ in der Bundesrepublik Deutschland gegen Störfälle mit dem Potential der Kernzerstörung
26./27.06.1984	Stellungnahme Transport von abgetrenntem Plutonium
26./27.06.1984	Stellungnahme zum Vorhaben der Firma Dyckerhoff und Widmann Qualitätssichernde Maßnahmen für die Stillegung von Kernkraftwerken zur Berücksichtigung bei der Planung und Anwendung
26./27.06.1984	Stellungnahme Wasserstoffüberwachungs- und -begrenzungssystem für KWU-Druckwasserreaktoren der Konvoi-Ausführung Konvoi, Wasserstoffüberwachungssystem, Wasserstoffbegrenzungssystem,
26./27.06.1984	Stellungnahme Übertragbarkeit der Ereignisse - Erhöhte Jodabgabe und - Störung in der Abgasanlage im Siedewasserreaktor Philippsburg 1 auf neuere Druckwasserreaktoren
26./27.06.1984	Stellungnahme Regelvorlage KTA 1202, Anforderungen an das Prüfhandbuch (Fassung April 1984)

26./27.06.1984	Stellungnahme Regelvorlage KTA 3704 Notstromanlagen mit Gleichstrom-Wechselstrom-Umformern in Kernkraftwerken, Fassung 4/84
26./27.06.1984	Regeländerungvorlage KTA 3602 Lagerung und Handhabung von Brennelementen, Steuerelementen und Neutronenquellen in Kernkraftwerken mit Leichtwasserreaktoren, Fassung 5/84
26./27.06.1984	Stellungnahme Regelentwurfsvorlage KTA 1408, Qualitätssicherung von Schweißzusätzen und -hilfsstoffen für druck- oder aktivitätsführende Komponenten in Kerntechnischen Anlagen, Teil 1: Eignungsprüfung, Teil 2: Herstellung, Teil 3: Verarbeitung, Fassung 5/
26./27.06.1984	Regelentwurfsvorlage KTA 3703 Notstromerzeugungsanlagen mit Batterien und Gleichrichtergeräten in Kernkraftwerken, Fassung 4/84
26./27.06.1984	Stellungnahme Regeländerungsenntwurfsvorlage KTA 1201 Anforderungen an das Betriebshandbuch, Fassung 4/84
26./27.06.1984	Stellungnahme Regeländerungsentwurfsvorlage KTA 3502, Störfallinstrumentierung, Fassung 2/84
26./27.06.1984	Regelentwurfsvorlage KTA 1404 Dokumentation beim Bau und Betrieb von Kernkraftwerken, Fassung 2/84 (Fraktionsdurchgang)
26./27.06.1984	Stellungnahme Regelentwurfsvorlage KTA 1506, Messung von Ortsdosisleistung in Sperrbereichen von Kernkraftwerken, Fassung 2/84 (Fraktionsdurchgang)
19.06.1984	Stellungnahme Auslegung des Forschungsreaktors FRJ-2 (DIDO) gegen Erdbeben
19.06.1984	Stellungnahme Einführung der thermohydraulischen Analysemethode (THAM) bei Kernkraftwerken mit Siedewasserreaktor
18.04.1984	Stellungnahme Wiederaufarbeitungsanlage Karlsruhe Lagerung und Verdampfung hochaktiver Abfälle (LAVA) - Zwei-Behälter-Betrieb mit Hochaktiv-Waste-Lager als sicherheitstechnische Reserve
21.03.1984	Stellungnahme RSK-Leitlinien für Druckwasserreaktoren, 3. Ausgabe 1981 - Fragen zu Bruchannahmen
21.03.1984	Stellungnahme Kernkraftwerk Stade (KKS) Umrüstung des Sicherheitseinspeisesystems auf ausschließlich heißseitige Einspeisung und Nachweis der ausreichenden Prüfbarkeit der kernnahen Rundnaht des Reaktordruckbehälters
21.03.1984	Stellungnahme Leitlinien für Druckwasserreaktoren, Kap. 21.1 - Ergänzung des Leitlinien-Kapitels 21.1 um Bruchpostulate für austenitische Anschlußleitungen an die Hauptkühlmittelleitung
21.03.1984	Stellungnahme Versagen von Vorsteuerarmaturen der Frischdampf-Sicherheitsventile Relevanz des besonderen Vorkommnisses im Kernkraftwerk Unterweser für andere Kernkraftwerke
21.03.1984	Stellungnahme Offenbleiben eines Sicherheits-Entlastungsventils im Kernkraftwerk Würgassen (21. Oktober 1983)
21.03.1984	Regelvorlage KTA 3204, Reaktordruckbehälter-Einbauten Fassung: Februar 1984
21.03.1984	Stellungnahme Regeländerungsvorlage KTA 3401.3, Reaktorsicherheitsbehälter aus Stahl, Teil Herstellung, Fassung: Februar 1984
21.03.1984	Stellungnahme Regelvorlage KTA 3103 Abschaltsysteme von Leichtwasserreaktoren Fassung Januar 1984
22.02.1984	Stellungnahme Regelentwurfsvorlage KTA 2101.1 Brandschutz in Kernkraftwerken Teil 1: Grundsätze des Brandschutzes, Fassung Januar 1984
22.02.1984	Stellungnahme Regeländerungsvorlage KTA 3201.2 Komponenten des Primärkreises von Leichtwasserreaktoren Teil: Auslegung, Konstruktion und Berechnung, Fassung 10/83
22.02.1984	Stellungnahme Regelvorlage KTA 3203 Fassung 8/83 mit Änderung 12/83 Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserraktoren
25.01.1984	Stellungnahme 1300 MWE Druckwasserreaktoren der KWU Maßnahmen zur Minimierung des Gaseintrags in den Primärkreislauf bei Kühlmittelverluststörfällen - Druckspeicherabsperrung bei kleinen und mittleren Lecks
21.12.1983	Stellungnahme Kernkraftwerk Biblis, Block A Schäden an Dampferzeugerheizrohren
21.12.1983	Stellungnahme Brennelementschäden im Kernkraftwerk Philippsburg 1 (KKP 1) beim betrieblichen Abfahren zur Jahresrevision und Übertragbarkeit auf andere Anlagen

21.12.1983	Stellungnahme Erhöhte Jodfreisetzung in die Anlage und erhöhte Jodabgabe in die Umgebung beim betrieblichen Abfahren des Kernkraftwerkes Philippsburg 1 (KKP 1) zur Jahresrevision
21.12.1983	Stellungnahme Störung in der Abgasanlage (Entzündung von Aktivkohle in der Aktivkohleverzögerungsstrecke) am 14.9.1983 im Kernkraftwerk Philippsburg 1 (KKP 1)
23.11.1983	Stellungnahme Kernkraftwerk Stade (KKS) Stand der Strahlenversprödung des Reaktordruckbehälters
23.11.1983	Stellungnahme Ausfall der automatischen Reaktorschnellabschaltung im Kernkraftwerk Salem 1 (USA) am 22. und 25.02.1983 - Folgerungen für in Betrieb befindliche KWU-Druckwasserreaktoren
23.11.1983	Stellungnahme Regelentwurfsvorlage KTA 3704 Notstromanlagen mit Gleichstrom-Wechselstrom-Umformern, Fassung 9/83
23.11.1983	Stellungnahme Regelentwurfsvorlage KTA 2201.4, Auslegung von Kernkraftwerken gegen seismische Einwirkungen Teil 4: Auslegung der maschinen- und elektrotechnischen Anlagenteile, Fassung 9/82 mit Änderungen gemäß KTA Dok.-Nr. 2201.4/83/1 vom 11.10.1983
23.11.1983	Stellungnahme Regelentwurfsvorlage KTA 1202, Anforderungen an das Prüfhandbuch, Fassung 10/83
19.10.1983	Stellungnahme Brennelementfabrik ALKEM Vorabzustimmungsbedürftige Betriebsumstellungen - Verlegung der Plutoniumnitrat-Lagerung und -Konversion in den Bunkerbereich - Errichtung einer Abfallbehandlungsanlage und eines Abfallagers
19.10.1983	Stellungnahme Brennelementfabrik RBU I Genehmigungsverfahren nach § 7 AtG: Liste der Auslegungsstörfälle
19.10.1983	Stellungnahme Kernkraftwrk Kalkar (SNR 300) Nachweis der Lebensdauer der mechanischen Komponenten
19.10.1983	Stellungnahme KTA-Regelvorlage 3203, Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren, Fassung: August 1983
19.10.1983	Stellungnahme Regeländerungsvorlage KTA 3401.2 Reaktorsicherheitsbehälter aus Stahl Teil: Auslegung, Konstruktion und Berechnung, Fassung: September 1983
19.10.1983	Stellungnahme Regeländerungsvorlage KTA 3902 Auslegung von Hebezeugen in Kernkraftwerken, Fassung: September 1983
19.10.1983	Stellungnahme Sicherheitsbeiräte für Kernkraftwerke
29.09.1983	Stellungnahme Schachtanlage Konrad - Vergleich zweier Varianten zur Nutzung der Schächte
22.06.1983	Stellungnahme der RSK Konzeptrelevante sicherheitstechnische Fragestellungen zu den geplanten Wiederaufarbeitungsanlagen Wackersdorf und Dragahn
22.06.1983	Stellungnahme Sicherheitstechnisch relevante Betriebserfahrungen Ereignis am 1.10.1982 in Biblis B (Schulden an Rückschlagventilen im Not- und Nachkühlsystem, Ereignisnr. 127/82 des Quartalsberichtes IV/82)
22.06.1983	Stellungnahme Neufassung der Richtlinie für den Fachkundenachweis von Kernkraftwerkspersonal (Fassung 13. Mai 1983)
22.06.1983	Stellungnahme Kontinentales Bohrprogramm - Auswahl der Bohrstelle
22.06.1983	Stellungnahme Staatliche Förderung von Forschung und Entwicklung zur Reaktorsicherheit
19.05.1983	Stellungnahme Betriebsbeauftragter für nukleare Sicherheit in Kernkraftwerken
19.05.1983	Stellungnahme Kernkraftwerk Obrigheim (KWO) - Austausch der Dampferzeuger
19.05.1983	Stellungnahme Regelentwurfsvorlage KTA 3505, Typprüfung von Meßwertgebern und Meßumformern des Reaktorschutzsystems, Fassung 02/81
19.05.1983	Stellungnahme Regelentwurfsvorlage KTA 3703 Notstromerzeugungsanlagen mit Batterien und Gleichrichtergeräten, Fassung 04/83
19.05.1983	Stellungnahme Regelentwurfsvorlage KTA 2201.4, Auslegung von Kernkraftwerken gegen seismische Einwirkungen, Teil 4: Auslegung der maschinen- und elektrotechnischen Anlagenteile, Fassung September 1982
19.05.1983	Stellungnahme Regelvorlage KTA 3103 Abschaltsysteme von Leichtwasserreaktoren, Fassung 01/83

19.05.1983	Stellungnahme Regelentwurfsvorlage KTA 1507 Messung gasförmiger, aerosolgebundener radioaktiver Stoffe zur Überwachung der Ableitungen bei Forschungsreaktoren, Fassung 01/83
20.04.1983	Stellungnahme Kernkraftwerk Kalkar (SNR-300) Befunde an der Gitterplatte des Reaktortanks
20.04.1983	Stellungnahme Regeländerungsentwurfsvorlage KTA 3401.3 Reaktorsicherheitsbehälter aus Stahl, Teil: Herstellung
23.03.1983	Stellungnahme der RSK Sicherheitstechnische Realisierbarkeit der FEMO-Technik in der Wiederaufarbeitungsanlage Bayern
23.02.1983	Stellungnahme Kompakte Natriumgekühlte Kernreaktoranlage II (KNK II) Betrieb mit dem 2. Kern
23.02.1983	Stellungnahme Förderungsvorhaben BMI SR 245 Probabilistisches Konzept für die Erschütterungsprüfung von KKW-Einbauteilen hinsichtlich Erdbeben und Flugzeugabsturz
23.02.1983	Stellungnahme Entwurfsvorlage KTA 3506, Systemprüfung der leittechnischen Einrichtungen des Sicherheitssystems, Fassung 12/82
23.02.1983	Stellungnahme Regelentwurfsvorlage KTA 3204 Reaktordruckbehältereinbauten, Fassung 10/82
23.02.1983	Stellungnahme Regelentwurfsvorlage 1507 Messung gasförmiger, aersosolgebundener und flüssiger radioaktiver Stoffe zur Überwachung der radioaktiven Ableitungen bei Forschungsreaktoren, Fassung 1/83
15.12.1982	Stellungnahme Interpretation zu den Sicherheitskriterien für Kernkraftwerke Einzelfehlerkonzept - Grundsätze für die Anwendung des Einzelfehlerkriteriums - Stand 15.12.1982
10.11.1982	Stellungnahme Regelentwurfsvorlage KTA 3902 Auslegung von Hebezeugen in kerntechnischen Anlagen, Fassung Oktober 1982
10.11.1982	Stellungnahme Regelvorlage KTA 3903 Prüfungen und Betrieb von Hebezeugen in kerntechnischen Anlagen (Fassung Juli 1982)
10.11.1982	Stellungnahme Regelentwurfsvorlage KTA 3301 Nachwärmeabfuhrsysteme für Leichtwasserreaktoren (Fassung Oktober 1982)
10.11.1982	Stellungnahme KTA-Regelentwurfsvorlage 3203 Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren, Fassung September 1982
10.11.1982	Stellungnahme Regelvorlage KTA 3201.1 Komponenten des Primärkreises von Leichtwasserreaktoren, Teil Werkstoff, Werkstoffanhang Stahl 20 MnMoNi 55 (Fassung Oktober 1982)
10.11.1982	Stellungnahme KTA-Regheländerungsentwurfsvorlage 3201.2 Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Auslegung, Konstruktion und Berechnung, Fassung Juli 1982
10.11.1982	Stellungnahme Regeländerungsentwurfsvorlage KTA 3401.2 Reaktorsicherheitsbehälter aus Stahl, Teil: Auslegung, Kosntruktion und Berechnung, Fassung Juli 1982
10.11.1982	Stellungnahme Regeländerungsentwurfsvorlage KTA 3401.3 Reaktorsicherheitsbehälter aus Stahl, Teil: Herstellung, Fassung Juli 1982
13.10.1982	Stellungnahme Versuchatomkraftwerk Kahl (VAK) - Temperatur- und Druckerhöhung in Primärkreis auf die früher genehmigten Werte
13.10.1982	Stellungnahme ALKEM - Störfälle und Einwirkungen von außen
13.10.1982	Stellungnahme Regelvorlage KTA 3502 Störfallinstrumentierung (Fassung 9/82)
13.10.1982	Stellungnahme Regelentwurfsvorlage KTA 3507 Werksprüfung an leittechnischen Geräten des Sicherheitssystems (Fassung 9/82)
13.10.1982	Stellungnahme Regelvorlage KTA 2201.2 Auslegung von Kernkraftwerken gegen seismische Einwirkungen Teil 2: Baugrund (Fassung 9/82)
13.10.1982	Stellungnahme Regeländerungsentwurfsvorlage KTA 3401.2 Reaktorsicherheitsbehälte aus Stahl, Teil: Auslegung, Konstruktion und Berechnung (Fassung 7/82)
15.09.1982	Stellungnahme Kommentar zur Neufassung des Abschnitts 21.1 der RSK-Leitlinien für Druckwasserreaktoren
15.09.1982	Stellungnahme Kernkraftwerk Isar 1 Langzeitbetrieb des Reaktordruckbehälters
23.06.1982	Stellungnahme Prüfumfang zur Feststellung des Abscheidegrads von Jodsorptionsfiltern zur Störfallbeherrschung in Kernkraftwerken

23.06.1982	Stellungnahme zum weiteren Genehmigungsverfahren des THTR-300
23.06.1982	Stellungnahme zum weiteren Genehmigungsverfahren des Kernkraftwerkes Kalkar (SNR-300)
23.06.1982	Stellungnahme Auslegung der Gitterplatteneinsätze für den 2. Kern der Kompakten Natriumgekühlten Kernreaktoranlage II (KNK II)
23.06.1982	Stellungnahme Gemeinschaftskernkraftwerk Neckar (GKN 1)
23.06.1982	Stellungnahme Auswahl des Referenzverfahrens der direkten Brennelement-Endlagerung für die sicherheitstechnischen Vergleich zwischen Entsorgung mit und ohne Wiederaufarbeitung
23.06.1982	Stellungnahme der RSK zum "äußeren Gebäudesprühsystem" zur Eingrenzung von Folgen eines postulierten Unfalls in einem 1300 MWe-Druckwasserreaktor (Restrisikominderung)
19.05.1982	Regelvorlage KTA 3701.2 Übergeordnete Anforderungen an die elektrische Energieversorgung des Sicherheitssystems in Kernkraftwerken, Teil 2: Kernkraft-Mehrblockanlagen, Fassung 5/82
19.05.1982	Regelvorlage KTA 3503 Typprüfung von elektrischen Baugruppen des Reaktorschutzsystems, Fassung 3/82
19.05.1982	Stellungnahme Grundsätze für die Anwendung des Einzelfehlerkriteriums
19.05.1982	Stellungnahme Kabeldurchführungen Untersuchungsergebnisse zur Störfallfestigkeit
19.05.1982	Stellungnahme Wiederaufarbeitungsanlage Hessen Standortvorauswahl (Standorte Merenberg und Wangershausen)
19.05.1982	Stellungnahme Kernkraftwerk Brunsbüttel - Umrüstkonzept innerhalb des Sicherheitsbehälters
24.04.1982	Stellungnahme Regelentwurfsvorlage KTA 3407, Rohrdurchführungen in Reaktor-Sicherheitsbehälter, Fassung 10/84
21.04.1982	Stellungnahme Endlagerung radioaktiver Abfälle in Salzstöcken - Störfälle als Folge des Zuflusses von Wässern oder Salzlösungen in das Endlagerbergwerk
21.04.1982	Stellungnahme Kernkraftwerk Würgassen Austausch von Rohrleitungen innerhalb des Sicherheitsbehälters
21.04.1982	Stellungnahme der RSK zum IFEU-Bericht Nr. 16 1982 "Sekundärkreisemissionen von Leichtwasserreaktoren bei Normalbetrieb und ausgesuchten Störfällen"
17.03.1982	KTA-Regelentwurfsvorlage 3103 Abschaltsysteme von Leichtwasserreaktoren Fassung 1/8
17.03.1982	Regelentwurfsvorlage KTA 3301.1 Nachwärmeabfuhrsysteme für Leichtwasserreaktoren Teil: Systemtechnik, Anordnung und Betrieb, Fassung 1/82
17.03.1982	Regelentwurfsvorlage KTA 3203 Überwachung der Strahlenversprödung von Werkstoffen des Reaktordruckbehälters von Leichtwasserreaktoren, Fassung 2/82
17.03.1982	Regeländerungentwurfsvorlage KTA 3401.1, Reaktorsicherheitsbehälter aus Stahl, Teil: Werkstoffe, Werkstoffanhänge WA bis WG, Fassung 1/82
17.03.1982	Stellungnahme der RSK, Kernkraftwerk Stade - Strahlenversprödung des Rektordruckbehälters
17.03.1982	Stellungnahme der RSK, Kernkraftwerk Obrigheim - Stand der Strahlenversprödung des Reaktordruckbehälters
17.03.1982	Stellungnahme der RSK zur Regelentwurfsvorlage KTA 2201.2, Auslegung von Kernkraftwerken gegen seismische Einwirkungen, Teil 2 Baugrund, Fassung 6/81
17.03.1982	Stellungnahme der RSK zur Regelvorlage KTA 3201.4, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Wiederkehrende Prüfungen und Betriebsüberwachung, Fassung 2/82
17.03.1982	Stellungnahme derRSK zur KTA-Regel 3201.1, Komponenten des Primärkreises von Leichtwasserreaktoren, Werkstoffanhang WA, Fassung 1/82 Teil Werkstoffe
17.03.1982	Stellungnahme der RSK zur Regelvorlage KTA 3701.2, übergeordnete Anforderungen an die elektrische Energieversorgung des Sicherheitssystems in Kernkraftwerken, Teil 2: Kernkraft-Mehrblockanlagen Fassung 12/81
17.02.1982	Relevante Störfälle bei "Anderen Entsorgungstechniken"
17.02.1982	Stellungnahme Kommentar zur Neufassung des Abschnitts 21.1 der RSK-Leitlinien für Druckwasserreaktoren

20.01.1982	KTA-Regeländerungsentwurf 3902, Hebezeuge in Kerntechnischen Anlagen
16.12.1981	Stellungnahme "Kernkraftwerk Kalkar, SNR-300" Beschleifen ausgewählter Schweiß-nahtwurzelraupen der Hauptrohrleitungen
16.12.1981	Stellungnahme der RSK zur Regelvorlage KTA 3201.4, Komponenten des Primär-kreislaufs von Leichtwasserreaktoren, Teil: Wiederkehrende Prüfungen und Betriebs-überwachung, Fassung 10/81
16.12.1981	Stellungnahme der RSK zur Regelentwurfsvorlage KTA 3702.2, Notstromerzeu-gungsanlagen mit Dieselaggregaten in Kernkraftwerken, Teil 2, Prüfungen, Fassung 9/81
16.12.1981	Stellungnahme der RSK zur Regelentwurfsvorlage KTA 3507, Werksprüfung an leit-technischen Geräten
11.11.1981	Strahlungsinduzierte Zündung explosionsfähiger Stoffgemische aus organischen Lö-sungsmitteln und Entstehung explosionsfähiger Gemische durch Strahlungsquellen
11.11.1981	Entsorgung abgebrannter Brennelemente aus AVR und THTR-300
11.11.1981	Wassereinbruch über den Schacht eines Endlagers Schachtausbausystem mit einem Stahl-Beton-Verbundausbau mit Asphalthinterfüllung, wie für den Salzstock Gorleben vorgeschlagen
11.11.1981	Kernkraftwerk Biblis B Abriß von zwei Impulsleitungen am Druckhalter und mögliche Konsequenzen für die Druckwasserreaktoren Biblis A, Neckarwestheim (GKN), Un-terweser (KKU), Obrigheim (KWO) und Stade (KKS)
11.11.1981	Stellungnahme zur Regelentwurfsvorlage KTA 2201.2 Auslegung von Kernkraftwer-ken gegen seismische Einwirkungen, Teil 2: Baugrund, Fassung 6/81
11.11.1981	Stellungnahme zur Regeländerungsentwurfsvorlage KTA 3902, Auslegung von He-bezeugen in kerntechnischen Anlagen, Fassung 10/81
11.11.1981	Stellungnahme zur Regelentwurfsvorlage KTA 3903, Prüfungen an Hebezeugen in kerntechnischen Anlagen, Fassung 10/81
11.11.1981	Stellungnahme zur Regelentwurfsvorlage KTA 2207, Schutz von Kernkraftwerken gegen Hochwasser, Fassung 9/81
14.10.1981	Betriebsberichte zur Information der RSK
14.10.1981	KTA-Regelentwurf 3201.1 Komponenten des Primärkreises von Leichtwasserreakto-ren Teil: Werkstoff Anhang A: Werkstoffkenndaten f•r den Vergütungsstahl 20 MnMoNi 5 5
14.10.1981	Stellungnahme der RSK Regelentwurfsvorlage KTA 3502 Störfallinstrumentierung, Fassung 9/81
16.09.1981	Kernkraftwerk Kalkar (SNR 300) Anwendbarkeit mechanisierter Schweiß- und Schleifverfahren bei der Herstellung der Rundnähte der Hauptrohrleitung
16.09.1981	Stellungnahme Sicherheitsstudie für HTR-Konzepte unter deutschen Standortbedin-gungen
16.09.1981	Betriebszustände bei schwergestörtem Reaktorkern und Fachkunde des verantwortli-chen Schichtpersonals
16.09.1981	KTA-Regelentwurfsvorlage 3205.1, Komponentenstütz-Konstruktionen nichtintegral-ten Anschlüssen, Teil 1
16.09.1981	Stellungnahme der RSK Auslegung des Tanklagers LAVA der Wiederaufarbeitungs-anlage Karlsruhe gegen Flugzeugabsturz
16.09.1981	Stellungnahme der RSK Annahmen zur Freisetzung von Radiolysewasserstoff aus dem Lager für radioaktive Abfälle einer Wiederaufarbeitungsanlage
16.09.1981	Stellungnahme der RSK zu den Ausführungen über Fragen der Sicherheit von Leichtwasserreaktoren und Reaktorsystemen neuerer technologischer Varianten so-wie zu Problemen der Entsorgung im Sondergutachten ENERGIE UND UMWELT des Rates der Sachverständige
01.07.1981	Stellungnahme der RSK Richtlinie für den Inhalt der Fachkundeprüfung des verant-wortlichen Schichtpersonals in Kernkraftwerken Folgerungen aus dem TMI-Störfall bezüglich Ergänzungen und Korrekturen
01.07.1981	Leitlinien zur Beurteilung der Auslegung von Kernkraftwerken gegen Störfälle (§ 28 Abs. 3 StrlSchV)
01.07.1981	Regelentwurfsvorlage KTA 3401.1, Reaktorsicherheitsbehälter aus Stahl, Anhänge A bis G, Werkstoffkenndaten

01.07.1981	Stellungnahme der RSK Eignung des Salzstocks Gorleben für die Endlagerung radioaktiver Abfälle Ergebnisse der bisher durchgeführten Standortuntersuchungen
01.07.1981	Stellungnahme der RSK Gemeinschaftskernkraftwerk Neckar (GKN) Ausbau der Gleichstromanlage
01.07.1981	Stellungnahme der RSK Kernkaftwerk Brunsbüttel Befristeter Weiterbetrieb der Schnellabschaltbehälter
01.07.1981	Stellungnahme der RSK Standort Neupotz
01.07.1981	Stellungnahme der RSK Regelvorlage KTA 3601, Lüftungstechnische Anlagen in Kernkraftwerken, Kapitel 8: Leittechnik (Fassung 2/81)
01.07.1981	Stellungnahme der RSK Regeländerungsentwurfsvorlage KTA 3702.2, Notstromerzeugungsanlagen mit Dieselaggegaten in Kernkraftwerken, Teil 2: Prüfungen (Fassung 2/81)
20.05.1981	Kernkraftwerk Kalkar (SNR-300) System zur Detektion verzögerter Neutronen (DND-System)
29.04.1981	Kompakte Natriumgekühlte Kernreaktoranlage Karlsruhe (KNK-II) Versuchsbetrieb mit abgeschaltetem Reaktimetergrenzwert
29.04.1981	Zentrales Zwischenlager Gorleben für die trockene Lagerung von abgebrannten Brennelementen in Transportbehältern
29.04.1981	Politisierung der RSK-Beratungstätigkeit
18.03.1981	Kernkraftwerk Kalkar, SNR 300 - Untersuchungen und vorgesehene Sanierungsmaßnahmen am Reaktortank
18.03.1981	KTA-Regelentwurfsvorlage 3503, Gerätespezifische Eignungsprüfung von elektrischen Baugruppen und des Reaktorschutzsystems, Fassung Dezember 1980
18.03.1981	KTA-Regeländerungsvorlage 3901, Kommunikationsmittel für Kernkraftwerke, Fassung Dezember 1980
18.03.1981	Stellungnahme zu den im NRC-Bericht NUREG-0625 vorgeschlagenen neuen amerikanischen Standortkriterien für Kernkraftwerke
18.02.1981	Kernkraftwerk Grohnde - Sicherheit des Rektordruckbehälters und des Sicherheitsbehälters
18.02.1981	KTA Regelentwurfsvorlage 3202.1, Klassifizierung druckführender und aktivitätsführender Systeme und Komponenten in Kernkraftwerken mit Leichtwasserreaktoren und Grundsätze zur Festlegung qualitätssichernder Maßnahmen
17.12.1980	Kernkraftwerk Grafenrheinfeld Inbetriebnahme: Qualität der Druckführenden Umschließung und der äußeren Systeme
17.12.1980	Kernkraftwerk Würgassen Fehler in Elektroschlacke-Schweißnähten des Reaktordruckbehälters
17.12.1980	Befristeter Weiterbetrieb der Schnellabschaltbehälter
17.12.1980	Kernkraftwerk Kalkar (SNR 300) - Wiederkehrende Prüfungen am Reaktortank
17.12.1980	Stellungnahme der RSK zur Ermittlung des Brennelement-Zustands bei trockener Lagerung in Transportbehältern und zur Bereitstellung besonderer Vorkehrungen für den Transport beschädigter Behälter
12.11.1980	Grundsätze zur Überprüfung des Vollzuges angeordneter wiederkehrender Prüfungen
15.10.1980	Gemeinschaftkraftwerk Neckar (GKN) - Wiederkehrende Prüfungen
15.10.1980	Mehrzweckforschungsreaktor (MZFR) - Stellungnahme zum Weiterbetrieb
15.10.1980	Qualität längsnahtgeschweißter Rohrleitungen aus dem Stahl WB 36 Kernkraftwerk Mülheim-Kärlich
15.10.1980	Qualität längsnahtgeschweißter Rohrleitungen aus dem Stahl WB 36 Kernkraftwerk Grundremmingen 2
15.10.1980	Qualität längsnahtgeschweißter Rohrleitungen aus dem Stahl WB 36 Kernkraftwerk Krümmel
15.10.1980	Verwendung von Stahlguß GS 18 NiMoCr 37 für Hauptkühlmittelpumpen
15.10.1980	KTA-Regelentwurfsvorlage 3201.4, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Betrieb und Prüfung

15.10.1980	Regeländerungsvorlage KTA 3403, Kabeldurchführungen im Reaktorsicherheitsbehälter von Kernkraftwerken, Fassung 9/80
15.10.1980	Regeländerungsentwurfsvorlage KTA 3501, Reaktorschutzsystem und Überwachung von Sicherheitseinrichtungen Fassung 8/80
15.10.1980	Stellungnahme der RSK zur KTA-Regelvorlage 3601: Lüftungstechnische Anlagen in Kernkraftwerken
15.10.1980	Stellungnahme der RSK zur KTA-Regelentwurfsvorlage 3604, Lagerung und Handhabung radioaktiver Stoffe in Kernkraftwerken
24.09.1980	Stellungnahme der RSK zum Antrag der GSF für die rückholbare Zwischenlagerung von schwachradioaktiven Abfällen in der Asse II
24.09.1980	Kernkraftwerk Kalkar (SNR 300) Konzept für wiederkehrende Prüfungen
24.09.1980	Kernkraftwerke mit Siedewasserreaktoren Austausch von Rohrleitungen und Komponenten innerhalb des Sicherheitsbehälters - Kernkraftwerk Würgassen (KWW)
24.09.1980	Kernkraftwerke mit Siedewasserreaktoren Austausch von Rohrleitugen und Komponenten innerhalb des Sicherheitsbehälters - Kernkaftwerk Philippsburg (KKP 1)
24.09.1980	KTA-Regelvorlage 3201.2, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Auslegung, Konstruktion und Berechnung
24.09.1980	KTA-Regelentwurfsvorlage 3202, Klassifizierung druckführender und aktivitätsführender Systeme und Komponenten in Kernkraftwerken mit Leichtwasserreaktoren und Grundsätze zur Festlegung qualitätssichernder Maßnahmen
24.09.1980	KTA-Regelentwurfsvorlage 3401.4, Reaktorsicherheitsbehälter aus Stahl, Teil: Wiederkehrende Prüfungen
24.09.1980	KTA-Regeländerungsvorlage 3403, Kabeldurchführungen im Reaktorsicherheitsbehälter von Kernkraftwerken
25.06.1980	Kernkraftwerk Brokdorf - Sicherheitskonzept und Verwendung des Stahls Aldur 50/65 D für den Sicherheitsbehälter
25.06.1980	Kernkraftwerk Obrigheim Stand der Beurteilung der Strahlenversprödung und Vorgehen bei der zerstörungsfreien wiederkehrenden Prüfung des Reaktordruckbehälters
25.06.1980	Prototyp-Kernkraftwerk Hamm-Uentrop (THTR-300) Anlagenteile des Wasser-Dampf-Kreislaufs
25.06.1980	KTA-Regeländerungsvorlage 3403, Kabeldurchführungen im Rektorsicherheitsbehälter von Kernkraftwerken, Fassung März 1980
25.06.1980	Richtlinie über die Gewährleistung der notwendigen Kenntnisse der beim Betrieb von Kernkraftwerken sonst tätigen Personen i.S.v. § 7 Abs. 2 Nr. 2 AtG (BMI, Stand 31. Mai 1980)
21.05.1980	Fehlererkennung bei wiederkehrenden Prüfungen mittels zerstörungsfreier Verfahren
21.05.1980	VdTÜV-Werkstoffblätter Warmfester Feinkorn-Vergütungsstahl 20 MnMoNi 55, Werkstoff-Nr. 1.6310
21.05.1980	KTA-Regelvorlage 3401.1, Reaktorsicherheitsbehälter aus Stahl, Teil: Werkstoffe
21.05.1980	KTA-Regeländerungsvorlage, Ergänzung der KTA-Regel 3201.1, Teil: Werkstoffe-Ergänzung A: Werkstoffkenndaten für den Vergütungsstahl 20 MnMoNi 55 für nahtlose Hohlteile, geschmiedet oder gewalzt, Fassung März 1980
21.05.1980	KTA-Regelvorlage 2101, Anforderungen an das Betriebshandbuch, Fassung 4/80
23.04.1980	Möglichkeiten zur Verbesserung des Simulatortrainings - Vorhaben der GRS/KWU "Untersuchungen zu Verbesserungmöglichkeiten des Simulatortrainings von Betriebspersonal im Hinblick auf die Störfallbeherrschung in DWR"
23.04.1980	Kernkraftwerk Brunsbüttel
23.04.1980	Kernkraftwerk Brunsbüttel - Vorkommnis am 2.3.1980 ungeplanter Wasseranfall im Reaktorgebäude
23.04.1980	KTA-Regelvorlage 3401.2, Reaktorsicherheitsbehälter aus Stahl, Teil: Auslegung, Konstruktion und Berechnung
29.03.1980	Zusammenspiel zwischen Antragsteller und/oder Anlagenlieferer, Komponentenhersteller und den von den zuständigen Behörden zugezogenen Sachverständigen bei der Herstellung von Komponenten der Druckführenden Umschließung

19.03.1980	Kernkraftwerke Brunsbüttel, Isar, Philippsburg 1, Würgassen -Verfahrenstechnische Aspekte der Bespeisung von Siedewasserreaktoren, insbesondere bei An- und Abfahrvorgängen im Schwachlastbetrieb -Beanspruchungen der Speisewasserleitungen im Betrieb
19.03.1980	KTA-Regeländerungsentwurf 1201 (Fassung 10/79) Anforderungen an das Betriebshandbuch
19.03.1980	KTA-Regelvorlage (Entwurf) 3702.1, Notstromerzeugungsanlagen mit Dieselaggregaten, Teil 1: Auslegung, Fassung Dezember 1979
20.02.1980	KTA-Regelentwurf 3201.2, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Auslegung, Konstruktion, Berechnung KTA-Regelentwurf 3401.1, Reaktorsicherheitsbehälter aus Stahl, Teil: Werkstoffe, KTA-Regelentwurf 3401.2, Reaktorsicherheitsbehälte
20.02.1980	Rahmenspezifikation Basissicherheit Anwendung im Genehmigungsverfahren KKP II
18.02.1980	KTA-Regelentwurfsvorlage 3503, Gerätespezifische Eignungsprüfung von elektrischen Baugruppen des Reaktorschutzsystems, Fassung Dezember 1980
18.02.1980	KTA-Regelentwurfsvorlage 3701.2, übergeordnete Anforderungen an die elektrische Energieversorgung des Sicherheitssystems in Kernkraftwerken, Teil 2: Kernkraft-Mehrblockanlagen, Fassung November 1980
23.01.1980	Kernkraftwerk Unterweser (KKU) Ertüchtigungsmaßnahmen zur Beherrschung eines Frischdampfleitungsbruches mit gleichzeitigem Dampferzeugerheizrohrversagen uhnd Offenbleiben eines Sicherheitsventils mit gleichzeitigem Dampferzeugerheizrohrversagen
23.01.1980	Kernkraftwerk Mülheim-Kärlich- Restpunkte zum Reaktordruckbehälter aus der 140. RSK-Sitzung am 20. Dezember 1978
19.12.1979	Langfristige Kühlung der Brennelementlagerbecken in Kernkraftwerken nach einem Kühlmittelverluststörfall unter Berücksichtigung der Kompaktlagerung
19.12.1979	Gemeinschaftskernkraftwerk Neckar (GKN) Ausfall der Gleichstromversorgung am 31.8.1979 während der Revisionsphase
19.12.1979	Kernkraftwerk Stade Beurteilung und Absicherung des Tragverhaltens der kernnahen Bereiche des Reaktordruckbehälters
14.11.1979	Kernkraftwerk Kalkar (SNR 300) Integrität des Reaktor-Tankauflageträgers (RTAT) beim Bethe-Tait-Störfall
14.11.1979	Gedrehte Stutzen aus Blech am Sicherheitsbehälter der Kernkraftwerke Brunsbüttel und Isar
17.10.1979	Kernkraftwerk Würgassen Ergebnisse der wiederkehrenden Prüfungen am Reaktordruckbehälter
17.10.1979	Gemeinschaftskernkraftwerk Neckar (GKN) Ergebnis der Prüfungen am Druckhalter-Abblasebehälter und an den Druckspeichern
17.10.1979	KTA-Regeländerungentwurfsvorlage 1201 Anforderungen an das Betriebshandbuch, Fassung 10/79
19.09.1979	Stellungnahme der Reaktor-Sicherheitskommission zu den Ergebnissen über die Veranstaltung Rede-Gegenrede der Niedersächsischen Landesregierung über die Realisierbarkeit eines nuklearen Entsorgungszentrums vom 28.3. bis 3.4.1979 in Hannover
19.09.1979	Stellungnahme der Reaktor-Sicherheitskommission (RSK) zu kritischen Äußerungen von Prof. Dr. Grimmel bezüglich der Eignung der norddeutschen Salzstöcke für die Endlagerung radioaktiver Abfälle
19.09.1979	KTA-Regelvorlage 1401, Allgemeine Anforderungen an die Qualitätssicherung
19.09.1979	KTA-Regelentwurfsvorlage 3201.2, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Auslegung, Konstruktion, Berechnung
19.09.1979	KTA-Regelentwurfsvorlage 3401.2, Reaktorsicherheitsbehälters aus Stahl, Teil: Auslegung, Konstruktion, Berechnung
19.09.1979	KTA-Regelvorlage 3401.3, Reaktorsicherheitsbehälter aus Stahl, Teil: Herstellung
20.06.1979	GKN Stellungnahme zu Maßnahmen zur Erfüllung der Auflagen der 2. Teilbetriebsgenehmigung bezüglich der Beherrschung eines Lecks am Reaktordruckbehälter sowie zur Reduzierung der radiologischen Belastung infolge eines Frischdampfleitungsbruches
20.06.1979	Störfall im Kernkraftwerk Three Mile Island 2 (TMI 2) Maßnahmen für in der Bundesrepublik Deutschland in Errichtung befindliche Kernkraftwerke mit Druckwasserreaktoren aufgrund der aus dem Störfall gewonnenen Erkenntnisse

20.06.1979	Bestandsaufnahme der Sicherheitsvorkehrungen in der Bundesrepublik Deutschland in Betrieb befindlicher Kernkraftwerke
20.06.1979	KTA-Regelentwurf 3201.3 Komponenten des Primärkreises von Leichtwasserreaktoren Teil: Herstellung
20.06.1979	KTA-Regelentwurfvorlage 3401.1 Reaktorsicherheitsbehälter aus Stahl Teil: Werkstoffe
16.05.1979	KTA-Regelvorlage 3404, Abschließung der den Sicherheitsbehälter durchdringenden Rohrleitungen
16.05.1979	Versuchsatomkraftwerk Kahl (VAK) - Sicherheit der sekundärseitigen Speisewasserversorgung und der Wärmeabfuhrsysteme
16.05.1979	Stellungnahme zu sicherheitstechnischen Fragestellungen im "Sachstandsbericht zur Entsorgung von Kernkraftwerken" der Arbeitsgemeinschaft (AGK) der IGM/ÖTV vom 25.1.1978 (Fassung D)
16.05.1979	Stellungnahme der RSK zum Entwurf einer "Richtlinie für Prorgramme zur Erhaltung der Fachkunde des verantwortlichen Schichtpersonals in Kernkraftwerken" Stand: 27. November 1978
04.04.1979	Stellungnahme der Reaktor-Sicherheitskommission zum Störfall im Kernkraftwerk Three Mile Island 2 am 28.3.1979
21.03.1979	Anforderungen an das Prüfpersonal bei zerstörungsfreien Prüfungen
21.03.1979	Stellungnahme zu sicherheitstechnischen Aspekten der unterirdischen Bauweise von Kernkraftwerken, Beantwortung von BMI-Fragen
21.02.1979	Kernkraftwerk Brunsbüttel - Splitterschutzkonstruktion für die Schnellabschaltbehälter
21.02.1979	Beurteilung des langfristigen Weiterbetriebs des Druckhalters aus dem Stahl Altherm NiMoV
21.02.1979	Überarbeitung der KTA-Regel 3403, Kabeldurchführungen im Reaktor-Sicherheitsbehälter an Kernkraftwerken (11/76)
24.01.1979	KTA-Regelvorlage 3405: Integrale Leckratenprüfung des Sicherheitsbehälters mit der Absolutdruckmethode (Fassung 1/79)
24.01.1979	KTA-Regelvorlage 3401.3, Reaktorsicherheitsbehälter aus Stahl, Teil: Herstellung
20.12.1978	BBR-Reaktordruckbehälter Stellungnahme zum Reaktordruckbehälter und zur Prüfung des Sicherheitsbehälters des Kernkraftwerkes Mülheim-Kärlich und zur konstruktiven Weiterentwicklung der BBR-Reaktordruckbehälter für zukünftige Anlagen
20.12.1978	Kernkraftwerk Obrigheim Stellungnahme zur Strahlenversprödung des Reaktordruckbehälters
20.12.1978	Kernkraftwerk Kalkar (SNR 300) Stellungnahme zur erdbebensicheren Auslegung der maschinentechnischen und elektrotechnischen Komponenten unter Berücksichtigung seismisch induzierter Drehschwingungen um eine vertikale Achse des Reaktorgebäudes
11.10.1978	Kernkraftwerk Brunsbüttel, Störfall am 18.06.1978
11.10.1978	Kernkraftwerk Biblis C Werkstoffwahl für die Druckführende Umschließung und Druckbehälter Sprödbruchdiagramme des Reaktordruckbehälters
11.10.1978	KTA-Regel 3201.1 Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Werkstoffe
20.09.1978	NS Otto Hahn - Fragen zur Sicherheit des Schiffes
20.09.1978	Kernkraftwerk Obrigheim, Langfristige Absicherung des Betriebs des Reaktordruckbehälters
20.09.1978	Kernkraftwerk Obrigheim, Qualitätsstand der Hauptkühlmittelleitungen
20.09.1978	Kernkraftwerk Grundremmingen 1 (KRB 1) Ergebnisse von Untersuchungen an Komponenten aus unstabilisiertem Austenit
20.09.1978	Kernkraftwerk Philippsburg 2 (KKP 2) - Ausführung der Frischdampfleitung innerhalb des Sicherheitsbehälters
20.09.1978	Stellungnahme der RSK zum Konzept der Nachrüstung von Kernkraftwerken
20.09.1978	Stellungnahme der RSK zur KTA-Regelentwurfsvorlage 3401.3, Reaktorsicherheitsbehälter aus Stahl, Teil: Herstellung und Prüfung

20.09.1978	Stellungnahme der RSK zur KTA-Regelentwurfsvorlage 3201.1. Komponenten des Primärkreises von Leichtwasserreaktoren, Teil: Werkstoffe
20.09.1978	Stellungnahme der RSK zu KTA-Regelentwurfsvorlage 3201.3, Komponenten des Primärkreises, Teil: Herstellung
16.08.1978	Stellungnahme der RSK zum Weiterbetrieb des Kernkraftwerks Isar
21.06.1978	Stellungnahme der RSK zur KTA-Regelvorlage 3701.1: übergeordnete Anforderungen an die elektrische Energieversorgung des Sicherheitssystems, Teil: Einblockanlagen, Fassung Juni 1978
21.06.1978	Kernkraftwerk Kalkar (SNR-300), Stellungnahme zum Vorschlag des Antragstellers für den 2. Sprengversuch, 1 : 6-Tankmodell
21.06.1978	Stellungnahme über Restpunkte aus dem Leitlinienvergleich und über Restpunke aus der 133. RSK-Sitzung zur nuklearen Inbetriebnahme
21.06.1978	Stellungnahme der RSK zur Bewertung der Ergebnisse von integralen Leckratenwiederholungsprüfungen
21.06.1978	Kernkraftwerk Philippsburg 2 (KKP 2) - Konzept der Druckspeicher
21.06.1978	Stellungnahme zum Sicherheitsstatus des Kernkraftwerk Obrigheim (KWO)
17.05.1978	KTA-Regelentwurfsvorlage 1401: Allgemeine Anforderungen an die Qualitätssicherung
17.05.1978	KTA-Regel 3403, Stellungnahme zu den Kabeldurchführungen am Reaktorsicherheitsbehälter
19.04.1978	Kernkraftwerk Philippsburg 1 (KKP 1) Stellungnahme zu Inbetriebnahme und Betrieb
19.04.1978	Richtlinie für den Inhalt der Fachkundeprüfung des Schichtpersonals in Kernkraftwerken - Stellungnahme der RSK
19.04.1978	Kernkraftwerk Philippsburg 2 (KKP 2) Stellungnahme zur Auslegung der Druckspeicher
15.03.1978	Kernkraftwerk Kalkar (SNR 300) Stellungnahme der RSK zur Verfizierung des INTERATOM-Rechenprogramms ARES
14./15.02.1978	Stellungnahme zur KTA-Regelvorlage 3701.1, Notstromversorgung von Kernkraftwerken, übergeordnete Anforderungen, Januar 1978
14./15.02.1978	Stellungnahme zur Richtlinie für die Vorbereitung und Durchführung von Instandhaltungsarbeiten und Änderungsarbeiten in Kernkraftwerken (Stand: Dezember 1977)
18.01.1978	KTA-Regelentwurf 3201.1 Werkstoffe - Komponenten des Primärkreislaufs von Leichtwasserreaktoren
18.01.1978	Stellungnahme zur KTA-Regelvorlage 3404: Abschließung der den Reaktorsicherheitsbehälter durchdringenden Rohrleitungen (Fassung 11/77)
21.12.1977	Wiederaufarbeitung von Brennelementen aus deutschen Leichtwasserreaktoren in La Hague und französisches Verglasungsverfahren AVM (Atelier de Vitrification de Marcoule)
21.12.1977	SR-Vorhaben Nr. 8 des BMI "Bestandsaufnahme der sicherheitstechnischen Auslegung von Kernkraftwerken mit DWR, SNR, HTR und der mit dem Betrieb verbundenen Strahlenbelastung"
21.12.1977	Kernkraftwerk Brunsbüttel Vorgehensweise bei den Gußarmaturen in der Druckführenden Umschließung
21.12.1977	Reaktorsicherheitsbehälter aus Stahl Zähigkeitswerte und Druckprobenbedingungen an den Sicherheitsbehältern der Kernkraftwerke Grafenrheinfeld und Grohnde
23.11.1977	Brennelementfertigungsanlage der Firma Exxon 3. Teilerrichtungsgenehmigung
19.10.1977	Entsorgung Verknüpfung der Entsorgung mit der Genehmigung von Kernkraftwerken
21.09.1977	Kernkraftwerk Philippsburg 1 (KKP-1) Reparatur der Speisewasserstutzen sowie weitere Restfragen zur nuklearen Inbetriebnahme
21.09.1977	Druckhalter der Kernkraftwerke Biblis A, Biblis B, Gemeinschaftskernkraft-werk Neckar (GKN) und Kernkraftwerk Unterweser (KKU)
22.06.1977	Kernkraftwerk Gundremmingen 1 (KRB-1) Stufenplan für die Reparatur und die Ertüchtigungsmaßnahmen
22.06.1977	Kernkraftwerk Würgassen (KWW) Prüfung des Reaktordruckbehälters mit Ultraschall Erhöhung der Kraftwerksleistung auf über 80 % und Leistungsprüfungen bei 93 % der Nennleistung

22.06.1977	Kernkraftwerke Philippsburg 1 (KKP-1), Isar (KKI) Restpunkte aus der Inbetriebnahme - Empfehlung
22.06.1977	Kompaktlagerung von Brennelementen in Kernkraftwerken, die sich in Betrieb befinden bzw. zur Inbetriebnahme anstehen
22.06.1977	Auslegung von Kernkraftwerken gegen Erdbeben Stellungnahme zu eventuellen Erkentnissen aus den Erdbeben im Friaul (1976) und in Rumänien (1977) im Hinblick auf die Beurteilung der seisomologischen Verhältnisse in Deutschland
22.06.1977	Kernkraftwerk Philippsburg 2 (KKP-2) Ergänzende Stellungnahme zur RSK-Empfehlung der 119. RSK-Sitzung am 15. Dezember 1976
22.06.1977	Zuverlässigkeit von Relaissystemen in Kernkraftwerken
18.05.1977	Vorliegende Ergebnisse des Untersuchungsprogrammes Altherm NiMo V
18.05.1977	Sicherheitsbeiräte für Kernkraftwerke in der Bundesrepublik Deutschland
18.05.1977	Kernkraftwerk Kalkar (SNR-300) Errichtung des Biologischen Schildes
18.05.1977	KTA-Regelvorhaben
20.04.1977	Gemeinschaftskernkraftwerk Neckar (GKN) Programm der wiederkehrenden Prüfungen an den Druckspeichern
20.04.1977	Kernkraftwerk Würgassen Werkstoffauswahl, Auslegung und Konstruktion der neuen Schnellabschalt-behälter
20.04.1977	Kernkraftwerke Isar (KKI) und Philippsburg 1 (KKP-1) Wiederkehrende Prüfungen an den Schnellabschaltbehältern
16.03.1977	Werkstoffe für Primärkreiskomponenten (20 MnMoNi 55 und modifizierter NiMoCr 3 7)
16.03.1977	Kernkraftwerk Biblis A, B, KKI Auslegung und Konstruktion des neuen Speisewasserbeh„lters
16.03.1977	Sicherheitskriterien und Leitlinien des BMI
16.02.1977	Kernkraftwerk Brunsbüttel Zuverlässigkeit der Rückschlagarmaturen im Kernsprühsystem und im Nachkühlstrang Versagen des Dampferzeugertrennblechs
19.01.1977	Regelvorhaben des KTA
15.12.1976	Kernkraftwerke Grafenrheinfeld, Grohnde und Brokdorf Stellungnahmen zu Qualitätssicherungsmaßnahmen bei der Fertigung der Sicherheitsbehälter
15.12.1976	Kernkraftwerk Kalkar (SNR-300) Sprengversuch am 1 : 6-Modell des SNR-300-Reaktortanks
10.11.1976	Kernkraftwerke Grafenrheinfeld, Grohnde und Brokdorf Stellungnahmen zu Qualitätssicherungsmaßnahmen bei der Fertigung der Sicherheitsbehälter
10.11.1976	Kernkraftwerk Mülheim-Kärlich Notwendigkeit einer Notkühlheißeinspeisung
13.10.1976	Kernkraftwerk Würgassen Befristeter Weiterbetrieb mit den derzeitigen Schnellabschaltbehältern
22./23.06.76	Regelvorhaben des KTA
22./23.06.76	RSK-Leitlinien Annahmen zur Spaltproduktfreisetzung nach Störfällen in Druckwasserreaktoren
28.04.1976	Werkstoffe für Komponenten der Druckführenden Umschließung und für Sicherheitsbehälter Verwendung höherfester Stähle für Sicherheitsbehälter
28.04.1976	Kernkraftwerk Brunsbüttel Wirksamkeit des Druckabbausystems
28.04.1976	Nukleares Containerschiff NCS-80 Sicherheitskonzept
28.04.1976	Kernkraftwerk Kalkar (SNR-300) Regelung und Steuerung des Reventingsystems
28.04.1976	Regelvorhaben des KTA
15.10.1975	Regelvorhaben des KTA
17.09.1975	USAEC-Bericht WASH 1400 (Rasmussen-Report) und Risikobeurteilung
19.03.1975	Schnellabschaltbehälter

19.03.1975	Kernkraftwerk Biblis A Frischdampfleitungsbruch, Abfahren der Anlage über Sicherheitsventile des Sekundärkreises und Pumpenschwungrad Stellungnahme zur Fristverlängerung
19.02.1975	Schnellabschaltbehälter Kernkraftwerk Würgassen Kernkraftwerke Brunsbüttel, Philippsburg 1 und Isar
19.02.1975	Regelvorhaben des KTA
22.01.1975	Brennstoffkreislauf Versuchsprogramm über das Verhalten von Plutonium bei mechanischen Einwirkungen
11.12.1974	Entwurf der Strahlenschutzverordnung (SSVO) vom 01.10.1974
11.12.1974	Zusammenstellung der im Genehmigungsverfahren für Kernkraftwerke zur Prüfung erforderlichen Unterlagen - Unterlagen über die Eigenschaften des Standorts
22.05.1974	Kernkraftwerke mit Siedewasserreaktoren
22.05.1974	Richtlinien für die Bemessung von Stahlbetonbauteilen von Kernkraftwerken für außergewöhnliche äußere Belastungen (Erdbeben, äußere Explosionen, Flugzeugabsturz) des Instituts für Bautechnik (Fassung Februar) 1974
22.05.1974	KTA-Regel: "Erdbebenauslegung von Kernkraftwerken, Blatt 1: Grundsätze"
22.05.1974	KTA Vorberichte über Regelvorhaben
24.04.1974	Mehrzweckforschungsreaktor (MZFR) Betrieb mit angereichertem Uran
24.04.1974	Gutachten von A. Cottrell zur Integrität von Reaktordruckbehältern
20.03.1974	Sicherheitskriterien für Kernkraftwerke
23.01.1974	Umwelteinfluß der Kernenergie
17.10.1973	Fachfragen der vom KTA empfohlenen Sicherheitskriterien
18.04.1973	Kernkraftwerk BASF Sicherheitskonzept im Hinblick auf die Besonderheiten des Standortes

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Implementation in Practice of Articles 8a-8c of Directive 2014/87/EURATOM Reactor Safety Upgrades in Romania

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Abstract

The Defence-in-Depth principle and the 10th year periodic safety review have been already integral part of the Romanian nuclear legislation before the implementation of the Council Directive 2014/87/Euratom. The legislation and rules are in place, the regulator has confirmed the plan of upgrades for Cernavoda NPP and the upgrades of the nuclear power plant have been commissioned.

The report outlines the Romanian legal framework for nuclear safety, and the legal and technical implementation of the safety improvements in nuclear in Romania, followed by some remarks on the possible good practices and challenges.

The report will be used as an input to the final report summarizing the national practices in the implementations of safety upgrades in operating nuclear reactors.

The report is part of the Task 3 of the project ENER/17/NUCL/S12.769200 contracted by the European Commission, DG ENER to ETSON, to support Member States in achieving a consistent practical implementation of the provisions set out in the Council Directive 2014/87/Euratom of 8 July 2014.

Contents

List of Acronyms

Acronym	Definition
AAGM	Alarming Area Gamma Monitors
ACR	Abnormal Condition Report
ADP	Auto Depressurization
AECL	Atomic Energy of Canada Limited
APOP	Abnormal Plant Operating Procedure
BDBA	Beyond Design Basis Accident
BLC	Boiler Level Control
BPC	Boiler Pressure Control
CANDU	CANada Deuterium Uranium
CDF	Core Damage Frequency
CGN	China General Nuclear
CNCAN	National Commission for Nuclear Activities Control
CNRA	Committee of Nuclear Regulatory Activities
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CSP	Critical Safety Parameter
DC	Direct Current
DEC	Design Extended Condition
DiD	Defence-in-Depth
DN	Scan system (location of failed fuel)
DR	Discipline Based Reports
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
ENSREG	European Nuclear Safety Regulators Group

EPDM	Terpolymer of ethylene, propylene and a diene with the residual unsaturated portion of the diene in the side chain
EPP	Emergency Protection Plans
EPS	Emergency Power System
EQ	Environmental Qualified
EU Clearinghouse	European Union operating experience feedback project
EWS	Emergency Water System
FSAR	Final Safety Analysis Report
FW	Feedwater
GSR	General Safety Standards (IAEA)
HELB	High Energy Line Break
HTC	Heat Transfer Control
IAEA	International Atomic Energy Agency
INES	International Nuclear and Radiological Event
INPO	Institute of Nuclear Power Operations
IRRS	Integrated Regulatory Review Service (IAEA)
IRS	International Reporting System (IAEA)
LBD	Licensing Basis Document
LERF	Large Early Release Frequency
LISS	Liquid Injection Shutdown System
LLOCA	Large Loss of Cooling Accident
LOCA	Loss of Cooling Accident
LOUHS	Loss of Ultimate Heat Sink
LP ECC	Low Pressure Emergency Core Cooling
MCR	Main Control Room
MIT	Mitigation Program
MP ECC	Medium Pressure Emergency Core Cooling
MSLB	Main Steam Line Break
MSSV	Main Steam Supply Valve

MTC	Mass Transfer Control
NEA/OECD	Nuclear Energy Agency of the Organisation for Economic Co-operation and Development
NLC	Nuclear Law Committee (NEA/OECD)
NPP	Nuclear Power Plant
OPEX	Operating Experience with NPPs
PAR	Passive Autocatalytic Recombiners
PGA	Peak Ground Acceleration
PHT	Primary Heat Transport
PHTS	Primary Heat Transport System
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Reviews
R&D	Research and Development
R/B	Reactor Building
RCW	Recirculated Cooling Water (system)
RRS	Reactor Regulating System
RSW	Raw Service Water (system)
SA	Severe Accident
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBO	Station Black-out
SCA	Secondary Control Area
SCADA	Supervisory Control and Data Acquisition system
SDCS	Shutdown Cooling System
SDS # 1	Shutdown System no1
SERG	System to prevent explosion hazard
SF	Safety Fundamentals (IAEA)
SFB	Spent Fuel Bay
SFP	Spent Fuel Pool

SG	Steam Generator
SNN	Nuclearelectrica SA - National Company Nuclearelectrica, the owner and operator of Cernavoda NPP
SOER	Significant Operating Experience Report (WANO)
SSC	System, Structure and Component
TR	Topic Reports
U1, U2	Unit no.1, no.2
US NRC	United States Nuclear Regulatory Commission
VDNS	Vienna Declaration on Nuclear Safety
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators' Association
WENRA / RHWG	WENRA Working Group on Reactor Harmonization
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 *Background information*

Romania has only one nuclear power plant, Cernavoda NPP, with two units in operation, which cover approximately 18% of Romania's total electrical energy production.

Cernavoda NPP uses CANDU reactor technology from Atomic Energy of Canada Limited (AECL), using heavy water as its neutron moderator and as its coolant agent.

The initial plan was to build on site five units.

For Unit 1 the construction began in July 1982, but it was commissioned only on 16th of April1996.

Unit 2 began construction in July 1983. A consortium of AECL and Ansaldo Nucleare of Italy, along with the Nuclearelectrica (SNN) SA, Romania's nuclear public utility, was contracted in 2003 to manage the construction of the partially completed Unit 2 power plant and to commission it into service. Four years later, Unit 2 was operated at full capacity (on 12 September 2007).

The construction of Units 3 and 4 started in the early 1980s but was stopped in 1992 when the Government decided to focus resources on the completion of Unit 1. There are still plans to further increase nuclear generating capacity through the commissioning of Units 3 and 4 of the Cernavoda NPP (pre-licensing reviews have been successfully completed, with no application for a construction license been submitted yet). In 2016 the Romanian government gave support for the creation of a joint venture led by China General Nuclear (CGN) to progress the project.

The construction of Unit 5 has been cancelled by a decision of the General Shareholder Assembly of the National Company Nuclearelectrica, the owner and operator of Cernavoda NPP, with the intent to use the existing structures of Unit 5 for different activities connected to the operation of Units 1 and 2 and, in the future, of Units 3 and 4.

The situation is resumed in table 1.

Table 1. Cernavoda nuclear units

Reactor	**Type**	**Gross Capacity MW(e)**	**Construction Start**	**First Criticality**	**Operating Status**
Cernavoda-1	CANDU-6	706.5	1980	16th of April 1996	in operation

Cemavoda-2	CANDU-6	706.5	1980	6th of May 2007	in operation
Cemavoda-3	CANDU-6	720	1980	-	under preservation, plans for resuming construction
Cemavoda-4	CANDU-6	720	1980	-	under preservation, plans for resuming construction
Cemavoda-5	CANDU-6	-	1980	-	no plans for resuming construction; the existing structures will be used for supporting activities of the other units.

The National Commission for Nuclear Activities Control (CNCAN) (1) is the nuclear safety and security regulatory authority of Romania, responsible for the regulation, licensing and control of nuclear activities, ensuring the peaceful use of nuclear energy and the protection of public and workers from the harmful effects of ionising radiation. CNCAN elaborates the strategy and the policies for regulation, licensing and control with regard to nuclear safety, radiological safety, non-proliferation of nuclear weapons, physical protection of nuclear installations and materials, transport of radioactive materials and safe management of radioactive waste and spent fuel, as part of the national strategy for the development of the nuclear sector. CNCAN reports to the Prime Minister, through the General Secretary of the Government.

Following the Fukushima Daiichi accident, CNCAN has focused initially on the technical reviews of the protection of the plant against extreme external events and of beyond design basis accident analysis, severe accident management and emergency response. After more information became available on the organizational factors that have contributed to the accident, CNCAN has used the lessons learned to improve the national regulatory framework, its practices for regulatory oversight of licensee's safety culture and its own safety culture. (2)

New regulations and regulatory guides have been issued or are in process of being issued and work is on-going for the implementation of the National Strategy for Nuclear Safety and Security. (3)

1.2 *Romanian regulatory framework*

The Romanian legislative and regulatory framework is in place to govern the safety of nuclear installations, providing:

- the establishment of applicable national safety requirements and regulations;
- a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a license;
- a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses;
- the enforcement of applicable regulations and of the terms of licenses, including suspension, modification or revocation.

The Law no. 111/1996 (4) on the safe deployment, regulation, licensing and control of nuclear activities, republished in the Official Gazette no. 552/27.06.2006, provides the legislative framework governing the safety of nuclear installations and empowers CNCAN to issue mandatory regulations, to issue licenses for nuclear installations and activities, to perform assessments and inspections to verify compliance with the nuclear safety requirements and to take any necessary enforcement actions. The Law also clearly states the responsibilities of the licence holders.

All the regulations issued by CNCAN are mandatory and enforceable. (1) The regulations are developed in concordance with relevant international standards and good practices.

In accordance with the provisions of the Law, CNCAN has the responsibility for reviewing the regulations whenever it is necessary for these to be consistent with international standards and with relevant international legislation in the domain, and for establishing the measures for the application thereof.

Various sources of information relevant for updating the system of regulations and guides are used, including the new developed international legislation and safety standards, international cooperation, feedback from the industry and feedback from CNCAN inspectors based on their experience with the enforcement of the regulations, the results of research and development activities.

Besides the needs arisen from the licensing process, priorities for development of regulations were established as part of the harmonization process in the WENRA (Western European Nuclear Regulators' Association) countries.

1.3 *National Strategy for Nuclear Safety and Security*

The work on a National Strategy for Nuclear Safety and Security was initiated by a recommendation received from an Integrated Regulatory Review Service (IRRS) Mission (January 2011), being recognized the fact that most of the elements required by such a strategy were already in place. It was considered that a National

Strategy may bring better coordination and coherence in addressing all the aspects and measures that have an impact on nuclear safety and security, and it was recommended that "*the national policy and the strategy for safety should be implemented in accordance with a graded approach*".

The development of the national strategy started in 2013. At first, the strategy addressed only nuclear safety (including radiological protection and emergency preparedness and response), but the scope of the strategy was later expanded to cover also nuclear security (including physical protection, nuclear safeguards and cyber security). Based on the current regulatory framework and on the trends observed at international level with regard to the improvement of the synergy between safety and security, it was decided that a national strategy addressing both nuclear safety and security is justified.

The strategy includes a policy statement with nuclear safety and security principles, including the ten fundamental safety principles outlined in the IAEA SF-1, and takes account of the relevant provisions of the IAEA GSR Part 1 document.

In July 2014, the National Strategy for Nuclear Safety and Security (3) was officially approved by the Romanian Government and by the Supreme Council of National Defence, has been published and has come into force. The implementation of the strategy and of its corresponding action plan is monitored, and is planned that the strategy will be reviewed and revised as necessary.

1.4 *International obligations and arrangements for cooperation*

Romania is a contracting party to all relevant international conventions that establish common obligations and mechanisms for ensuring protection and safety: (5)

- Convention on Nuclear Safety,
- Convention on Early Notification of a Nuclear Accident,
- Convention on Assistance in Case of a Nuclear Accident or Radiological Emergency,
- Vienna Convention on Civil Liability for Nuclear Damage, as amended in 1997,
- Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention,
- Convention on Physical Protection of Nuclear Material as amended in 2005
- Convention on Supplementary Compensation for Nuclear Damage,
- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management,
- Treaty on Non-Proliferation of Nuclear Weapons,

- International Convention for the Suppression of Acts of Nuclear Terrorism.

Romania made political commitments to the Codes of Conduct that promote the adoption of good practices in the relevant facilities and activities, namely:

o Code of Conduct for the Safety and Security of the Radioactive Sources and supplementary Guidance;

o Code of Conduct on the Safety of Research Reactors.

Romania is a member of the Western European Nuclear Regulators Association (WENRA) (6) since 2003. WENRA membership is useful in developing a common approach to nuclear safety, in providing an independent capability to examine nuclear safety in applicant countries and in participating in a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues. (6) Romania is also member of European Nuclear Safety Regulators Group (ENSREG) (7) which was established in 2007. CNCAN participates in ENSREG to support establishment of the conditions for continuous improvement and to reach a common understanding in the areas of nuclear safety and radioactive waste management.

Through commitments under international conventions and bilateral arrangements, CNCAN maintains relations with a number of nuclear regulatory authorities and organisations worldwide, and also takes an active part in the international peer reviews of the regulatory control and safety of facilities.

There is a continuous effort of CNCAN to harmonize its practices with those already implemented by international community. The Law no. 111/1996 on the safe deployment, regulation, licensing and control of nuclear activities (4) empowers CNCAN to issue mandatory regulations on nuclear safety, and clearly establishes an obligation to harmonize these regulations with IAEA safety standards. Specifically, it is required to “review the regulations whenever it is necessary for these to be consistent with international standards and with ratified international conventions”.

Based on the review of documents and discussions with counterparts, an IRRS Team concluded that the Romanian government effectively fulfils its international obligations, participates in the relevant international arrangements, including international peer reviews, and promotes international cooperation to enhance safety globally. (5)

As a part of bi-lateral agreements with regulatory bodies from other countries, CNCAN made several arrangements for the exchange of information relevant to nuclear safety, including:

- Participation in the annual meetings of the CANDU Senior Regulators Group,
- Participation as a member of WENRA technical working groups,
- Participation in the biannual meetings of the European High-Level Group on Nuclear Safety and Waste Management (ENSREG) and its working groups,

- Participation in the annual session of the Nuclear Law Committee (NLC) of the NEA/OECD (Nuclear Energy Agency of the Organisation for Economic Co-operation and Development),
- Participation in the Committee of Nuclear Regulatory Activities (CNRA) and Committee on the Safety of Nuclear Installations (CSNI) of NEA/OECD.

Furthermore, CNCAN participates in the exchange of regulatory and operating experience through the IAEA International Reporting System (IRS) (8) and through the EU Clearinghouse (9) (European Union operating experience feedback which is closely linked to IRS).

2 SAFETY UPGRADES

The safety upgrades are implemented on a continuous basis to maintain safety margins and incrementally enhanced safety at the site.

The main developments are represented by the improvement actions implemented as a result of the safety reviews performed after the Fukushima Daiichi accident. In addition, other important developments are related to design improvements based on insights from probabilistic safety assessments and operational experience feedback. (2) Periodic safety reviews are considered robust and reliable mechanisms for safely operating the Cernavoda NPP, including comprehensive and systematic plant condition assessments and the identification of safety improvements that are reflected in integrated implementation plans.

The implementation of plant safety upgrades is driven by two approaches:

- *Bottom-up*: the conceptual proposals are prepared by the licensee, sent for revision to CNCAN, reviewed, possibly modified, and after are accepted by the regulatory body, implemented by the licensee.
- *Top-down*: the changes in legislation usually trigger the requirement of a specific modification/ action of the licensee.

Each proposed technical solution requires the safety evaluation of its impact on the plant.

The implementation of the Council Directive 2014/87/Euratom (10) in the national legislation has been performed using the top-down approach.

The bottom-up approach with plant upgrades proposed by the licensee has been the major driver for the implementation of the post Fukushima plant upgrades.

The technical implementation of the Cernavoda NPP safety upgrades, associated with the European Stress tests and the Council Directive 2014/87/Euratom (10), is summarized in the 2017 Update of the Romanian Post-Fukushima Action Plan (11), (12).

The action plan was developed before the WENRA definition of DEC in 2014 (13) or the Council Directive 2014/87/Euratom (10), taking into account the context of many other drivers for safety upgrades, including:

- International organizations recommendations;
- Periodic Safety Review;
- Post-Fukushima stress tests (14), (15), (16).

2.1 *External operating experience and international recommendations*

The international nuclear organizations require a prompt notification regarding events occurred at the plant in order to offer well-timed information to the world community.

Information obtained from the internal and external operational experience is used for multiple purposes, such as:

- Improving the operating practices and plant staff training programs;
- Improving the plant design;
- Input for Ageing Management Program;
- Assessment of necessity for updating the safety analyses (deterministic and probabilistic), etc.

The external information on operating experience proved to be a very important tool in improving the performance of Cernavoda NPP. Therefore, the second main topic of the Operating Experience Program is the Information Exchange Program, with bi-directional use:

- collecting of external information and distribution to the appropriate plant personnel;
- submitting the internal operating experience information to external organizations.

The Cernavoda plant procedure “External Operating Experience Feedback” is in place for screening for applicability the information provided by external organizations like COG (17), WANO (18), INPO (19) and IRS (8).

For any applicable external event identified an external ACR is issued and recorded in OPEX database. External ACRs are analyzed using a specific template.

For the major events (e.g. WANO Significant Operating Experience Reports/Significant Event Reports, IRS events level 2 or higher on INES scale), an Abnormal Condition Report is issued, and the analysis is performed using a gap analysis template.

This means that the actual processes, procedures and work practices are compared with the recommendations given in the reports, a gap is identified (if exists) between current situation and recommended aspects, and actions are defined to fill in the gap. Further processing is performed according to plant instruction “Abnormal Conditions Reports”.

For example, it was a development and implementation of the corrective action plan by the licensee in response to the event reported by US NRC (7950/13.11.2008) - the licensee took the initiative to analyze the event, to identify the lessons to be learned and to develop a corrective action plan. The implementation of the corrective action plan was verified by CNCAN site inspectors.

Except this formal processing and tracking of significant industry events, plant personnel has access to the COG Operating Experience Database and to WANO/INPO websites and operating experience posts and monitors daily the new events posted on these websites. The majority of the records is posted only for information, but might be used while reviewing in-house events, design modifications or looking for relevant just-in-time operating experience for certain operational evolutions or other activities.

Romania has committed to fulfilling the 2015 Vienna Declaration on Nuclear Safety (VDNS) (20), which provides principles for implementing the Convention's objective: to prevent accidents and mitigate radiological consequences).

Principle 2 of the VDNS requires comprehensive and systematic safety assessments to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the objective of the VDNS. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.

Cernavoda NPP is member of WANO (18), an organization dedicated to help its members to achieve the highest levels of operational safety and performance. WANO conducts periodic evaluations to promote excellence in the operation, maintenance and support of operating NPPs, with a focus on safety and reliability. These evaluations are not required by law or regulation but are requested on a voluntary basis by WANO members.

In addition to membership in WANO, Cernavoda NPP is member of the CANDU Owners Group (COG) (17), a not-for-profit organization dedicated to providing programs for cooperation, mutual assistance and exchange of information for the successful support, development, operation, maintenance and economics of CANDU technology. COG has provided the mechanism for many projects to improve the safety of CANDU reactors. The purpose of this collective effort is to pool understanding and expertise (when appropriate), coordinate and prioritize the resolution of issues and improvement initiatives and enhance overall adherence to regulatory requirements.

In addition to its R&D program, COG facilitates the execution of licensee responsibility by: (17)

- sharing operating experience and providing support to resolve technical and operating problems for all COG members
- initiating and managing jointly funded projects and services
- adopting common strategies and plans for the resolution of regulatory issues related to nuclear safety
- sharing best practices, delivering jointly developed training programs and developing knowledge-retention tools such as the CANDU textbook

The safety improvements which, in part are a result of lessons learned from the Fukushima accident, are aimed primarily at Defence-in-Depth (DiD) Levels 3, 4 and 5, and further reduce the risk of events that may lead to a large off-site release of radioactivity.

The specific improvements recommended by COG include the following areas: (21)

- Provision of additional make-up water for fuel cooling;
- Pressure relief for heat sinks such as the shield tank and for the calandria vault;
- Additional hydrogen mitigation;
- Portable electrical power supplies for key equipment and instrumentation and control;
- Updated Severe Accident Management Guidance which incorporates multi-unit effects and effective use of Emergency Mitigating Equipment, including for irradiated fuel bays;
- Incorporation of severe accident and multi-unit scenarios in emergency planning and exercises.

On the same direction, some generic recommendations compiled by ENSREG (7) were implemented using the improvement actions from National Action Plan, being related to: (11)

Containment integrity

The objective was the enhancement of the containment integrity.

Seismic Monitoring

The recommendation regarded the installation of seismic monitoring systems with related procedures and training. Based on the seismic margin assessment performed for Cernavoda NPP, all SSCs involved in the safe shutdown after an earthquake, will continue to perform their safety function up to a PGA corresponding to an earthquake with an estimated frequency of 5E-5 events/year.

Alternate Cooling and Heat Sink

The recommendation was related to provision of alternative means of cooling including alternate heat sinks. Examples include steam generator (SG) gravity alternative feeding, alternate tanks or wells on the site, air-cooled cooling towers or water sources in the vicinity (reservoirs, lakes) as an additional way of enabling core cooling.

AC Power Supplies

The objective was the enhancement of the on-site and off-site power supplies. Examples include adding layers of emergency power, adding independent and dedicated back-up sources, the enhancement of the grid through agreements with the grid operator on rapid restoration of off-site power, additional and/or reinforced off-site power connections, arrangements for black start of co-located or nearby gas or hydro plants, replacing standard ceramic based items with plastic or other material that are more resistant to a seismic event. Another example is the possible utilization of generator load shedding and house load operation for increased robustness, however, before introducing such arrangements the risks need to be properly understood.

DC Power Supplies

The objective was the enhancement of the DC power supply. Examples include improving the battery discharge time by upgrading the existing battery, changing/diversifying battery type (increasing resistance to common-mode failures), providing spare/replacement batteries, implementing well-prepared load shedding/ staggering strategies, performing real load testing and on-line monitoring of the status of the batteries and preparing dedicated recharging options.

Instrumentation and Monitoring

The objective was the enhancement of instrumentation and monitoring. Examples include separate instrumentation and/or power sources to enable monitoring of essential parameters under any circumstances for accident management and the ability to measure specific important parameters based on passive and simple principles.

Main and Emergency Control Rooms

The recommendation regarded the enhancement of the main control room (MCR), the emergency control room (ECR) and emergency control center to ensure continued operability and adequate habitability conditions in the event of a station black-out (SBO) and in the event of the loss of DC.

Spent Fuel Pool

The objective was the improvement of the robustness of the spent fuel pool (SFP). Examples include reassessment/upgrading SFP structural integrity, installation of qualified and power-independent monitoring, provisions for redundant and diverse sources of additional coolant resistant to external hazards (with procedures and drills), design of pools that prevents drainage, the use of racks made of borated steel to enable cooling with fresh (un-borated) water without having to worry about possible re-criticality, redundant and independent SFP cooling systems, provision for additional heat exchangers (e. g. submerged in the SFP), an external connection for refilling of the SFP (to reduce the need for an approach linked to high doses in the event of the water falling to a very low level) and the possibility of venting steam in a case of boiling in the SFP

Mobile Devices

The recommendation was related to provision of mobile pumps, power supplies and air compressors with prepared quick connections, procedures, and staff training with drills. Mobile devices are intended to enable the use of existing safety equipment, enable direct feeding of the primary or secondary side, allow extended use of instrumentation and operation of controls, allow effective fire-fighting, and ensure continued emergency lighting. The equipment should be stored in locations that are safe and secure even in the event of general devastation caused by events significantly beyond the design basis.

Bunkered/Hardened Systems

The recommendation regarded the provision for a bunkered or hardened system to provide an additional level of protection with trained staff and procedures designed to cope with a wide variety of extreme events including those beyond the design basis

SAM Hardware Provisions

The recommendation was related to adequate hardware provisions that will survive external hazards (e.g. by means of qualification against extreme external hazards, storage in a safe location) and the severe accident environment (e.g. engineering substantiation and/or qualification against high pressures, temperatures, radiation levels, etc.), in place, to perform the selected strategies.

Enhancement of Severe Accident Management Guidelines (SAMG)

The objective was the enhancement of SAMGs taking into account additional scenarios, including a significantly damaged infrastructure, including the disruption of plant level, corporate-level and national-level communication, long-duration accidents (several days) and accidents affecting multiple units and nearby industrial facilities at the same time.

2.2 *Periodic Safety Review*

In the past, the Romanian licensing system required a safety review to be carried every two years, in order to support the license renewal. The main safety issues addressed, with the Safety Analysis Report as the main document under review, corresponded largely to the 14 safety factors proposed by IAEA's Safety Guide NS-G-2.10 (22). The scope of the Periodic Safety Reviews (PSR) in the general understanding being more comprehensive, the benefit of carrying such reviews was recognised and this has led to a change in the Romanian licensing approach. (23)

In 2006, following a recommendation received from an IRRS Mission organised by IAEA and also as a result of the participation in the study "Harmonisation of Reactor Safety in WENRA Countries", CNCAN issued a regulation on Periodic Safety Review of Nuclear Power Plants (24), as a first step towards the changing of the licensing system. The regulation requires a PSR to be conducted every ten years. The Romanian regulation is based on the Safety Guide NS-G-2.10 (22), having the 14 "safety factors" defined as "areas of review", for each of these having specified most of the "generic review elements" given in the Appendix to the IAEA guide.

In 2005, Cernavoda NPP started for Unit 1 the preparation phase of PSR. The main objectives of the Cernavoda Unit 1 PSR were the following:

- To undertake a systematic review of the current plant design and safety analysis against internationally accepted safety standards, codes and practices.
- To review ageing mechanisms and their management, in order to confirm that the plant is safe to operate for at least the next 10-year period, subject to continuing routine maintenance, testing, monitoring and inspection.
- To review the operating history of the plant, and plants of similar design, to identify and evaluate factors that could limit safe operation during the next 10-year period.
- To identify PSR Findings of safety significance, and determine those safety enhancements which are reasonably practicable, and that should be implemented as Corrective Actions to resolve the issues that have been identified.

Main PSR Activities were undertaken between May 1, 2008 and June 30, 2012. As a first step, a preliminary analysis has been done and documented in 6 Discipline Based Reports (DRs) and 39 Topic Reports (TRs) that constitute the primary low-level documentation. The consolidated, systematic safety review has been further completed as the next step. The PSR results were documented in 4 Information Reports (Summary; Operational and Safety Performance; Systems, Structures and Components; Safety Analysis).

In Phase 3 of PSR **(**Corrective Action Plan & Implementation)**, f**or each of the findings identified in PSR, an analysis has been performed, that includes the final assessment of the safety impact of the finding, and the final proposal of the corrective actions. Corrective actions and improvements have been identified and used for the development of the PSR Corrective Action Plan (36 as results of PSR, 4 proposed by CNCAN) together with the proposed target dates for implementation, and were submitted to CNCAN for approval.

PSR for Unit1 was performed between Oct2008 and June2012. Further, Cernavoda NPP has performed an analysis of the applicability of Unit 1 PSR results on Unit 2, submitted to CNCAN as licensing support document.

The weaknesses identified through PSR, the corrective actions, the target dates proposed for implementation and the analysis of the impact of Unit 1 PSR results on Unit 2 were considered adequate by CNCAN.

After PSR completion, the main licensing document - Unit 1 Final Safety Analysis Report (FSAR) - has been updated and submitted to CNCAN, in order to support the Operating License Application.

Based on PSR results for Unit 1 and Unit 2 licence renewal process (Unit 2 is in operation since November 2007), in 2013 CNCAN has granted new licence for Unit 1 for 10 years and only for 7 years for Unit 2, so that Unit 2 completes its own full-scope PSR. (23)

In the next period, Cernavoda NPP is preparing for conducting the second PSR evaluation of Unit1 and the first PSR for Unit2. Both PSR evaluations will be based on CNCAN NSN10 and IAEA SSG-25.

2.3 *Design changes*

Any improvements in the existing design or redesign of the systems or components are subject to the same verification as the original design in order to confirm that all the existing analyses are valid and the design is correct.

The key objective of the CNCAN inspection program for Cernavoda NPP is to monitor compliance with the legal, regulatory and licensing requirements, and to take enforcement action in the event of non-compliance. A set of internal procedures establishing the administrative rules for conducting review and assessment activities of the facilities, activities and practices related to the safety is issued and they are included in the CNCAN quality management system manual.

The inspections for Cernavoda NPP are planned in a systematic manner by the staff from CNCAN headquarters and the resident inspectors, with the aim of ensuring a proactive identification of the deficiencies and deviations from good practices that could result in non-compliances.

The inspection planning for Cernavoda NPP is periodically reviewed and modified as new information on the facility or organization is obtained. The inspections are normally focused on those areas that would pose a significant risk, or for which a poor performance has been recorded.

Since the early stages of the development of the Romanian Nuclear Program, the contractual arrangements between the licence holder and the designer/vendor have been focused on ensuring that sufficient design information is provided to ensure the safe operation and maintenance of the plant and to support the development of national competence and expertise with regard to CANDU design.

Arrangements are in place also to obtain technical advice and support with regard to any safety related issues for which external expertise would be needed, as Cernavoda NPP maintains a close relation with the plant designer and vendor (Atomic Energy of Canada Limited - AECL) and with the other CANDU operators worldwide (through the CANDU Owners Group - COG).

Cernavoda NPP, Unit 1

Unit 1 of Cernavoda NPP was commissioned in 1996. The design installed and commissioned in Romania has incorporated most of the significant safety related design changes already made by other organisations operating CANDU-6 up to late 80's. Supplementary, during commissioning a few other hundreds of design changes were incorporated that originated from:

 CANDU 600 operating experience, especially Point Lepreau, Gentilly 2 and Wolsung;

 safety assessments performed in Canada following the occurrence of some incidents at other nuclear power plants;

 the probabilistic safety evaluations performed to verify the adequacy of design.

Some examples of modifications incorporated in the "as-commissioned" Cernavoda NPP Unit 1 are given below: (2)

- modification of the control room design to consider human error factors;
- new material used for the pressure tubes (Zr-2.5%Nb);
- improved trip coverage;
- automation of the low power conditioning for the trip of shutdown systems on low pressurizer level and low boiler level;
- improvements to increase ECCS reliability;
- provisions for the post LOCA collection of leakage from ECC pumps;
- provision of redundant back-up cooling for RSW system;

- improvements of instrument air reliability;
- improvements of the containment liner to minimise the leak rate;
- provisions for annulus gas recirculation;
- provision for a facility for post LOCA sampling of Containment Atmosphere;
- improvements of the fire protection, etc.

Examples of design changes implemented after the start of operation: (2)

- Removal of ADP functions from BLC program to an independent program - MIT (Mitigation Program) in order to avoid the failure of the ADP function at BLC program failure (clear separation between the safety function and process function);
- Modification of the start-up system to ensure complete independence of the redundant Diesel generators of the EPS;
- As a result of the thermo-hydraulic analyses for review of LPECC flow capacity in case of LOCA event, a design modification for replacement of the two 100% capacity strainers for Cernavoda Unit1 has been implemented in 2002, in order to prevent sump filter clogging in case of LLOCA and to ensure the required performance of the pump under the design basis operating conditions for a minimum mission period of three months;
- Replacement of Chiller Units;
- Replacement of the LISS injection valves;
- A new portable vacuum subsystem has been installed to clean the underwater surface of the spent fuel bay;
- The silicon rubber seals of the airlocks have been replaced with EPDM perimeter seals, that have better design parameters and are EQ qualified;
- The original strainer located on the suction line of the EWS pumps was replaced by a new strainer made by stainless steel and corrosion resistant.

The process for initiating, assessing and implementing design changes is defined by a set of plant procedures, with the aim of ensuring effective configuration control and conformance with the design basis of the plant.

Cernavoda NPP has a feed-back program to implement the design modifications and improvements from Unit 2 to Unit 1 that ensure safety enhancement and that are reasonably practicable for Unit 1, in order to maintain an equivalent level of nuclear safety with Unit 2. Some of the design changes considered in the LBD for Unit 2 have already been implemented also in Unit 1, e.g.: (2)

- lowering of the calandria outlet temperature to increase moderator subcooling, and consequently, improved moderator system capacity to act as a heat sink;

- PHT Liquid Relief Valves and Degasser Condenser relief valves modification that increase the PHT system overpressure protection;

- changes that minimise the positive reactivity at the reactor in the event of failure of the Liquid Zone Control pumps;

- improved valve in Feedwater System to allow the auxiliary feedwater pump operation with depressurized boilers, in case of MSLB;

- manual actuation of SDS # 1 from SCA - a seismic qualified area;

- actuation of ECC System on a new parameter (sustained low pressure on PHTS);

- automatic transfer from ECCS Medium-Pressure Injection phase to the Low-Pressure Injection phase.

The assessment of the reasonable practicability of the above-mentioned changes has been completed in the framework of the first PSR of Cernavoda Unit 1. Also, recommendations for Unit 1 design changes resulted from the Unit 1 PSR, based on the comparison with the Unit 2 newer project, which refer to:

- manual actuation of SDS # 1 from a seismic qualified area, such as SCA;

- the environmental qualification up-grade of some Unit 1 system components.

The planned refurbishment of Cernavoda Unit 1 will provide an opportunity to upgrade the existing project, by implementing additional reasonable and practical modifications to enhance the safety of the facility to a higher level. Integrated implementation plans will identify strengths and shortcomings for each of the safety factors identified in the PSR, rank the shortcomings in terms of safety significance, and prioritize corrective measures, including design and other safety improvements.

Cernavoda NPP, Unit 2

The work on Unit 2 restarted in 2001. The engineering documentation for Unit 1 was updated to be used as reference for Unit 2 and the existing facilities and buildings were re-certified.

In the period for which the construction of Unit 2 was stopped, there have been many developments in the nuclear industry worldwide (CANDU plants similar to Cernavoda have been built and placed in service in South Korea -3 units at Wolsung and in China -2 units at Qinshan). In addition, during this period, additional experience has been gained from the operation of CANDU plants worldwide.

All the improvements resulting from the commissioning and operating experience were considered in the process of identification of the feasible design changes for Unit 2, account being taken of the stage of the construction work. After thorough review, the changes selected for implementation on Cernavoda Unit 2 (156 design changes) can be categorised as follows: (23)

- Design changes to meet revised licensing requirements, in response to revision of codes, standards or regulatory requirement documents. Since the original design of Unit 1 was completed, some of the codes, standards and regulatory licensing requirements have been revised to improve consistency and to increase the margin of safety. In general, these changes can be categorised as safety improvements.
- Changes due to development of CANDU technology. In general, these changes result in improved performance or reliability of operation.
- Design changes to replace equipment where the equipment used in Unit 1 is approaching obsolescence, and modernisation will result in improved availability of spare parts and maintenance.
- Other design improvements for enhancing system or plant performance.

Examples of safety improvements are given below: (2)

- Provision of a second independent steam generator crash cooldown system, to improve reliability of the secondary circuit as a heat sink for the intact loop in case of LOCA and for the failed loop for small breaks;
- Improved EWS reliability (protection against single failures);
- Automation of start-up of LP ECC to eliminate the need for operator action to manually switch from MP to LP ECC operation 15 minutes after a LOCA;
- Provision for redundant flow paths for ECC pump suction from dousing tank and redundant dousing tank level instrumentation;
- Provision of an on-power gross containment leakage monitoring system, to give additional assurance of containment boundary integrity for the periods between the full-scale leak rate tests;
- Provision of Hydrogen igniters to prevent Hydrogen accumulation in the Reactor Building in case of LOCA;
- Increased chromium content of lower outlet feeders, to ensure better protection against flow-induced corrosion and erosion;
- Post-Accident Monitoring System;
- Modification to ensure Environmental qualification for all systems' components required to manage and mitigate consequences in Reactor Building after steam line or heat transport pipe break (LOCA).

Since the approval of the LBD, there were more than 200 additional changes implemented in Unit 2. All the design changes were implemented through a rigorous Design Changes process that required the approval of the designer for all the special safety systems. All design changes were assessed for impact on plant safety and when it was the case (for the modifications classified as major) they were also submitted to CNCAN for review and approval.

Examples of design changes implemented after the start of operation of Unit 2:

- The original strainer located on the suction line of the EWS pumps was replaced by a new strainer made by stainless steel and corrosion resistant;
- The Alarming Area Gamma Monitors (AAGM) have been upgraded by replacing the silicon detectors with ion chamber detectors and also, a new gamma detection loop has been installed in Service Building, near ECC pumps;
- A connection bridge was built between Unit 1 and Unit 2 service buildings in order to ensure a better operation of both units.

Cernavoda NPP, Units 3 and 4

When construction works on Units 3&4 were halted (1992), the civil buildings and structures, including the reactor building, the service building and the turbine-generator building were significantly developed. The existing civil structures have been assessed against the requirements of the latest codes and standards and improvements will be implemented as far as reasonably practicable.

The Reference Plant for Cernavoda Units 3 and 4 will be the as-built Cernavoda 2 plant, and will include the changes required to meet the latest Codes and Standards, any licensing mandated changes, design modifications to deal with obsolete equipment and address operational experience feedback from other CANDU plants identified before the project start.

The preliminary list of design changes has been derived from the following sources: (2)

- The Deloitte feasibility study produced for Cernavoda Unit 3 to identify potential design changes;
- Canadian Nuclear Safety Commission (CNSC) generic action items;
- Identification of design changes resulting from Cernavoda Units 1 and 2 and other CANDU 6 operating experience (OPEX) available from AECL's feedback monitoring system;

- Identification of design modifications resulting from new editions of codes and standards;
- Identification of design changes not implemented on Cernavoda Unit 2 due to the advanced state of construction and which result from known issues such as generic action items;
- Identification of design changes resulting from the Cernavoda Units 1 and 2 probabilistic safety assessments;
- Identification of potential design changes resulting from the review of WENRA reactor safety reference levels and CNSC RD-337;
- Identification of design changes resulted from Fukushima lessons learned.

The design changes currently under consideration aim to ensure that the design is in line with the current requirements for new NPPs. The recommended targets for CDF and LRF for new reactors (CDF<10-5, LRF<10-6) are also a target for the design of the Units 3 and 4.

Latest improvements

Since 2010, significant progresses have been made in improving the Cernavoda NPP project, to achieve safe and reliable plant operation. Examples of design changes implemented are listed below: (23)

- As recommended based on U1 Level 1 PSA results, feedwater transmitters were environmental qualified to support automatic depressurization of the steam generators in the event of a HELB in the turbine building;

- Improvement of the performance of the Reactor Regulating System by reduction of the amplitude variances of the reactor power error (implemented at both units);

- Reduction of SDS # 2 Log Rate High Trip Setpoint, based on the “Loss of Regulation – Loss of Reactivity Control” safety analyses (implemented at both units);

- In order to monitor in real time the U2 process systems, a Supervisory Control and Data Acquisition (SCADA) system has been installed;

- Replacement at U1 of the coil encapsulated in-core flux detectors with the new HESIR type detectors, used at Unit 2;

- Replacement of some U1 obsolete snubbers installed on seismic qualified systems (e.g. ECCS, EWS, SDCS, PHT system and auxiliaries, etc);

- Extension of the fire suppression system to level 107.5 in Turbine Building (implemented at both units);

- Based on safety assessment and thermo-hydraulic analysis, the Moderator Temperature Control setpoint has been reduced, to increase moderator sub-cooling and to improve moderator system heat sink capacity (implemented at both units);

- In order to ensure the operation of RSW system with one pump even in case of Danube level is lower than minimal submergence of the RSW pumps, one RSW pump at each unit has been modified. Thus minimal submergence of the modified pump is decreased to ensure the cooling water for the essential safety systems in case of both units shutdown due to very low Danube level;

- Unit 1 control programs (RRS, HTC, MTC, BLC and BPC) have been modified, to decrease the occurrence probability of the “dual computer failure” event (implemented at both units).

Since 2013, to further improve the Cernavoda NPP project, and to achieve safer and more reliable plant operation, several design changes were implemented: (2)

-Following the recommendations of WANO GAP SOER 11-1 “Large Power Transformers Reliability”, the station large power transformers have been equipped with an on-line dissolved gases detection and alarm system;

- Implementation, at large power transformers, of a system to prevent explosion hazard (SERGI);

- Design modification to increase redundancy of the cooling circuit for the Auxiliary FW pump, to be used during plant outages;

- Change of the EPS Diesels starting system in order to ensure two independent groups, each one composed by a rectifier, a battery and a starter;

- Modification to improve reliability of the cooling system for U1 Instrument Air compressors;

- In case of a non-seismic induced SBO event a design modification has been implemented in order to ensure a water make-up path from fire water hydrant to the Calandria Vault;

- Considering the OPEX from an external event “Step-back with Control Absorber (CA) Rod 2 or 3 stuck”, a design modification of Mechanical Control Absorber Logic within RRS control program was implemented;

- The controls of the DN Scan system (location of failed fuel) were refurbished. This design modification replaces the old control system with a modern system;

- The redundancy of electrical power supply Reactor Building air dryers (D2O recovery system) was increased by providing a second source for each drier;

- The 110 kV Station has been refurbished to increase its reliability;

- Replacement of the obsolete temperature control loops for the spent fuel bays cooling; the new loops are equipped with programmable digital controllers for better temperature control;

- Refurbish the Cathodic protection for the U1-EPS underground fuel tanks;

- Increase the reliability of the U1 main electrical generator by replacing the Excitation System with a new generation one;

- manual actuation of SDS # 1 from SCA - a seismic qualified area;

- Improve reliability of U1 Standby Diesel Generators by installation of an air-drying system on the starting air system;

- actuation of ECC System on a new parameter (sustained low pressure on PHTS);

- automatic transfer from ECCS Medium-Pressure Injection phase to the Low-Pressure Injection phase.

2.4 *Safety upgrades following the Fukushima Daiichi accident*

After the Fukushima Daiichi accident, a complex safety review of the design was undertaken in the context of the European "stress tests". Before the requirements for the stress tests were issued, the licensee, National Company Nuclearelectrica, owner and operator of Cernavoda NPP, had already initiated measures in response to the Significant Operating Experience Report issued by the World Association of Nuclear Operators (WANO) SOER 2011-02 (Fukushima Daiichi Nuclear Station Fuel Damage Caused by Earthquake and Tsunami), including a thorough plant walkdown for verifying protection against seismic, fire and flooding events; acquisition and testing of mobile Diesel generators; development of new operating procedures for response to SBO and to total and extended Loss of Spent Fuel Pool Cooling events.

The level of DiD of Cernavoda NPP was assessed in light of the Fukushima accident, and it was concluded that the NPP met the design requirements. It was also concluded that the risk to the public from BDBAs at NPP was very low. Given the design features and DiD for NPP, adequate time would be available for long-term mitigation of a BDBA. Besides the consideration of specific hazards, the licensee has systematically verified, and supplemented where appropriate, the effectiveness of the existing NPP capabilities in BDBA and severe accident conditions. In terms of design aspects relevant to lessons learned from the Fukushima accident, the design of Cernavoda NPP include several features that prevent accidents and can help mitigate impacts should an accident occur. (16)

The seismic walkdowns and subsequent seismic robustness analyses done as part of the seismic margin assessment have not revealed a need for any safety significant design change. However, several recommendations resulted from these inspections and have been included in the regular plant seismic housekeeping program.

Several design improvements have been identified and have been implemented to maintain fuel cooling during severe accident conditions and to enhance the capability to maintain containment integrity in case of severe accidents. These include the provision of water make-up to calandria vessel and calandria vault to arrest the progression and relocation of the core melt, the provision of hydrogen monitoring systems and passive autocatalytic recombiners for hydrogen management and the installation of filtered containment venting systems.

Several measures to improve protection against flooding by flood resistant doors and penetrations sealing have been implemented. Also, sand bags have been provided on-site to be used as temporary flood barriers, if required.

The reviews performed after the Fukushima accident confirmed the safety margins available and the design robustness against severe accidents and conditions caused by extreme external events. (23), (16)

The provision of a supplementary uninterruptible power supply for the SCA was also implemented by the licensee as an improvement.

Improvements have been made to Spent Fuel Bay (SFB) for water level and temperature monitoring from outside the SFB building, to facilitate operator actions in preventing a severe accident in SFB. Also, a seismic qualified line to Spent Fuel Bay has been installed, to ensure cooling under severe accident conditions, and have been assured provisions for natural ventilation of vapours and steam evacuation. (23)

The on-site Emergency Control Centre seismic qualification and the communication system during an emergency situation have been improved. (23)

Cernavoda NPP has implemented a set of Severe Accident Management Guidelines (SAMGs), to cope with situations in which the response based on APOPs is ineffective and the accident conditions progress to severe core damage. The SAMGs objectives are: (2)

- to terminate core damage progression;

- to maintain the capability of containment as long as possible;

- to minimize on-site and off-site releases.

The SAMGs for Cernavoda NPP have been developed based on the generic CANDU Owners Group (COG) SAMGs for a CANDU-600 type of plant. In developing the generic SAMGs, COG adopted the Westinghouse Owners Group (WOG) approach, with the necessary technical modifications suitable for implementation in CANDU plants, based on extensive CANDU specific severe accident analysis and research.

Preparation of plant-specific SAMGs was done by customization of the generic COG documentation package for Cernavoda NPP, removing extraneous information not applicable to the plant, incorporating plant-specific details and information and making any other adjustments required to address unique aspects of the plant design and/or operation.

The interface between APOPs and SAMGs was established by introducing the severe accident entry conditions into the APOPs. The interface with the Emergency Plans was provided by making revisions to the existing EPP documentation, to reflect the new responsibilities and requirements arising from the implementation of the SAMGs. Also, all categories of plant personnel involved in the emergency response organisation were trained for SAMG use, and drills are currently being incorporated in the overall Emergency Response Training Program.

The SAMGs have been developed based on the existing systems and equipment capabilities. A limited and focused set of information requirements was defined to support SAMG diagnostics and evaluations. The primary source is from plant instrumentation, supplemented by additional measurements and data expected to be available through emergency response procedures and Computational Aids where appropriate.

The above-mentioned modifications refer to Units 1 and 2 of Cernavoda NPP. The majority of the safety upgrades dedicated to increased protection against severe accidents had been included in the LBD for Units 3 and 4 before the Fukushima accident. (23)

The detailed results of the post-Fukushima Daiichi accident safety reviews have been subject to several public reports, including:

- National Report of Romania for the 2nd Extraordinary Meeting under the Convention on Nuclear Safety (May 2012) - "stress tests" results; (16)

- Report on the implementation of the European ‘stress tests’ by Romania (December 2011); (9)

- National Report of Romania under the Convention on Nuclear Safety, 7th edition (August 2016). (2)

Taking into account the insights from the Fukushima Daiichi accident, new regulations have been issued to reflect necessity of the concept of design extension conditions, qualification of external events, and improving emergency operating procedures and severe accident management guidelines.

After the Fukushima Daiichi accident, in order to increase the protection against severe accidents, several design improvements have been implemented or are under implementation in accordance with the National Action Plan Post-Fukushima (11), (12). The Plan was developed for bringing together the actions identified from regulatory reviews, self-assessments, peer reviews and generic recommendations at international level.

This action plan has been elaborated by CNCAN, based on the safety reviews performed after the Fukushima accident, taking account of the guidance provided by ENSREG. The action plan was issued for the first time in December 2012 and has been reviewed and revised in December 2014 and in December 2017, respectively. WANO Peer Review Missions at Cernavoda NPP (October 2013, November 2015) had a specific section to evaluate the actions taken in response to Fukushima event.

The action plan was reviewed annually by CNCAN to verify the progress with its implementation and revised, as necessary, to reflect any relevant new information and developments.

Although the risk of an accident is very low, several modifications to improve the capability to withstand prolonged losses of power and other challenges, such as the loss of all heat sinks, were implemented.

The licensee has taken measures to increase the protection against SBO (Station Black-Out) and LOUHS (Loss of Ultimate Heat Sink) scenarios, has implemented operational provisions and performed a number of design changes for this purpose:

- Two new emergency operating procedures for responding to SBO and Loss of Spent Fuel Pool Cooling events were developed and issued.
- Procurement for each unit of an additional mobile DG set (1.2 MW) (to cover entirely the EPS loads). In order to minimize the time for connecting the mobile Diesel generators to the critical loads, the licensee has installed special connection panels to the existing EPS buses.
- Provision of a mobile Diesel engine driven pump and flexible conduits to supply fire water trucks, under emergency conditions;
- Provision of two electrical mobile submersible pumps powered from mobile DG to supply firewater truck, under emergency conditions;

- Provision of two mobile diesel generators (110Kw) for electrical power supply to two domestic water pumps to supply firewater truck, under emergency conditions;
- The seismic robustness of the existing Class I and II batteries has been improved;
- Provision of two separate means to manually open the MSSVs (Main Steam Supply Valves) after SBO;
- Provision of connection facilities required to add water using fire fighters' trucks and flexible conduits to supply the primary side of the RSW/ RCW heat exchangers and SGs under emergency conditions;
- Facilities for water addition to the calandria vessel and to the calandria vault, and increase of the in-vessel retention reliability;
- Installation of PARs for hydrogen management;
- Provisions of a seismic qualified fire-water line to Spent Fuel Bay from the S/B exterior, and of natural ventilation of vapours and steam evacuation;
- Seismic qualification improvement of the on-site Emergency Control Centre;
- Installation of satellite phones in each unit Main Control Room (Intervention Support Centre) and Secondary Control Area.
- Emergency filtered containment venting systems;
- Improvements to the instrumentation necessary to support SAMG implementation (monitoring safety parameters in severe accident situations);
- Special system for hydrogen concentration monitoring in different areas of the Reactor Building.
- Completion of the off-site Emergency Control Centre.

All the design changes associated with the proposed improvements have been subject to CNCAN review and approval. Numerous on-site inspections have been performed to assess the progress of improvement actions resulted from the stress tests.

CNCAN monitors the licensee's progress in the implementation of the planned improvements and continues to perform safety reviews and inspections to ensure that all the opportunities for improvement are properly addressed taking account of the lessons learned from the Fukushima accident.

3 CASE STUDY

3.1 *Legal Implementation in Romania*

In Romania, the legislation and rules are in place, the regulator has confirmed the plan of upgrades in the country and the upgrades of the nuclear power plant have been commissioned. Currently, the commissioning of the upgrades has been completed.

The DiD principle and the 10th year periodic safety review have been already integral part of the Romanian nuclear legislation when Council Directive 2014/87/Euratom of 8 July 2014 (10) was published.

<u>Article 8a</u>

Article 8a states that

> "*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.*"

The regulation "Fundamental Nuclear Safety Requirements for Nuclear Installations" (25) includes requirements transposing the provisions of the Council Directive 2014/87/Euratom (July 2014) (10), establishing a Community framework for the safety of nuclear installations. The regulation includes the provision of Article 8a of the new directive:

It is specified that to fulfil the general nuclear safety objective, *[…] 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.

This objective will be 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.

Safety goals currently in use in Romania include:

- *Dose- frequency criteria* (maximum doses allowed for accidents of specified frequencies and / or maximum frequency allowed for accidents leading to doses in a certain range); these are established in the regulation on design and construction of NPPs (26);
- *CDF (Core Damage Frequency) and LERF (Large Early Release Frequency)* values based on INSAG-12; these are not formalised in regulations, but are used in review and assessments, in accordance with the principles outlined in paragraph 27 of INSAG-12. The results of Level 1 PSA (frequencies of occurrence for different accident sequences leading to core damage) and Level 2 PSA (type and amount of radioactive substances that could be released in the environment as the containment safety function is lost, together with the frequency of these releases) are necessary. (27)

It is recommended that the cumulative frequency of all events that could result in the exposure of a person outside the exclusion area, beyond the effective dose limit for the population, to be less than 1E-3 per year. (26)

The Romanian regulation uses the same wording as in the directive.

"Early" refers to any measures which are required before there is sufficient time to implement them. In this case, early will be judged in relation to the time needed for taking various protective actions, such as evacuation, for example (evacuation should be used as reference, because sheltering and administration of KI pills can be done very fast).

"Large radioactive releases that would require protective measures that could not be limited in area or time" are understood as long-term contamination of the environment / contamination with long-lived radionuclides (Cs-137, Cs-134 being the most significant).

A guidance on assessing the achievement of the general nuclear safety objective (28) which specify the recommendations of CNCAN, regarding the assessment of the fulfilment of the general nuclear safety objective established by the regulation Fundamental Nuclear Safety Requirements for Nuclear Installations, was approved by the CNCAN President in December 2018.

The following quantitative nuclear safety objectives are recommended: (28)

a) Frequency of releasing into the environment a quantity of radioactive material that would require the temporary evacuation of the population from the vicinity of the nuclear site, quantified as the sum of the frequencies of all accident sequences with the source term higher than 1000 TBq of Iodine-131, to be less than 1E-5 / year.

This quantitative objective aims at avoiding early / intempestive releases of radioactive materials, which would require off-site emergency response measures without sufficient time to implement them. For accident sequences for which the source terms exceed 1000 TBq of Iodine-131, it should be demonstrated that the emission of radioactive material cannot occur in such a short time that it does not allow the population to be evacuated from the vicinity of the site.

b) Frequency of releasing into the environment a quantity of radioactive material that would require relocation of the population near the site, quantified as the sum of the frequencies of all accident sequences with the source time higher than 100 TBq of Cesium-137, be smaller than 1E-6 / year. This quantitative objective aims at avoiding high releases of radioactive materials which would require protection measures that cannot be limited in space or time.

c) The cumulative frequency of all accident sequences that can lead to effective doses higher than 100 mSv in the first 7 days for which the population in the vicinity of the nuclear facility is required to be evacuated in accordance with the generic criteria of the Regulation of Emergency Situations Management for nuclear or radiologic risc, to be less than 1E-5 / year.

d) The cumulative frequency of all accident sequences that can lead to effective doses higher than 100 mSv in the first year for which temporary relocation of population located near the site is required according to the Generic Criteria of the Regulation of Emergency Situations Management for nuclear or radiologic risc, be less than 1E-6 / year.

The use of the quantitative nuclear safety objectives referred above is not required for nuclear installations where it is demonstrated that a hypothetical, physically possible, sequence with an estimated frequency higher than 1E-7 / year, which represents the worst case scenario for the respective installation, cannot lead to population exposure and / or environmental contamination so that it is necessary to evacuate or relocate the population from the site vicinity.

In order to assess the fulfilment of the quantitative nuclear safety objectives, deterministic and probabilistic nuclear safety assessments are necessary to be developed, in accordance with the requirements of applicable CNCAN norms and internationally recognized standards and best practices. The analyses should cover all operational modes of the nuclear installation and should take into account all internal and external initiating events relevant to the installation and to site.

-both design basis accidents as well as the design bases extending conditions, including severe accidents, will be considered;

-accident scenarios affecting several nuclear installations located on a common site will also be considered;

-systems, structures, components and equipment that are qualified and / or can be justified with a sufficient degree of confidence that they will perform their functions and will ensure the control, respectively the mitigation of the consequences of the accidents considered in the analyses and evaluations should be credited in the assessments;

-operators' actions that are feasible under the considered conditions will be credited, taking into account all relevant factors including the availability of control rooms, measurement instrumentation and nuclear facility status information, operator training and qualification, radiological conditions and other potential hazards;

-only realistic assumptions will be used, as far as practicable. Uncertainties and sensitivity analyses should be performed. It is recommended that the methodology and assumptions used in analysis be submitted to CNCAN for verification and acceptance prior to performing these analyzes and evaluations;

The time for implementation of measures to protect the population, particularly with regard to the evacuation of the population near the nuclear site will be established and justified based on an assessment that takes into account the existing emergency plans and the results of the emergency exercises.

Article 8b

In article 8b of Council Directive 2014/87/Euratom the main elements of the DiD concept are required. These are:

- "*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*".

In addition, it is required that

> "*the impact of extreme external natural and unintended man-made hazards is minimised*"

In Article 8b it is now specified, that the required management system in Article 6(d)

> "*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*".

Further requirements deal with the operational experience feedback and the reporting obligations of safety relevant events to the regulatory body. With the requirements stated in Article 8b

> "*arrangements by the licence holder to register, evaluate and document internal and external safety significant operating experience*"

and

> “*the obligation of the licence holder to report events with a potential impact on nuclear safety to the competent regulatory authority*”

In Romania, it is required by regulations that an effective DiD shall be implemented in all nuclear power plants, in order to prevent, or if prevention fails, to mitigate harmful radioactive releases.

It is required that the concept of DiD should be applied to nuclear installations, in all activities with impact on nuclear safety, with the following objectives: (25)

- minimizing the impact of extreme external hazards of natural origin as well as those caused unintentionally by humans;
- prevention of failures and abnormal operating conditions;
- detection and repair of defects and control of abnormal operating conditions, in order to prevent accidents;
- control of design basis accidents in such a way as to prevent the exceeding of the nuclear safety margins;
- control of severe conditions that could be caused by multiple failures such as the complete loss of all functions of a safety system or an extremely unlikely event, including the prevention of large-scale accidents and the mitigation of the consequences of severe accidents;
- on-site emergency management.

The safety philosophy of CANDU reactors, based upon the principle of DiD, employs redundancy (using at least two components or systems for a given function), diversity (using two physically or functionally different means for a given function), separation (using barriers and/or distance to separate components or systems for a given function), and protection (seismically and environmentally qualifying all safety systems, equipment, and structures). (2)

An important aspect of implementing DiD in the NPP design is the provision of a series of physical barriers to confine radioactive material at specified locations. In CANDU design these barriers are the fuel matrix, the fuel clad, the Heat Transport System, and the Containment. An additional administrative barrier is the exclusion area boundary.

For design purposes, the safety related systems and structures have been defined as those which, by virtue of failure to perform the safety functions in accordance with the design intent, could cause the regulatory dose limits for the plant to be exceeded, in the absence of mitigating system action.

The safety related systems and structures of a CANDU NPP can be categorized in:

Preventive: Systems and structures that perform safety functions during the normal operation of the plant, to ensure that radioactive materials remain within their normal boundaries. These are systems and structures whose failure could cause a release exceeding the regulatory dose limits during normal plant operation, in the absence of further mitigating actions, or whose failure as a consequence of an event could impair the safety functions of other safety related systems.

Protective: Systems and structures that perform safety functions to mitigate events caused by failure of the normally operating systems or by naturally occurring phenomena.

Some systems may perform both protective and preventive safety functions, and therefore may have more than one safety category designation.

The protective systems defined above are further identified as:

- Special Safety Systems, which include Shutdown System No. 1, Shutdown System No. 2, Emergency Core Cooling, and Containment.

- Safety Support Systems, which provide essential services needed for proper operation of the Special Safety Systems (e.g., electrical power, cooling water). These systems may have normal process functions as well.

For the purpose of safety assessment all major systems in CANDU reactors are categorized as "process systems" and "special safety systems". The Special Safety Systems are always in stand-by during the normal operation of the plant and ready to mitigate the consequences of any serious process failure. They are totally independent from all process systems and from each other.

The CANDU safety philosophy is based on the concept of single/dual failures. "Single failure" is a failure of any process system which is required for the normal operation of the plant and "dual failure" represents a combination of the single failure events and a simultaneous failure or impairment of one of the special safety systems. Coincident failure analysis is a systematic assessment of postulated dual failures.

Each postulated process failure is systematically coupled with a failure of one of the special safety systems. Loss of the shutdown systems is excluded from required dual failure sequences because the design includes two independent shutdown systems which are each capable of shutting down the reactor.

The integrated and coordinated management system of Cernavoda is ensuring that the nuclear safety requirements have priority over any other requirements, considerations, and interests.

The licensee has established and implemented a systematic process for reporting, collecting, sorting, analyzing and documenting the operating experience and events from the nuclear installation for which it is responsible, as well as the relevant events and operating experience reported by other nuclear installations, nuclear organizations and organizations from other industrial sectors at national and international level. (29) The licensee is promptly informing CNCAN of any event with potential impact on nuclear safety.

The regulation NSN-06 on the protection of nuclear installations against external events of natural origin (30) has been published in January 2015. No regulatory guides in relation to hazards were issued yet, but CNCAN has endorsed the general guide issued by WENRA / RHWG to support the implementation of the reference levels in Issue T.

<u>Article 8c</u>

According to Article 8c the periodic safety review shall ensure

> ”*(…) compliance with the current design basis and identifies further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards, using as a reference the objective set in Article 8a.*”

Periodic safety reviews are performed in accordance with a national regulation which is based on the IAEA safety standards and WENRA reference levels. Opportunities for improvement, including plant upgrades, are identified based on the review against the latest standards and implemented. In addition, safety reassessments, including new or revised safety analyses, are performed every time new information, significant in relation to the prevention and / or mitigation of nuclear power plant accidents, including severe accidents, becomes available, from operational experience or from research activities.

The Romanian regulation is based on the IAEA Safety Guide NS-G-2.10 (22), having the 14 “safety factors” defined as “areas of review”, for each of these having specified most of the “generic review elements”. (2)

Periodic review of nuclear safety must be carried out systematically and periodically, at least once every 10 years. and should be done to ensure compliance with the current / updated design bases and to identify new improvements in nuclear safety, taking into account the cumulative effects of ageing of the nuclear installation, modifications, operational experience as well as the results of the updated nuclear safety analyses, the latest research results, scientific and technological developments and the latest internationally recognized standards and best practices.

Deterministic and probabilistic analyses and assessments, as well as engineering judgment, should be used in the review. (25)

The authorization holder shall review and revalidate nuclear safety analyses periodically at least once every 10 years from the start of operation of the nuclear installation to demonstrate that their assumptions remain valid and that the effects of ageing are effectively controlled, that the nuclear safety margins are maintained throughout its lifetime. (31)

The PSR must determine: (24)

- to what extent the NPP meets current international nuclear safety standards and good international practice;
- the completeness and validity of the nuclear safety documentation;
- if adequate measures are in place to continue to safely operate the plant until the next PSR or the end of the plant lifetime;
- the necessary corrective actions to be implemented and the improvements that can be made to increase the nuclear safety of the plant.

3.2 *Latest safety upgrades*

Using as model the Slovenian report (32), below are specified the safety upgrade of existing structures, systems and components and other measures and new systems that are important to provide nuclear safety during severe conditions.

The safety upgrades were separated into the prevention and mitigation part. The prevention part of the equipment serves to preserve adequate fuel cooling in case of DEC events.

The situation of implementation of Cernavoda (U1 and U2) Safety Upgrades is presented in the below table.

Table 2: Status of implementation, 2018

Upgrade (equipment, system, logistic)	Description	Purpose	Scheduled finish

Installation of PARs for hydrogen management	Installation of passive autocatalytic recombiners for hydrogen management in Cernavoda Units 1 and 2. The installed recombiners are FR1-380T type. At the same time, a hydrogen monitoring system (Hermetis), to monitor R/B H2 concentration was installed.	*Mitigation* -Improve the existing R/B Hydrogen Control Strategies	*2012 (U1) 2013 (U2)* *Implemented*
Filtered venting system	Installation of dedicated emergency containment filtered venting system for each NPP unit – capable to depressurize the containment and filter the volatile fission products	*Mitigation* -Improve the existing R/B Envelope protection strategies. -Reduce the environmental impact of a controlled release.	*2013 (U1)* *2014 (U2)* *Implemented*
Improvement of the seismic robustness	Seismic margin assessment has not revealed a need for any safety significant design change. However, several recommendations resulted from these inspections, such as Increasing the seismic robustness of the existing Class I and II batteries, implemented by the licensee as part of the regular plant seismic housekeeping program	*Prevention* -Reduce operator burden – improve operating crew response time. -Improve CSP monitoring capabilities.	*2011* *Implemented*
Increase the flood protection	Design modifications to replace selected doors with flood resistant doors and penetrations sealing (for improving the volumetric protection of the buildings containing safety related equipment located in rooms below plant platform level). Provision of on-site of sand bags to be used as temporary flood barriers, if required.	*Prevention and mitigation* -Improvement of volumetric protection of the buildings containing safety related equipment located in rooms below plant platform level (so that	*2011* *Implemented*

		protection does not rely solely on the elevation of the platforms)	
Additional instrumentation for SA management	Improvements of the reliability of existing instrumentation by qualification to severe accident conditions and extension of the measurement domain. Implementation of hydrogen concentration monitoring in different areas of the reactor building. The design changes implemented at both Cernavoda Units to improve survivability to SA addressed the following parameters: - R/B pressure, - Calandria Vault level, - moderator level, - Heat Transport temperature	*Mitigation*	*2014 (U1)* *2015 (U2)* *implemented*
Design modification for water make-up to the calandria vessel and the calandria vault	Implementation of a design modification for water make-up to the calandria vessel and the calandria vault. Installed connections to inject water to the Calandria Vessel (via the Moderator Purification lines). Installed new line to inject water into the Calandria Vault from outside of the R/B.	*Mitigation*	*2012 (U1)* *2013 (U2)* *Implemented*
Spent fuel pool improvements for severe accident management	Improvement of the existing provisions to facilitate operator actions to prevent a severe accident in the spent fuel pool (install hydrogen vent capability, natural ventilation for vapours and steam evacuation, seismically qualified fire-water pipe for water make-up, water level and temperature monitoring from outside the SFP building)	*Prevention*	*2014 Implemented*
Mobile equipment for emergency	Procurement and testing of mobile equipment (e.g. mobile diesel generators, mobile pumps, connections, etc.). Provision of connection facilities required to add	*Mitigation* -Improve the existing level	*Implemented*

	water using fire fighters trucks and flexible conduits to supply the primary side of the RSW/RCW heat exchangers and SGs under emergency conditions.	of DiD for SBO	
New seismically qualified location for the on-site emergency control centre and the fire fighters	Cernavoda NPP is establishing a new seismically qualified location for the on-site emergency control centre and the fire fighters. The location includes important intervention equipment (mobile DGs, mobile diesel engine pumps, fire-fighter engines, radiological emergency vehicles, heavy equipment to unblock roads, etc.) and will be protected against all external hazards.	*Prevention and mitigation*	*In progress*

It can be concluded that all the most important safety-related upgrades have been implemented. The implementation of establishment of a new seismically qualified location for the on-site emergency control centre and the fire fighters is in progress. This location will include important intervention equipment (mobile diesel generators, mobile diesel engine pumps, fire-fighter engines, radiological emergency vehicles, heavy equipment to unblock roads, etc.) and will be protected against all external hazards.) The target date for implementation was initially set for the end of 2015, but it has been changed several times due to legal and administrative issues related to transfer of property of the physical location. Until the completion of this action, equivalent measures have been implemented to ensure that all intervention equipment (mobile Diesels, Diesel fire pump, fire trucks, etc.) are protected from external hazards (e.g. the equipment have been relocated so that they would not be impaired by external events).

3.3 *Discussion*

The Romanian safety upgrades have been driven mainly by the licensee, benefiting from the strong support of the regulatory body in elaboration of the concepts, technical solution and the schedules for the implementation. CNCAN has actively monitored the implementation status of the safety improvements. Most relevant safety upgrades were implemented after Fukushima accident.

Maintaining an equivalent level of nuclear safety for all units on site could be also a good driver for safety upgrades. Taking advantage of the similar design of all nuclear units on Cernavoda site, and on the fact that commissioning was realized one by one for the units (they were not commissioned in the same year), the feedback program on Cernavoda site was very useful, assuring that design modifications and improvements from Unit 2 that are reasonably practicable were implemented also on Unit 1. This could be seen as a good practice for safety upgrades.

Periodic safety reviews represent real opportunities to identify and implement the necessary safety improvements.

Safety benefits should be the strongest argument for changing the regulations. There is a continuous effort of CNCAN to harmonize its practices with those already implemented by international community. Being a member of international working groups relevant to nuclear safety, CNCAN is following the developments and recommendations arising from these dedicated group efforts. This effort leads also to development of new regulations and as result, to requirements for new safety upgrades in the plant.

Challenges

Regulatory driven safety upgrades require substantial resources for elaboration of well-defined regulations.

There is a continuous effort of CNCAN to harmonize its practices with those already implemented at international level. The process of updating regulations is in progress, and there are plans for reviewing all the standards for harmonization. Due to the on-going process of updating, currently both concepts “beyond design basis” and “design extended conditions” are used in the regulations, which might be confusing. The new developed regulations are referring to “Design Extended Conditions” (30), while in the old ones - for instance in (26), only the “Beyond Design Basis” concept is mentioned.

4 CONCLUDING REMARKS

The legislation is available only in Romanian language (with the exception of (25)). The technology vendor's country requirements (Canada) together with international requirements have also been endorsed.

The approach for legal implementation of the Articles 8a-8c of Directive 2014/87/EURATOM was consistent with developments in the international nuclear communities.

Implementation of plant safety upgrades in Romania is driven by both approaches, bottom-up and top-down.

Safety upgrades can result from the following situations: due to development of technology (resulting in improved performance or reliability of operation); following PSR results; following international recommendations; following the revision of codes, standards and regulatory requirements.

Licensee driven safety upgrade process is fully consistent with the primary responsibility for the nuclear safety. In this process, the licensee has been proposing the concepts, technical solutions and the schedules for the implementation. CNCAN supported these activities by approvals of those concepts, technical solutions and the implementation schedules.

The licensee is benefiting from the membership in international organizations, seeking to achieve the highest levels of operational safety and performance. The membership in similar technology owners' associations provide the necessary support to resolve technical and operating problems for all members, and the recommendations arising from international memberships may constitute drivers for safety upgrades in the plant.

Maintaining an equivalent level of nuclear safety for all units (same design) on site could be a good practice for implementing safety upgrades.

Periodic safety reviews remain an efficient mean and opportunity to identify the week points in technology and to correct them.

Regulatory driven safety upgrades require substantial resources for elaboration of well-defined regulations. There is a continuous effort of CNCAN to harmonize its practices with those already implemented by international community, which leads also to development of new regulations and as result, to requirements for new safety upgrades in the plant.

Complex changes in safety field should receive sufficient attention, time and resources, and should involve all the necessary parties to achieve good results.

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IMPLEMENTATION IN PRACTICE OF ARTICLES 8A-8C OF DIRECTIVE 2014/87/EURATOM
REACTOR SAFETY UPGRADES IN HUNGARY

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Table of Contents

1. Introduction

By enacting Council Directive 2009/71/Euratom Member States are obliged to establish a common framework for nuclear safety. In the aftermath of the accident at the Japanese Fukushima Daiichi Nuclear Power Plant this Council Directive was amended by Council Directive 2014/87/Euratom. By Article 8a of this Council Directive a nuclear safety objective common for all Member States has been established.

In Article 8a it is stated, that

> *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.*"

It is further specified that the above mentioned nuclear safety objective has to be applied to all

> *"nuclear installations for which a construction licence is granted for the first time after 14 August 2014"*.

In addition, it is required that the nuclear safety objective serves as a

> *"reference for the timely implementation of reasonably practicable safety improvements to existing nuclear installations"*.

As Council Directive 2014/87/Euratom introduces several terms, but without definition or glossary, the meaning of the terms leaves room for interpretation by the Member States and thus challenging a harmonized approach in Europe.

The objective of Task 3 of the project under Contract No. ENER/17/NUCL/S12.769200 is to perform an analysis of the practices and approaches used to implement safety upgrades of nuclear power plants in selected Member States.

This report not only outlines the legal framework for nuclear safety in Hungary, but safety improvements on Paks Nuclear Power Plant, furthermore efforts on implementation in practice of Articles 8a-8c of Directive 2014/87/Euratom are also described in this document.

2. Legal framework in Hungary

The Hungarian Parliament approved the Act CXVI on Atomic Energy (hereinafter referred to as the Act on Atomic Energy) in December 1996, it entered into force on July 1, 1997. The Act regulates not only the nuclear facilities containing reactors, but all uses of the atomic energy.

Act on Atomic Energy stipulates that "in the use of atomic energy, safety has priority over all other aspects", and that “the Licensee is obliged to undertake continuous activities to improve safety, taking account of its operational experience and the new safety related information” in harmony with the spirit of the Convention on Nuclear Safety.

The main characteristics of the Act on Atomic Energy are:

- declaration of the overriding priority of safety;
- definition and allocation of tasks of ministries, national authorities and bodies of competence in licensing and oversight procedures;
- entrusting the facility-level licensing authority of nuclear installations to the Hungarian Atomic Energy Authority (HAEA) and declaration of its organizational and financial independence;
- declaration of the need for utilizing human resources, education and training, and research and development;
- definition of the responsibility of the Licensee for all damage caused by the use of nuclear energy, and fixing the sum of indemnity in accordance with the Revised Vienna Convention;
- giving the national authority (HAEA) the right to impose fines.

Several government decrees and ministerial decrees have been issued to implement the requirements of the Act on Atomic Energy, the most important are the following:

- Govt. decree 118/2011. (VII. 11.) on the nuclear safety requirements of nuclear facilities,
- Govt. decree 487/2015. (XII. 30.) on the protection against ionizing radiation and the corresponding licensing, reporting and inspection system.

New version of nuclear safety requirements (Nuclear Safety Code – NSC) was issued as an annex to Govt. Decree 118/2011 (VII. 11.) in July, 2011.

Volume 3 contains general nuclear safety-related requirements concerning the design of nuclear power plants. It specifies the detailed safety requirements for nuclear facilities, the related authority procedures (licensing, approval, inspection, assessment and enforcement), and also covers some additional general safety approach issues. The requirements lay down the principles and rules well known from international practice. The requirements reflect the most recent nuclear safety standards and stipulate in detail the principles being commensurate with the international practice.

It also contains the definition of three safety goals:

- the general nuclear safety goal: defence against ionizing radiation by realization and adequate maintenance of effective safety measures,
- the radiation protection goal: radiation doses for professionals and population being “As Low As Reasonably Achievable” (ALARA) and always below the prescribed limits in all phases of the life-cycle of the nuclear facility, what shall be maintained for events

belonging to design basis conditions (DBC) and – as much as rationally practicable – for events belonging to design extension conditions (DEC),

- the technical safety goal: occurrence of incidents shall be avertable or precluded by high reliability, possible consequences shall be within the acceptable margins for all postulated initiating events taken into account in the design of the nuclear facility, the probability of the accidents shall be as low as required.

These safety goals shall be effective in all phases of the life-cycle, such as design, siting, fabrication, construction, commissioning, operation and decommissioning. Nuclear Safety Code contains also the requirement of continuous improvement of the safety level (including modifications for any reason) taking into account the internal and external experience.

It requires that the defence-in-depth principle shall apply to each safety related activity in such a way that any failure can be compensated for or corrected, and the occurrence of a severe accident situation can be prevented. In addition, such specific supplementary systems, structures or components shall be provided for the protection of the public and the operating staff that are designed to mitigate the consequences of beyond design basis accidents.

According to the requirement of the Act on Atomic Energy, the regulations shall be periodically (at least every 5 years) reviewed.

The Hungarian regulations are reviewed and revised on a more frequent basis than the 5-year review period stipulated by law. The reviews resulted in amendments to the Govt. decree 118/2011. (VII. 11.) on the nuclear safety requirements for nuclear facilities and the corresponding regulatory activity.

An important aspect of the regulation development was to take into account the recommendations of international organizations, especially the recommendations of the Western European Nuclear Regulators' Association (WENRA) on international good practice, the so-called reference levels, as well as the good practice of certain countries with developed nuclear technology. WENRA reference levels were established for the first time in 2008, the review of the requirements was completed in 2013. The new reference levels have been incorporated by Hungary in the Nuclear Safety Code.

The legislator utilized the experience gained regarding the Finnish and British regulations, as well as the national licensing experience.

In order to be prepared for the construction of the new nuclear power plant units, the Volume 3/A of the NSC on nuclear safety requirements to be applied during the design was published; in addition, the HAEA developed further the requirements for the design and construction period as well.

Annexes of Govt. decree 118/2011. (VII. 11.) are:

1. Nuclear safety regulatory procedures of nuclear facilities;
2. Management systems of nuclear facilities,
3. Design requirements for nuclear power plants;
3a. Design requirements for new nuclear power plant units;
4. Operation of nuclear power plants;
5. Design and operation of research reactors;
6. Interim storage of spent nuclear fuel;
7. Survey and evaluation of the site of nuclear facilities;
8. Decommissioning of nuclear facilities;
9. Requirements for the design and construction periods of a new nuclear facility;
10. Terminology used in the Nuclear Safety Code (definitions).

Similar to the Nuclear Safety Code published as annexes to the Govt. decree 118/2011. (VII. 11.) the Repository Safety Code (hereinafter referred to as RSC) was developed, which volumes were published as annexes to the Govt. decree 155/2014. (VI. 30.).

2.1 Assessment of safety

The method of preparation and implementation of safety analysis reports is set out in acts and in Government Decrees. The official procedure related to a nuclear construction is based on the Preliminary Safety Analysis Report necessary for the commencement of commissioning, which is followed by the Final Safety Analysis Report necessary for the commencement of operation of a given nuclear installation.

Safety requirements are differentiated for the existing and planned NPP units, at the same time requiring the implementation of the best international safety practices through periodic safety reviews for the sake of the continuous safety improvement.

The requirements regarding the contents of safety reports are based on the requirements of Reg. Guide 1.70 of the US NRC (United States Nuclear Regulatory Commission) taking into account national characteristics.

Govt. decree 118/2011. (VII. 11.) stipulates that the Final Safety Analysis Report should be updated annually, so that the safety analysis report can serve as an authentic and continuous basis assessing the safety of the nuclear installation throughout its entire life-time.

Periodic nuclear safety review is performed within ten years of the first day of the validity of the Operating License issued for the initial commencement of operation, and it is repeated every ten years following the first one. Authority issues a decision based on its own safety assessment and the Periodic Safety Review Report of the Licensee, in which it lays down the conditions for future operation.

3. Safety improvements on Paks Nuclear Power Plant

The Paks NPP Co. operates four VVER-440/213 type reactor units, originally designed to produce 1375 MW_{th} and 440 MW_e each. Both the moderator and the coolant of the reactors are light water. On the basis of its safety philosophy, the power plant belongs to the group of second-generation VVER-440 nuclear power plants. The reactor has 6 cooling loops, each one is connected to a steam generator. Each power plant unit is supplied with a so-called localizing tower (operating on the bubble condensing principle) connected to airtight compartments for handling any incidents caused by pipe ruptures. In these towers, trays filled with water containing boric acid are layered one above the other, completed with air traps. This system of hermetic compartments and localizing towers makes up the containment for the reactors.

Each unit is installed with three active safety systems, and in case of incidents their electrical supply might be ensured by diesel generators. These systems are supplemented by passive systems. Two saturated (wet) steam turbines operate in each unit.

The four units were commissioned between 1983 and 1987 and are currently in good technical condition. Improvement of safety of Paks NPP has been implemented in several steps in the last decades. Milestones are summarized in this session.

3.1 AGNES project and safety enhancement measures

During the commission of the Paks NPP Co., Hungarian practices followed those accepted in developed countries. Based on the Technical Design provided by the supplier, a Preconstruction Safety Analysis Report was prepared, which was followed by the Pre-commissioning Safety Analysis Report that was aimed at providing the basis for the Final Safety Analysis Report. As time passed gradually more deficiencies were revealed in the Safety Analysis Reports when compared to Western requirements. For this reason, re-evaluation of the safety of the nuclear power plant was necessary.

It was demonstrated that safety level of the Paks NPP corresponds to international standards, however, some safety enhancement measures were necessary to comply completely with increased demands of reactor safety. For this reason, Atomic Energy Research Institute initiated and the Hungarian Atomic Energy Commission established the project of Advanced, General and New Evaluation of Safety (AGNES) in 1991.

The reassessment of safety of the Paks NPP according to internationally recognized criteria was performed during the project between 1991 and 1994. The following groups of analysis have been performed: system analysis and description; analysis of design basis accidents; severe accidents analysis; level 1 probabilistic safety analysis. Postulated accidents (PA) and Anticipated Operational Occurrences (AOO) are estimated in detail for the following initiating events: increase/decrease in secondary heat removal; decrease in primary coolant inventory; increase/decrease of reactor coolant inventory; reactivity and power distribution anomalies; analysis of transients with the failure of reactor scram (ATWS); pressurized thermal shock analyses. Severe accident analysis was made for the accidents on in-vessel phase and containment phase, for radioactive release and for accident management.

It was demonstrated that there were several alternatives to enhance the safety of the nuclear power plant, e.g. some equipment should be replaced, complementary safety systems should be built in. Feasible measures were identified and implemented.

Site-specific earthquake characteristics were comprehensively re-evaluated, resulting in new input data with higher Peak Ground Acceleration (PGA) values for the design basis. Complex earthquake resistance measures were implemented for all relevant systems, components and structures.

In line with the analyses and results of the AGNES project, the safety of the plant has been enhanced significantly in proportion to the expenditure.

3.2 Power uprate and implementation of 15-month operation cycle

The original nominal thermal power of each unit of Paks NPP was 1375 MW, and the nominal electric power outputs of each unit were 440 MW. As a result of the power uprating programme realized between 2006 and 2009, the thermal power of each unit has been increased to 1485 MW and the electric power to 500 MW. As a first step upgrades of the secondary circuit and turbine resulted in about 470 MW_e with an unchanged thermal capacity at all four units. Later an upgrade of the primary side has been performed to increase the nominal power by 8% to 1485 MW_{th}, resulting in about 500 MW_e generated power by each unit. The power increase was primarily reached with refined primary pressure regulation, core control system upgrade and new type of fuel assemblies. Additional modifications have been performed in certain technological components and by the end of 2009 the upgrading process has been completed successfully on the four units.

Following the power uprate of the reactor units and also for economical reasons, a new fuel type with higher enrichment was introduced at the Paks NPP. As a result of this both the amount of necessary fresh fuel and the amount of the spent fuel was decreased. Another important feature of the new fuel is its burnable poison content, which is meant to increase operational safety.

In the beginning of 2013 Paks NPP declared its intention to operate the units of the nuclear power plant at a 15-month operation cycle, instead of the previous 12-month cycle period,. The most visible change due to the increase of the cycle period is that instead of 5 only 4 main refuelling outages were performed during five years at a unit. The increased operation cycle required the application of a new, higher enriched nuclear fuel type.

3.3 Stress-test after Fukushima

After the accident at the Fukushima Daiichi NPP the safety of all EU nuclear power plants had to be reviewed, on the basis of comprehensive and transparent risk and safety assessment [1]. The stress tests (Targeted Safety Re-assessment – TSR according to the official Hungarian denomination) consisted of three main steps:

- self-assessment by licensees,
- independent review of the results and preparation of a national report by the national authorities,
- international peer reviews.

Hungarian Atomic Energy Authority issued the requirements for operator's re-assessment [2], shortly after the publication of the ENSREG requirements [1]. The nuclear power plant completed the re-assessment and then the Authority prepared and submitted the national report [3] to the European Commission by deadline.

The following general statements were declared on the basis of international peer-review:

- Paks NPP is in compliance with the licensing conditions, is able to withstand the loads induced by a design basis earthquake, flood or by extreme weather conditions and the facility is prepared for those design basis events which entail the total loss of the electric power supply or the ultimate heat sink.

- The design basis established during the construction of the plant was extended through a series of safety improvement programmes.
- Regulatory requirements did not exist for events beyond the design basis at the time of the construction of the plant, but were established later and the plant is in compliance with them thanks to the completed modifications.

Several corrective actions were proposed in order to increase the safety margins, a detailed implementation action plan was elaborated and executed. Implementation of these modifications were required by the authority for the lifetime extension. [4]

After the implementation of all corrective actions, the following results were achieved:

- The occurrence probability of severe accidents due to the permanent loss of electric power supply and ultimate heat sink is decreased.
- Severe accidents of reactors and spent fuel pools can be prevented or mitigated by the provision of an alternative water supply and electric supply routes.
- Extreme external events may cause damages to the site, but the risk of damage occurrence and the consequences of such events are reduced.
- The capability to prevent and/or mitigate accidents simultaneously affecting more units is enhanced.
- The solutions that can be utilized for emergency response are extended, including accident situations simultaneously affecting more than one unit.

3.4 Lifetime extension

Subsequent to the end of the design service lifetime of 30 years, lifetime extension of additional 20 years was performed on Unit 1 in 2012, on Unit 2 in 2014, on Unit 3 in 2016, on Unit 4 in 2017.

3.5 Periodic safety review

Periodic safety review (PSR) is performed every ten years. In line with the recommendation of the International Atomic Energy Agency (IAEA), in the framework of PSR the technical status and safety level of the facility shall be reviewed. As an outcome of the review, in order to decrease the risk, the licensee has to implement safety improvement measures.

Detailed requirements regarding the PSR are included in Nuclear Safety Code, e.g. the minimal contents of PSR shall contain:

- The design of the nuclear facility documented in the Final Safety Analysis;
- Site characteristics, resistance to external hazard factors;
- Decommissioning;
- The current condition of systems and system components;
- Equipment qualification;
- Ageing;
- Deterministic safety analysis;
- Probabilistic safety assessment;
- Analysis of hazard factors;
- Safety indicators of the nuclear facility;
- Evaluation and feedback of relevant technical and scientific results, and operational experience; Utilization of research results and the experience of other similar nuclear facilities;
- Organization, human factors, management system and safety culture;
- Procedures;

- Accident management;
- Nuclear emergency preparedness;
- Radiation protection of employees, the population and the environment;
- Experimental equipment in the case of research reactors.

For supporting the review and assessment procedure, specific guidelines were released by the Authority.

The first Periodic Safety Review of units 1 and 2 of Paks NPP took place in 1995-1996. The Periodic Safety Review of Units 3 and 4 was performed in 1997-1998 in accordance with the new Act on Atomic Energy (entered into force in 1997) and the related regulations. Within this process, among others the analysis of international and domestic operating experience, scientific and technical advances, as well as the implementation of new safety objectives and requirements are compulsory topics.

A very comprehensive safety upgrade process was defined and approved as part of the first Periodic Safety Review of Paks NPP. Technical measures were designed, approved and implemented.

As a result of the first PSR, it was decided to elaborate and introduce the symptom oriented emergency management procedures. It was implemented in two steps, first for the power operation states and then for the low-power and shutdown states of the reactors as well.

After this, PSR was conducted at the same time for each unit. The second Periodic Safety Review Report was performed in 2008 and 169 safety improvement measures were ordered to be executed in the approval resolution. Last one was completed in 2017, goals for safety improvements were identified.

As a result of the second PSR decisions were made to implement a set of measures to deal with accidents, including

- installation of additional hydrogen recombiners in the confinement for DEC scenarios, including core melt,
- modification of the spent fuel pool cooling system to exclude loss of water from the pool due to large breaks in the cooling system,
- implementation of external cooling of the reactor pressure vessel (RPV) by flooding reactor cavity with the aim to prevent the vessel failure and make possible in-vessel retention of the corium after fuel melt,
- introduction of possibility of connecting part of I&C system and electrical valves needed in severe accident management to an independent electrical system fed by small mobile diesel generators,
- auxiliary severe accident measurement system,
- implementation of blow-down possibility of the pressurizer to prevent high-pressure failure of the RPV.

Implementation of these measures took several years, although after the Fukushima accident it was accelerated and all these measures were completed around 2014.

Last PSR of the units of Paks NPP was finished in January 2019 by the decision of the Authority. This decision reinforced the prescription to implement the remaining open issues from the post-Fukushima action plan and required execution of additional measures. Several new analyses are among them, their results could even initiate modifications of the design and implementation of safety improvement measures.

3.6 International safety reviews

Since its commissioning, Paks NPP has paid special attention to utilizing international experience and, at the initiative of the NPP, more than 40 international reviews have taken place since 1984.

Several international reviews were carried out at Paks Nuclear Power Plant and at the regulatory body by international organizations (e.g. OSART and IRRS mission of IAEA, Peer Review of WANO). Hungary took obligation to adopt the WENRA safety reference levels and status was peer-reviewed by the WENRA RHWG. IAEA IRRS mission in 2015 stated as well, that the Hungarian regulatory system is basically in line with the IAEA Safety standards.

As described above, the safety of the units was improved continuously. Final Safety Analysis Report is regularly updated in line with the regulatory requirements. It is a living document, which follows and analyses the safety impact of different measures and modifications and evaluates safety performance according to international practice. In compliance with the latest international recommendations and the requirements of the European Union, the analyses of incidents in the extended design basis and demonstration of the compliance with the respective criteria have taken place, as well as the safety analysis of external hazard factors was achieved. Technical modifications of the plant were made as a consequence of the implementation of safety improvement measures, and taking into account the results of scientific development.

4. Sustaining the nuclear capacity

Hungary has committed to a new NPP construction project in 2014, with the objective to build two VVER-1200 type units next to the existing units at Paks site. The new NPP project is currently in its early phase, availability of technical and design information is limited.

In January 2014, the Government of Hungary and the Government of the Russian Federation concluded an agreement on the cooperation in the field of peaceful use of nuclear energy, which was promulgated in the Act II of 2014. Among others, the Agreement deals with the cooperation in relation to new nuclear power plant units.

The basic licensing principles of the establishment of nuclear facilities and radioactive waste repositories, and the concerned authorities taking part in licensing proceeding are regulated by Chapter III of the Act on Atomic Energy. To establish a new nuclear power plant, for starting the preparatory work a preliminary consent in principle of the Parliament is required, whereas the acquisition of ownership of a nuclear power plant that is in operation or to transfer the right of operation the consent in principle of the Government is required.

In accordance with regulations in force, licenses shall be obtained from the authorities for all phases of life-cycle of a nuclear power plant, including site selection and evaluation, construction, expansion, commissioning, operation, decommissioning. Moreover, a separate license shall be obtained for all plant level or safety related equipment level modifications.

According to the relevant European Union and international regulations the construction of a nuclear power plant shall be subject to environment impact assessment.

In April 2014 the MVM Paks II Nuclear Power Plant Development Ltd. submitted its application for obtaining the license for site survey and evaluation, and started the preparation for fulfilling the obligations and tasks of its role as a licensee. On 14 November 2014 the HAEA approved the site survey and evaluation programme with certain conditions. The fulfilment of the licensee functions is being continuously reviewed by the HAEA during inspections. In November 2015 the MVM Paks II Nuclear Power Plant Development Co. submitted the Preliminary Safety Information Report (hereinafter referred to as PSIR) to the HAEA. Its aims to provide sufficiently detailed preparatory information to the authority on the safety of the planned nuclear power plant. It includes general design information, but it does not include the site specific design solutions.

Design requirements are different for existing and new nuclear power plants. Some of the reasons for this approach are the following:

- It represents a significant challenge to create design and assessment requirements fully applicable for existing and new units at the same time without being either unimplementable (too strict) or too permissive;
- Design lifetime of new units are at least 60 years, so in order to meet future challenges reasonably stricter requirements should be created (“future-proofing”);
- The scope of design and analysis activities are different in case of an existing and a new unit (plant level initial design of new NPPs vs. modification of an existing plant/technology);
- Design and assessment codes and standards have significantly evolved in the last decades (Gen II vs. Gen III+);
- Some situations that are considered as “beyond design” for existing plants, are considered in the design of new plants (e.g. core melt) as design extension conditions.

In terms of quantitative requirements, the differences are summarized in the following tables. Table 1. shows the probabilistic limiting values for screening out different initiating event types and hazards that have to be considered in the design basis of the NPPs; and Table 2. shows the probabilistic acceptance criteria for CDF and LERF for different NPPs.

Table 1.: Screening limits for initiating events and hazards to be considered within the design

Plant type	Screening limits of initiating events and hazards to be considered within the design basis		
	Internal Initiating Events and Internal Hazards [1/year]	Human induced External Hazards [1/year]	Natural Hazards [1/year]
New NPP	10^{-6}	10^{-7}	10^{-5}
Existing NPP	10^{-5}	10^{-7}	10^{-4}

Table 2.: Probabilistic acceptance criteria for NPPs

Plant type	Acceptance criteria	
	CDF [1/year]	LERF* [1/year]
New NPP	$< 10^{-5}$	$< 10^{-6}$
Existing NPP	$< 10^{-4}$	$< 10^{-5}$**

* Cumulated for all plant conditions (DBC and DEC - taking into account internal initiating events, internal and external hazards) with exclusion of malicious acts and earthquakes

** With implementation of all reasonable modifications and interventions advances shall be made towards the value of 10^{-6}/ry

5. Implementation on Articles 8a-8c

The defence in depth (DiD) concept is implemented in the Hungarian regulatory framework. Requirements for nuclear facilities are in the Government Decree 118/2011 (VII. 11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities.

Latest version with only minor modification in the text regarding defence in depth concept of the decree came into force on 11April 2018.

Eleven Annexes are connected to the Govt. Decree:

1. Nuclear safety authority procedures of nuclear facilities
2. Management systems of nuclear facilities
3. Design requirements for operating nuclear power plants

3a. Design requirements for new nuclear power plant units

4. Operation of nuclear power plants
5. Design and operation of research reactors
6. Interim storage of spent nuclear fuel
7. Site survey and assessment of nuclear facilities
8. Decommissioning of nuclear facilities
9. Requirements for the design and construction period of a new nuclear facility
10. Nuclear Safety Code definitions

Definitions in Annex 10 are the following:

“Large release: A radioactive release where off-site protective measures cannot be limited in space and time.

Early release: Radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction.

Reasonably achievable: Such a degree of actions that considers the present standards of science and technology, while graded to the severity of the different risks and undesirable consequences, which is determined by the authority based on the proposal of the licensee.

Defence In Depth: Multi-level defence, which is a hierarchically structured system of engineering solutions, nuclear safety principles and measures which guarantee the expected level of nuclear safety. On the physical level an important part of this system is the system of multiple barriers”.

Nuclear Safety Code issued in July, 2011 already contained definition of the different levels of DiD. In the spring of 2012 amendments were introduced to the Code, which determined the DBC and DEC plant states for the existing and new reactors with some differences, and additional requirements on how to apply the DiD levels in the design. This was based on the draft versions of updated reference levels for the existing power plants and the safety objectives for new power reactors some later published by the WENRA.

Article (7) of the above mentioned Government Decree declare that the release of radioactive materials into the environment shall be prevented by the application of defence in depth, and it shall be ensured that failures or the combination of failures resulting in accidents resulting in significant radioactive material discharges may only occur with adequately low probability.

The five levels of defence in depth are the following:

a) prevention of deviations from normal operational conditions and faulty actuations;

b) detection of abnormal operating conditions and prevention that the anticipated operational occurrences become design basis accidents;

c) management of design basis accidents according to pre-determined procedures;

d) termination of accident and severe accident processes and mitigation of their consequences;

e) in the case of a significant release of radioactive materials, mitigation of radiological consequences;

The most important components are the design solutions applying the appropriate safety margins, implementation and operation to a high standard; application of regulatory, limiting and protection systems and testing and monitoring solutions as well as documents regulating operation; safety systems, instructions and trainings; application of supplementary measures and accident management guidelines as well as organisation of drills; and preparation for carrying out nuclear emergency response activities on and off the site.

The independence of the levels of defence in depth shall be ensured to the reasonably achievable extent.

There are different requirements for operating and new nuclear power plant units. Volume 3 of Annex of Govt. Decree 118/2011 (VII. 11.) is dealing with the design requirements for operating nuclear power plants, whereas Volume 3a of the Annex is for the design requirements for new nuclear power plant units.

Objectives are defined to the 5 levels of DiD, but these are not different for the existing and the new NPP units to be constructed. At the same time, the associated plant states (operational states and accident conditions) are defined differently for existing and new NPP units.

It is declared for operating nuclear power plants in Annex 3 that multiple physical barriers shall be applied during the design to prevent uncontrolled release of radioactive materials to the environment and these barriers shall be protected. The design of the barriers shall be conservative, their implementation shall be of the highest standards to ensure that the probability of failures and deviations from normal operating conditions is as low as reasonably achievable, DBC4 and DEC conditions will be prevented to a reasonably achievable level and no cliff edge effect can arise.

It shall be as far as reasonably achievable ensured that events that challenge the integrity of the barriers are prevented, and the endangering circumstances are tolerated; concurring failure of more than one barrier is avoided; a barrier shall not fail due to the failure of another barrier or another system component; the detrimental consequences of human errors during operations or maintenance are avoided.

Table 3. contains the DiD levels for existing NPP units, the associated objectives, plant state definitions, their abbreviations in relation to the design basis and the radiation acceptance criteria set by the Code.

Table 3.: Defense-in-depth levels, the associated safety objectives and radiological requirements for existing units

Level of DiD	Objective	Associated plant condition/accident categories	Associated design category	Radiological acceptance criteria
1	Prevention of abnormal operation and failures	Normal operation	DBC1	Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year)
2	Control of abnormal operation and detection of failures	Anticipated operational occurrences	DBC2	Dose for the reference group of the population shall not exceed the value of the dose constraint (90 µSv/year) Initiating events resulting in DBC2 shall not cause doses exceeding 1 mSv/event outside the controlled area of the nuclear power plant, within its operational areas authorized for human staying
3	Control of accidents within the design basis	Design basis Accidents	DBC4	1., Dose for the reference group of the population is not exceeded 5 mSv/event 2., Initiating events resulting in DBC4 may not cause a dose exceeding 10 mSv effective dose or 100 mGy dose for the thyroid outside the controlled area of the nuclear power plant and within its operational areas authorized for human presence
4	Control of severe conditions, including prevention of accident progression and mitigation of consequences of severe accidents	Complex accidents	DEC1	Radioactive releases shall be minimized to the reasonably achievable extent.
		Severe accidents	DEC2	The release of radioactive materials shall be limited both in time and quantity
5	Mitigation of radiological consequences of significant releases of radioactive materials	Very severe accidents		Shall be practically eliminated

Additional technical requirements are also given in the Code. These are typically not numerical criteria, but result oriented, requiring adequate justification of the design by analysis:

- The design shall confirm by deterministic safety analyses that the initiating events resulting in DBC2 operating conditions will not lead to the loss of function of any barrier even with the assumption of a single failure.
- The integrity of the reactor pressure vessel against brittle fracture shall be maintained by ensuring that the realistic actual transition temperature of the critical elements of the pressure vessel is less than its maximal critical transition temperature derived from proper analysis of initiating events resulting DBC1 to 4 conditions i.e. discontinuities of material in the structure may not increase during scenarios resulting in DBC2-4 operating conditions.

- Following initiating events resulting in DBC2 to 4 plant states the control and safety devices controlling reactivity, the nuclear fuel assemblies, as well as the structural elements of the nuclear reactor shall not be damaged or deformed to such extent that the movement of control and safety devices to terminate the fission chain reaction becomes impossible.
- Following initiating events resulting in DBC2 to 4 plant states the nuclear fuel assemblies, the primary circuit of the nuclear reactor, and the connected systems shall remain in such a condition that the short- and long-term cooling and management of the irradiated nuclear fuel can be ensured, furthermore the systems necessary for heat removal shall be able to perform their function both on short- and long-term.
- For events resulting in DBC2 state the criteria ensuring the integrity of the fuel rods shall be defined during design by defining limits for the temperature of the nuclear fuel, the critical heat flux and the temperature of the cladding. For DBC4 design basis accidents, to fulfil criteria for long-term cooling and fuel management, the acceptable maximum degree and type of fuel damage shall be determined.
- To perform a safety function, criteria for maximum pressure, maximum and minimum temperatures, thermal and pressure transients, degradation and stresses depending on the temperature range shall be defined for systems, structures and components confining radioactive releases or performing retaining physical barrier functions during their entire lifetime.
- To fulfil nuclear safety requirements criteria shall be defined for the temperature, pressure and leakage rate of the containment throughout its lifetime.

It is stated in Annex 3a for new nuclear facilities that in addition to the requirements under Article 7, supplementary requirements shall be met during the application of the five levels of defence in depth. It shall be ensured with independent protection levels that possible failures and abnormal operation can be detected, compensated and managed. During design, multiple physical barriers shall be applied for preventing an uncontrolled release of radioactive materials into the environment.

In order to apply the defence in depth principle, the following four physical barriers shall be distinguished:

a) fuel matrix;

b) fuel element cladding;

c) boundary of the primary circuit of the reactor;

d) containment system.

The protection of the barriers shall be ensured. The fulfilment of the safety functions and the acceptance criteria shall be ensured by design solutions even in the case of damage to any level of protection.

Five levels of defence in depth are defined, from which level 3 is split into 2 sub-levels (3a and 3b). Table 4. gives an overview of the levels of defence in depth for new nuclear power plants with the objectives, means to be applied, the radiological consequences and the relevant operating condition.

Table 4. The levels of defence in depth for new nuclear power plants

A	B	C	D	E
Level of defence in depth	**Objective**	**Means to be applied**	**Radiological consequences**	**Relevant operating condition**
1.	Deviations from the normal operating condition and prevention of failures	Conservative design, implementation and operation to a high standard; maintaining the main operating parameters between the prescribed limits	No off-site radiological effects exceeding the regulatory limits	Normal operation (DBC1)
2.	Management of deviations from the normal operating condition and failures	Control and safety protection systems; other surveillance methods		Anticipated operational occurrences (DBC2)
3. 3.a	Management of design basis accidents in order to limit radioactive releases and to prevent fuel melting	Safety systems, emergency operating procedures	No or only minimum off-site radiological effects	Design basis accident (DBC3-4)
3.b		Added safety features for the elimination of complex accidents, emergency operating procedures, on-site emergency response measures		Complex accidents (Postulation of multiple failures) (DEC1)
4.	A practical exclusion of large or early releases, management of accidents involving a fuel melting in order to limit off-site releases	Supplementary safety features to limit fuel melting, accident management guidelines, on-site emergency response measures	An off-site radiological effect may warrant the introduction of protective measures limited in space and time for the population	Severe accident (DEC2)
5.	Mitigation of radiological consequences of a significant release of radioactive materials	On- and off-site emergency response measures; intervention levels	An off-site radiological effect warrants protective measures for the population	Very severe accident

Prevention of events jeopardising the integrity of barriers and toleration of hazard factors, avoidance of the failure of more than one barrier at the same time and avoidance of the failure of one barrier as a result of the defect of another barrier or other system components shall be ensured to the extent reasonably practicable.

Design of the barriers shall be conservative, and they shall be implemented to high quality norms in order for the possibility of failures and deviations from the normal operating condition to be as low as reasonably achievable; for DBC4 and DEC conditions to be able to be excluded at a reasonably achievable level, and for the cliff edge effect to be avoided.

The above regulatory definitions are translated into requirements the following way. For existing NPPs a radioactive release where off-site protective measures cannot be limited in space and time or releases that require urgent protective measures outside of the site but there is not enough time to implement them practically means:

- Containment failure or severe fuel damage in the spent fuel pool has been occurred,
  and
  - Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor;
    or
  - There is a need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor;
    or
  - There is a need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor.

For new NPPs the regulatory definition is transposed into the requirements as:

- Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor;
  or
- There is a need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor;
  or
- There is a need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor;
  or
- There is a need for any long-term restriction on food consumption.

There is a requirement, that a situation with any of the above consequences shall be practically eliminated and the “Limited Environmental Impact” criteria shall be met.

For existing NPPs the following external hazards or hazard magnitudes can be screened out from the scope of design basis:

- hazards resulting from external human activities typical of the site, if the frequency of the occurrence is less than 10^{-7}/year, or if the hazard may only occur at such a distance, that it can be demonstrated that it is not able to cause an effect on the nuclear power plant unit; and
- external hazards of natural origin with magnitudes below 10^{-4}/year occurrence frequency, or external effects of natural origin for which it can be demonstrated that they are not capable of causing any damage to the systems important to safety of the plant.

The screening conditions for external hazards for new NPPs are:

- hazards resulting from external human activities typical of the site, if the frequency of occurrence is less than 10^{-7}/ry, or if the hazard may only occur at a such a distance, that it can be demonstrated that it is not able to cause an effect on the nuclear power plant unit; and

- hazard magnitudes of natural origin, with a recurrence frequency below 10^{-5}/ry, or external effects of natural origin for which it can be demonstrated that they are not capable of causing any damage to the systems important to safety of the plant.

External events outside of the scope of the design basis are requested to be assessed above the recurrence frequency of 10^{-7}/ry both for new and existing NPPs as a part of the probabilistic safety assessment (PSA). External events are within the scope of all PSA requirements both for the existing and for the new plants, therefore the 10^{-4}/ry CDF and 10^{-5}/ry LERF criteria for existing NPPs and 10^{-5}/ry CDF and 10^{-6}/ry LERF criteria for new NPPs shall be demonstrated with external hazards included. The external hazard assessment shall be extended to all realistic combinations of these hazards, as well.

Although it is required by the Code to consider, up to date, no relevant combination of these hazards were identified, however the external hazard assessment in general is under re-view and the re-assessment is ongoing right now.

It also should be noted that as new information occurs due to further site studies and/or the development of science and technology, the list of external hazards that need to be taken into account in the analysis can be and shall be extended.

6. Conclusions

Legal and regulatory framework for nuclear safety in Hungary was presented in the report, main steps of safety upgrades performed in Paks NPP are described. Implemented safety improvements of Paks NPP before as well as after the accident at Fukushima are presented. Experiences and findings regarding existing and planned nuclear reactors are summarized.

Requirements of Article 8a to 8c of Directive 2014/87/EURATOM are implemented in the Hungarian regulation.

- Introductory text of paragraph 1. of Article 8a is covered by the "Basic principles" of the Act on Atomic Energy and the first 3 bullets of additional issues related general safety approach taken from the 118/2011 Governmental decree; points (a) and (b) are covered by safety requirements in the Nuclear Safety Code.
- Paragraph 2. of Article 8a is covered in general at policy level, differentiation between new and existing nuclear facilities given by Volume 3 (contains requirements for the existing NPPs) and 3a (contains requirements for the new NPPs) of the Code.
- The points (a) to (e) of paragraph 1. of Article 8b are basically covered by the "Basic principles" of the Act, however the detailed requirements of the protection against natural and man-made hazards, as well as the specific requirements about the five levels of defence in depth are given in the Governmental decree, especially in the design related volumes of the Code. The point (f) is covered in volume 2 of the Code (Management system of nuclear facilities) and in volumes containing requirements to operation of these facilities.
- The introductory text of paragraph 2. of Article 8b is covered in general at policy level, while the specific requirements – points (a) to (d) of this paragraph – are covered in Volume 2 of the Code.
- Basic principles in relation to Article 8c are covered in the Act by requiring the fulfilment of all safety requirements. The detailed requirements for demonstrating the safety in order for obtaining construction and operational licenses, as well as the requirements for periodic safety reassessment (including the content, criteria, methodology, time-frame, etc.) are set in the Volumes 1, 3, 3a and 5 of the Code.

Safety upgrades have been continuously implemented. During Periodical Safety Reviews and stress tests performed after the Fukushima accident, several opportunities for safety improvements were found, from which several have been implemented.

Concept of Design Extension Conditions (DEC) has been integrated in all respective regulatory frameworks and is clearly considered as an important concept for identification of further safety improvements.

A new method for the determination and application of release criteria has been developed, in line with the international requirements. Guideline for application of the release criteria was elaborated based on this method and is in the process of publication by the Hungarian Atomic Energy Authority.

A full scope probabilistic safety assessment (PSA), including internal initiating events, internal hazards, external hazards in all operating states is required by law both for operating and new NPPs including their spent fuel pools as well. PSA is updated on a yearly basis and changes are introduced into the model with the annual version update.

Effects of safety improvements were quantified by PSA. Level 1 PSA is continuously developed and extended. Value of core-damage frequency was calculated and sensitivity and uncertainty analyses were performed. In order to determine the risk of a large radioactive release, a Level 2 PSA containing all formerly analyzed operational states and initiating events was elaborated. Level 3 PSA is not required for any licensing activity at the moment, however, requirements and methodologies are under research and development.

Based on the experience gained from deterministic incident analyses, Level 1 and 2 probabilistic safety analyses, severe accident analyses and on the summarized evaluation of all results, recommendations were made for safety improvement modifications and further complex analyses. Periodic safety reviews (PSR) seem to be an efficient way to identify the weak points in technology.

Continuous evaluation of operating experiences is also an important driver for safety improvements. As a consequence of the implemented measures, the safety of the units was further increased.

Safety upgrade program has been mostly driven by the licensee, with identification of the concepts, technical solutions and the schedules for implementation.

Safety upgrades require substantial human resources, not only for implementation (from the licensee) but also for ensuring clear and stable regulation. Legislation must be well defined and easy to understand, significant human resources are necessary, especially in countries with less nuclear facilities. Legislation only available in Hungarian could bring a potential challenge for international vendors, especially in the cases with very specific and/or complex regulatory requirements and small number of affected facilities. However, it might cause additional uncertainties.

7. References

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[4] Hungary, Peer review country report of Stress tests performed on European nuclear power plants http://www.ensreg.eu/sites/default/files/Country/Report/HU/Final2.pdf

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