

Experiences in Implementing Safety Improvements at Existing Nuclear Power Plants

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EXPERIENCES IN IMPLEMENTING
SAFETY IMPROVEMENTS AT EXISTING
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NUCLEAR POWER PLANTS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2020

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FOREWORD

In response to ongoing Member State interest, in 2017 the IAEA began developing a publication summarizing how regulatory bodies, licensees, technical support organizations (TSOs) and designers have identified and implemented safety improvements at existing nuclear power plants. From 2017 to 2019, a team of technical consultants from 22 Member States identified relevant experiences and approaches to making these safety improvements. These are presented here to facilitate the exchange of information on the topic among interested Member States.

This publication describes a variety of technical approaches taken to assessing safety at these facilities and to implementing safety improvements through various processes. It provides valuable information on the continuous evaluation of nuclear safety, which can be useful in meeting obligations under Articles 6 and 14 of the Convention on Nuclear Safety and to the comprehensive and systematic safety assessments and safety improvements adopted in the second principle of the Vienna Declaration on Nuclear Safety.

The IAEA wishes to thank all contributions to this publication. The IAEA officer responsible for the preparation of this TECDOC was C. Toth of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

Since the beginning of the use of nuclear energy, new information (based on operating experience, research, new regulatory approaches, etc.) has been considered on an ongoing basis in the interest of identifying safety improvements at existing nuclear power plants.

Safety is not a static concept, and States respond to new information with regulatory, design, and operational changes at nuclear power plants. For example, through the use of periodic safety reviews (PSRs) or other equivalent arrangements, many Member States have identified safety issues necessitating new safety measures and plant modifications. Member States respond to operating experience (e.g. issues identified with electrical power, containment sumps) and address insights gained from near-misses and accidents (e.g. severe accident management, emergency procedures, focus on defence in depth). In addition, risk insights from probabilistic safety analysis (PSA) or other related tools are used to identify key contributors to plant risk and implement meaningful safety improvements (e.g. improvements to fire safety, reactor coolant pump seals, seismic strengthening). These activities have yielded significant improvements with the potential for further improvements to safety at nuclear power plants, based on various factors, and safety standards have been enhanced to reflect new experiences.

Safety improvements have been implemented on an ongoing basis as well as in response to significant events. The accident at the TEPCO Fukushima Daiichi NPP in March 2011 focused Member States' safety improvement efforts on the specific issues highlighted by the accident. Individual States and international organizations responded swiftly to the accident by identifying lessons learned and initiating coordinated efforts to re-evaluate existing plant designs in the light of new information on external hazards. For example:

- IAEA took a leadership role in preparing and implementing an Action Plan on Nuclear Safety, issued in September 2011 [1]. This plan applied new insights from the accident at Fukushima Daiichi NPP to identify updates to IAEA safety standards, revisions to IAEA peer review processes, and new research on relevant topics within the member states.
- Many States conducted collective efforts to systematically evaluate nuclear power plants' response to extreme situations and verify that appropriate preventive and mitigative measures were in place to provide defence in depth for these events. These evaluations were called "stress tests" in many cases.
- Individual States developed national action plans to address lessons learned from the accident, including prompt assessment of plants' margins against flooding and seismic hazards, re-evaluation of hazards using modern standards, and implementation of mitigation strategies for events that had not been considered in establishing the plants' original design bases.

Related activities also took place under the auspices of the Convention on Nuclear Safety (CNS) peer review process, for which the IAEA serves as the secretariat. For example:

- At the end of 2012, the Contracting Parties to the CNS held an Extraordinary Meeting focused on the lessons learned from the accident at Fukushima Daiichi NPP. In the summary report of the meeting [2], among considering the other aspects, the Contracting Parties were encouraged to reinforce efforts for continuous improvement by performing periodic reassessments of safety, through PSRs or alternate methods.

In addition, a diplomatic conference was held in 2015 to explore whether revision to the Convention itself would be adopted. The result of this conference was the Vienna Declaration on Nuclear Safety[3], discussed in more detail in Section 2.1.2. The Vienna Declaration on Nuclear Safety reaffirmed the safety principles provided by the CNS and the commitment it entails to the continuous improvement of the implementation of these principles.

International organizations prepared extensive documentation regarding the progression of the accident at Fukushima Daiichi NPP, lessons learned, and actions taken. An example of this documentation is the IAEA Director General’s Report on the Fukushima Daiichi Accident [4], which resulted from the collaboration of 180 experts from 42 Member States and several international bodies. The report presented in technical detail the accident’s causes, evolution, and consequences. In addition, the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) conducted complementary technical activities and prepared reports including Five Years after the Fukushima Daiichi Accident: Nuclear Safety Improvements and Lessons Learnt [5]. This report described work undertaken by the NEA and its (at the time) 31 member countries to improve safety since the accident in 2011.

These various activities taken in response to the accident at Fukushima Daiichi NPP resulted in safety improvements that were, in many cases, a response to common principles across Member States. Several examples of safety improvements taken in many Member States include:

- Re-evaluation of and protection against more extreme external hazards that were appropriate to specific sites;
- Enhancements to nuclear power plants’ response to some newly postulated accident scenarios using on-site equipment (installed, mobile, as well as new means of providing off-site support to manage accidents);
- Revision of accident management strategies to address concurrent accidents at multiple units onsite.

The IAEA identified the need for a summary publication describing how regulatory bodies, technical support organizations (TSOs), licensees, and designers have successfully identified and implemented safety improvements at existing nuclear power plants. In 2017-2019, the IAEA assembled a technical consultancy with participants from 22 Member States to identify relevant experiences and approaches for making these safety improvements. This resulting publication describes in more detail many of the modifications highlighted above and, more generally, Member States’ strategies for identifying and implementing safety improvements at their facilities.

Furthermore, based on the 2017 safety resolution of the 61st IAEA General Conference [6], the IAEA held a technical meeting (TM) in June 2018, with participation of 21 Member States, to “Share Experience on Implementing Safety Improvements at Existing Nuclear Power Plants”. The TM consisted of three sessions, namely (i) regulatory processes driving safety improvements, (ii) key aspects underpinning safety reassessment and improvements, (iii) experiences on safety improvements at NPPs. Presentations and discussions addressed the following general topics:

- Legal systems, regulatory requests;
- International practice on comprehensive evaluation of safety;

- Important drivers and methods for safety reassessment and for identification of safety improvements;
- Vienna Declaration on Nuclear Safety and implementation of reasonably practicable or achievable safety improvements in a timely manner;
- Periodic safety reviews;
- Current standards and codes;
- Operation experience feedback and comprehensive modification programmes.

Presentations and discussions led to the general conclusion that many Member States share the objective of continuously improving safety, but they use different strategies and methods to reach this objective. Member States have effective regulatory provisions to require safety reassessments of existing NPPs and to identify reasonably practicable improvements to ensure an adequate level of plant safety. Insights from the June 2018 technical meeting on Member States' experiences with technical safety improvements were considered in preparing this TECDOC.

In the 2018 safety resolution of the 62nd IAEA General Conference [7], the Member States requested (in paragraph 3) the Agency to continue to build upon Principles for the Implementation of the Objective of the CNS and use them for defining its nuclear safety strategy and programme of work, including priorities, milestones, timelines, and performance indicators. This paragraph also acknowledged actions taken in response to the Fukushima Daiichi nuclear power plant accident, requested that the IAEA continue to build upon them and use them for refining its nuclear safety strategy and programme of work, and requested periodic reporting by the Secretariat. Further, the resolution specifically requested in paragraph 52 that the IAEA (as Secretariat) continue to facilitate information exchange between interested Member States to share experience on implementing safety improvements at existing nuclear power plants.

1.2. OBJECTIVE

The objective of this TECDOC is to provide an overview of the latest experiences of Member States on implementing safety improvements at existing nuclear power plants.

This TECDOC provides insights that may be useful to licensees in planning and implementing safety improvements at their nuclear power plants. This TECDOC may also be useful to designers, vendors, or TSOs in considering design modifications or other changes, and regulatory bodies in updating their regulatory frameworks to take account of new information.

This TECDOC presents different approaches used by Member States that participated in the development of this publication in identifying, evaluating, prioritizing, and implementing safety improvements at existing nuclear power plants. The experiences in this TECDOC support the States in conducting comprehensive and systematic safety assessments of existing nuclear power plants in line with the principle of preventing accidents and, if an accident does occur, avoiding early radioactive releases or radioactive releases large enough to require long term protective measures and actions. Furthermore, the experiences in this TECDOC are examples of approaches taken to implement reasonably practicable or achievable safety improvements in a timely manner. Many of these safety improvements are aimed at reducing risk by decreasing the frequency and effect of events within and beyond the original design basis.

This TECDOC also addresses approaches for design assessment of existing nuclear power plants in comparison to requirements issued after their original licensing and identifying related areas for safety improvement. Another focus area is the comprehensive management of physical ageing, particularly under extended operating time frames, as well as updating initiating events and accounting for state-of-the-art science and technology as time progresses. These safety improvements are often those required to maintain nuclear power plants within their current licensing basis.

The regulatory frameworks described in the TECDOC emphasize a need for reassessment of plant safety throughout plant lifetime; consequently, this publication aims at sharing experiences in implementing safety improvements for dealing with retrospective scenarios at nuclear power plants.

1.3. SCOPE AND STRUCTURE

The scope of this TECDOC includes the following areas:

- The legally binding, discretionary, and informative documents that apply both internationally and nationally when assessing existing nuclear power plant designs, safety and operation;
- A summary of national and international approaches used to identify safety improvements that are reasonably practicable (or achievable) to support the Convention on Nuclear Safety principle of preventing accidents and mitigating their consequences;
- Examples of approaches and strategies for planning and implementing identified safety improvements, including prioritization of improvements and approval by the regulatory body.

This TECDOC provide the following information:

- Background on national regulatory frameworks and international obligations, standards and guidance, drafted primarily from the perspective of regulatory bodies (Chapter 2);
- Discussion of the triggers that have led to safety improvements at existing nuclear power plants, drafted from the perspective of licensees and regulatory bodies (Chapter 3);
- Tools and methodologies used in assessing existing nuclear power plant designs in a systematic manner, drafted from the perspective of designers and licensees including severe accident prevention and mitigation (Chapter 4);
- Means of selecting, prioritizing, and implementing particular safety improvements, drafted from the perspective of licensees and regulatory bodies (Chapters 5 and 6).

Annexes to this TECDOC (Annexes I and II) provide nearly two dozen national experiences. These annexes collect experiences from member states that were involved in the development of this publication with respect to their national regulatory frameworks, including the timely implementation of reasonably practicable safety improvements (either completed or planned), as well as additional information on collective efforts and supporting documentation collected by the preparers of the TECDOC. These annexes were developed independently based on the topics of particular interest to the contributing Member State. No attempt was made to harmonize the format or content; rather, the annexes reflect the diversity of safety-focused approaches and experiences across the global nuclear community. References within the main body of the TECDOC to specific annexes are intended to provide selected examples of interest

regarding topics discussed in the TECDOC, not an exhaustive cross-reference. Inclusion of selected examples does not imply that a topic is not addressed in other annexes or considered by other participating Member States.

The main issues covered in the annexes are:

- Recent changes in the regulatory framework and legislative action;
- Measures to implement lessons learned from significant events;
- Safety-related modification programmes;
- Triggers of safety improvements;
- Approaches to demonstrate nuclear safety;
- Severe accident management;
- Addressing obligations and commitments arising from the international safety framework.

This TECDOC uses terminology in a manner consistent with the IAEA Safety Glossary [8].

2. GENERAL REGULATORY FRAMEWORK FOR EXISTING NUCLEAR POWER PLANTS AND CURRENT IAEA SAFETY STANDARDS

This chapter provides an overview of national regulatory frameworks taking into account international obligations (such as treaties and conventions) that support development and implementation of safety improvements at existing nuclear power plants. Each framework has evolved over several decades in response to internal and external factors demonstrating a commitment to continuous improvement in the overall regulatory approach. A key consideration of the regulator is the determination of whether and how new or updated requirements, policies, and guidance need to be applied to existing nuclear power plants. In this chapter, some general concepts on safety improvements and retrospective application of new or updated requirements are presented. Underlying this overview are two essential responsibilities: for the regulator within a State to establish the appropriate safety-focused framework for its situation and for the licensee to uphold safety at its facility². Therefore, the discussion provides examples that could be useful to regulatory bodies in carrying out their responsibilities and to licensees in understanding the contextual safety objectives.

Figure 1 provides a schematic of the hierarchy of the international and national frameworks for nuclear safety regulation, composed of (1) obligatory binding requirements to be followed within a Member State, (2) discretionary guidance documents that represent approved or recommended approaches to meet binding requirements, and (3) informative or supportive documents (such as this one) that elaborate on technical or regulatory topics of interest. At each level, documents and publications prepared by the IAEA, as well as other international organizations, either inform the development of the national documents or are directly applied (as mentioned, for example in part of Annex I that relates to Netherlands). Selected examples of these framework elements are presented in the subsections below.

² When the term “facility” is used in this TECDOC, it refers to existing nuclear power plants.

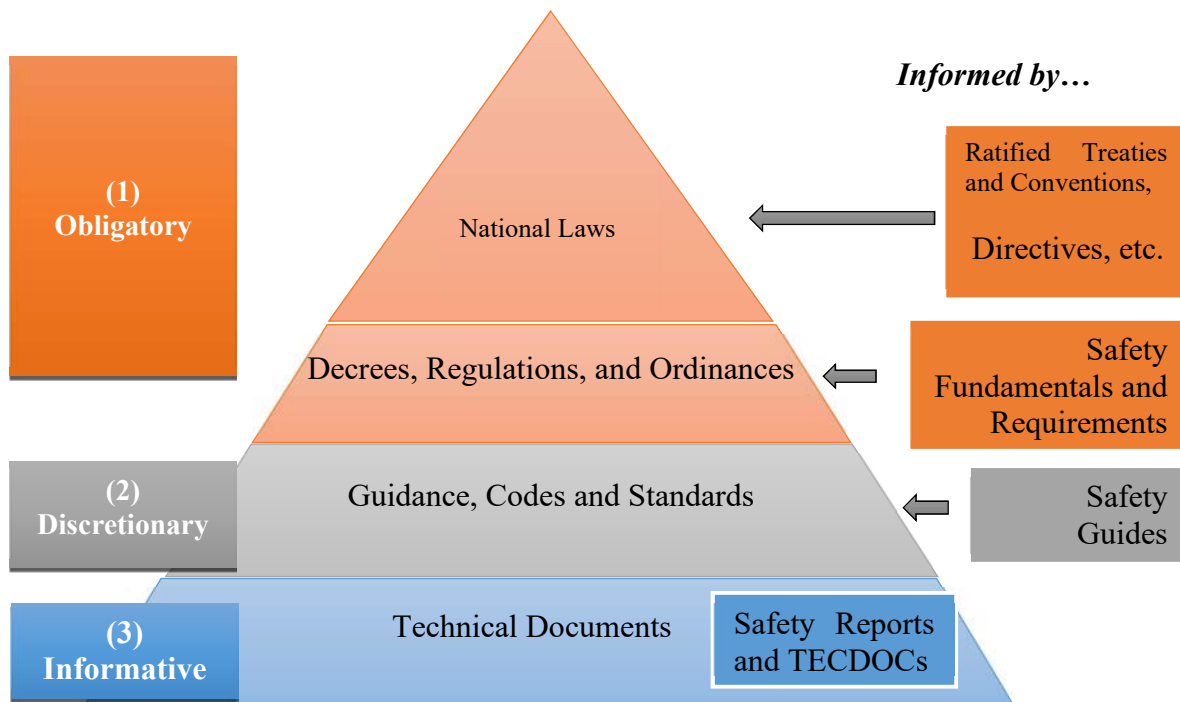


FIG. 1. Schematic of national regulatory frameworks and relationship to international documents.

2.1. INTERNATIONAL TREATIES AND AGREEMENTS

2.1.1. Convention on Nuclear Safety (CNS)

“The Convention on Nuclear Safety aims to commit participating States operating land-based civil nuclear power plants to maintain a high level of safety by establishing fundamental safety principles to which States would subscribe.” [9]

Additional information is available through the IAEA website on the CNS³, as IAEA serves as the Secretariat for the CNS.

As stated in Article 1 of the CNS 10], the objectives of the CNS are:

- (i) “to achieve and maintain a high level of nuclear safety worldwide through the enhancement of national measures and international co-operation including, where appropriate, safety-related technical co-operation;
- (ii) to establish and maintain effective defences in nuclear installations against potential radiological hazards in order to protect individuals, society and the environment from harmful effects of ionizing radiation from such installations;
- (iii) to prevent accidents with radiological consequences and to mitigate such consequences if they occur.”

Article 6 of the CNS provided for a review of nuclear power plants that were operating when the CNS came into force—specifically that they be “reviewed as soon as possible” and that

³ IAEA website on the Convention on Nuclear Safety is available in the following link < <https://www.iaea.org/topics/nuclear-safety-conventions/convention-nuclear-safety> >

“[w]hen necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation.” Furthermore, Article 14 of the CNS includes an ongoing requirement that “comprehensive and systematic safety assessments” occur throughout the life of a nuclear power plant. Such assessments “shall be well documented, subsequently updated in the light of operating experience and significant new safety information and reviewed under the authority of the regulatory body.” [10] This TECDOC presents the results of several such assessments that resulted in safety improvements at existing nuclear power plants.

In the context of the CNS, the Contracting Parties engage in a peer review every 3 years, including preparation of National Reports, written questions and answers, and a 2-week review meeting at which subjects including good practices and challenges for each Contracting Party are considered. These National Reports generally document safety improvements made since the previous review cycle. National Reports from the Seventh Review Meeting in 2017, as well as several reports from previous review meetings, are available on the IAEA website⁴. Furthermore, in response to issues identified during this peer review, Contracting Parties may consider changes to their national regulatory frameworks that could result in safety improvements at existing nuclear power plants.

2.1.2. Vienna Declaration on Nuclear Safety

In February 2015, a Diplomatic Conference of the Contracting Parties of the CNS adopted by consensus the Vienna Declaration on Nuclear Safety [3]. The Vienna Declaration on Nuclear Safety is an international political agreement under which the contracting parties to the CNS adopted the following principles to guide them, as appropriate, in the implementation of the objective of the CNS to prevent accidents with radiological consequences and mitigate such consequences if they occur:

- (1) New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, if an accident occurs, mitigating possible releases of radionuclides causing long term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long term protective measures and actions.
- (2) Comprehensive and systematic safety assessments are 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 above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.
- (3) National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the Review Meetings of the CNS.

⁴ IAEA website on the Convention of Nuclear Safety National Reports is available in the following link < <https://www.iaea.org/topics/nuclear-safety-conventions/convention-nuclear-safety/documents> >

Furthermore, the Contracting Parties agreed to continue reporting on these principles in subsequent National Reports prepared under the obligations of the CNS, including an overview of the implementation measures, planned programs, and measures for the safety improvements identified for existing nuclear installations.

Examples of application of the Vienna Declaration on Nuclear Safety by different Member States can be found in the parts of Annex I that relate to Argentina, Spain, and Switzerland.

2.2. NATIONAL REGULATORY FRAMEWORKS

Individual Member States have each developed national regulatory framework appropriate to their specific situation and in accordance with relevant international treaty obligations. These frameworks are informed by, or in some cases are changed as a direct result of, the international cooperative inputs described further in Section 2.3.

National frameworks provide both requirements and guidance for all stages of the life cycle of nuclear power plants, including specific provisions for the following topics that are discussed in more detail in the following chapters:

- Processes for considering new or updated requirements in terms of safety improvements;
- Regulator-initiated comprehensive safety assessments, when warranted;
- Renewal or extension of licenses, in consideration of PSRs or equivalent as appropriate
- Programs for ageing and obsolescence management.

National frameworks are discussed in more detail in the annexes. For example, the parts of Annex I that relate to Hungary, Japan and Spain provide descriptions of the regulatory frameworks in those States. The parts of Annex I that relate to Russian Federation and Sweden discuss the update of the regulatory framework in recent years, taking into account requirements recently established in IAEA safety standards. A more specific example is provided in the part of Annex I that relates to U.S., which discusses a policy statement on Safety Goals for the Operation of Nuclear Power Plants [11]; this policy statement articulates two qualitative safety goals for nuclear power plants and two quantitative objectives related to the risk to individuals from potential reactor accidents.

2.3. INTERNATIONAL COOPERATIVE INPUTS

2.3.1. Multinational directives

In some cases, multinational efforts result in specific binding directives that drive changes to national regulatory frameworks. For example, stress tests following the Fukushima Daiichi NPP accident contributed a 2014 revision to the EURATOM Nuclear Safety Directive [12], which included requirements to implement specific nuclear safety objectives, conduct PSRs, and perform a peer review mission looking at the regulatory framework at least every 10 years and a topical peer review every 6 years. The first topical peer review addressed ageing management of nuclear power plants and research reactors over 1MW power. Each European Union country was required to incorporate this directive into national requirements.

2.3.2. International safety standards established under the aegis of the IAEA

The international safety standards established under the aegis of the IAEA (or IAEA safety standards, in short) includes Safety Fundamentals, Safety Requirements, and Safety Guides—a similar hierarchical structure to that outlined for national regulatory frameworks. They reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. The IAEA safety standards are applicable throughout the entire lifetime of facilities and activities utilized for peaceful purposes, and to protective actions to reduce existing radiation risks. Details on the IAEA safety standards can be found on the IAEA website⁵, and summary information in this section is drawn from that location.

The regulatory bodies in many Member States use IAEA’s suite of safety standards to develop and update their national regulatory frameworks. These Member States may apply the IAEA safety standards directly, or they may assess revised IAEA publications to determine whether revisions are needed to national documents

IAEA Safety Fundamentals

As the primary publication in the IAEA Safety Standards Series, the Fundamental Safety Principles convey the basis and rationale for the safety standards for those persons at senior levels in government and regulatory bodies.

IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [13], published November 2006, includes a fundamental safety objective—to protect people and the environment from harmful effects of ionizing radiation—that applies to all circumstances that give rise to radiation risks. The safety principles are applicable, as relevant, throughout the entire lifetime of all facilities and activities, existing and new, utilized for peaceful purposes, and to protective actions to reduce existing radiation risks. They provide the basis for requirements and measures for the protection of people and the environment against radiation risks and for the safety of facilities and activities that give rise to those risks.

SF-1 [13] includes specific measures to be taken to (i) control the radiation exposure of people and the release of radioactive material to the environment; (ii) restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation; and (iii) mitigate the consequences of such events if they were to occur. With respect to existing nuclear power plants, paragraph 3.16 of SF-1 [13] addresses the repetition of safety assessments as necessary to take into account changed circumstances, operating experience, modifications, and the effects of ageing. In addition, paras 3.21 and 3.22 of SF-1 [13] address periodic reassessments throughout the lifetime of facilities.

IAEA General Safety Requirements

Safety Requirements publications establish the requirements that have to be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. The format and style of

⁵ IAEA website on the IAEA Safety Standards is available in the following link < <https://www.iaea.org/resources/safety-standards> >

the requirements facilitate their use by Member States for the establishment, in a harmonized manner, of their national regulatory framework. The applicable IAEA General Safety Requirements to the relevant safety improvements are the following:

IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [14], published February 2016, includes Requirement 6 for compliance with regulations and responsibility for safety. Paragraph 2.15A of GSR Part 1 (Rev. 1) [14] states:

“The person or organization responsible for a facility or an activity, having prime responsibility for safety, shall actively evaluate progress in science and technology as well as relevant information from the feedback of experience, in order to identify and to make⁸ those safety improvements that are considered practicable.”

⁸ Making safety improvements may require authorization by or notification of the regulatory body.”

Paragraph 4.39A of GSR Part 1 [14] states:

“[C]omprehensive safety reviews [such as periodic safety reviews for nuclear power plants] are submitted to the regulatory body for assessment or are made available to the regulatory body. The regulatory body shall ensure that any reasonably practicable safety improvements identified in the reviews are implemented in a timely manner.”

IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [15], published June 2016, establishes requirements for establishing, sustaining and continuously improving leadership and management for safety and an integrated management system. Its scope of application includes licensees and regulatory bodies. A sound management system provides for a strong safety culture, regular assessment of performance, and the application of lessons from experience. Many of the ongoing licensee and regulatory activities discussed in this publication are required or supported by management systems that have been developed consistent with GSR Part 2.

IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [16], published February 2016, presents requirements for safety assessments, particularly focused on defence in depth, quantitative analyses, and the application of a graded approach to assessments depending on their scope. As stated in para. 1.7 of GSR Part 4 (Rev. 1) [16]:

“Safety assessment plays an important role throughout the lifetime of the facility or activity whenever decisions on safety issues are made by the designers, the constructors, the manufacturers, the operating organization or the regulatory body.”

The later chapters of this TECDOC describe in more detail how these safety issues are identified and decisions for safety improvements are made. Revisions to GSR Part 4 (Rev. 1) [16] of particular relevance to the safety improvement described in this TECDOC include consideration of external events and multi-unit sites in accident scenarios. (“Facility” here includes all nuclear facilities, not just NPPs).

IAEA Specific Safety Requirements

The two publications described in this section provide key requirements for design of nuclear power plants, as well as the commissioning and the operation of nuclear power plants. The design requirements, while they might not be applied to the full extent or systematically to existing nuclear power plants, and the operational requirements are expected to be valuable information sources for regulatory bodies and licensees in identifying areas for improvement at existing nuclear power plants. The applicable IAEA Specific Safety Requirements to the relevant safety improvements are the following:

IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [17], published March 2016, states in para. 1.3:

“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.”

Furthermore, Requirement 20 of SSR-2/1 (Rev. 1) [17] addresses the derivation of design extension conditions and states that these:

“shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”

IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [18], published February 2016, also addresses the fundamental safety objective. Paragraph 3.2(e) includes as one of the activities of the management system:

“Review activities, which include monitoring and assessing the performance of the operating functions and supporting functions on a regular basis. The purpose of monitoring is: to verify compliance with the objectives for safe operation of the plant; to reveal deviations, deficiencies and equipment failures; and to provide information for the purpose of taking timely corrective actions and making improvements. Reviewing functions shall also include review of the overall safety performance of the organization to assess the effectiveness of management for safety and to identify opportunities for improvement. In addition, a safety review of the plant shall be performed periodically, including design aspects, to ensure that the plant is operated in conformance with the approved design and safety analysis report, and to identify possible safety improvements.”

Requirement 16 SSR 2/2 (Rev. 1) [18] specifically addresses the programme for long term operation. In particular, paragraph 4.53 states:

“The justification for long term operation shall be prepared on the basis of the results of a safety assessment, with due consideration of the ageing of structures, systems and components. The justification for long term operation shall utilize the results of periodic safety review and shall be submitted to the regulatory body, as required, for approval on the basis of an analysis of the ageing management programme, to ensure the safety of the plant throughout its extended operating lifetime.”

SSR 2/2 [18] also addresses PSRs, stating in Requirement 12 that:

“Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant’s operating lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.”

Additionally, para. 4.44 of SSR 2/2 [18] states that:

“Safety reviews such as periodic safety reviews or safety assessments under alternative arrangements shall be carried out throughout the lifetime of the plant, at regular intervals and as frequently as necessary (typically no less frequently than once in ten years). Safety reviews shall address, in an appropriate manner: the consequences of the cumulative effects of plant ageing and plant modification; equipment requalification; operating experience, including national and international operating experience; current national and international standards; technical developments; organizational and management issues; and site related aspects. Safety reviews shall be aimed at ensuring a high level of safety throughout the operating lifetime of the plant.”

Para. 4.47 of SSR-2/2 (Rev. 1) [18] states:

“On the basis of the results of the systematic safety assessment, the operating organization shall implement any necessary corrective actions and reasonably practicable modifications for compliance with applicable standards with the aim of enhancing the safety of the plant by further reducing the likelihood and the potential consequences of accidents.”

Obsolescence of equipment (technological obsolescence) as well as conceptual obsolescence is also a consideration as nuclear power plants become older, separate from ageing of specific equipment in place. Therefore, regulatory bodies commonly consider this point in their regulatory approaches as well.

Existing nuclear power plants may have been designed using earlier requirements or for example the requirements of IAEA Safety Standards Series No. NS-R-1, “Safety of Nuclear Power Plants: Design, [19]”, which was superseded by IAEA Safety Standards Series No. SSR-2/1, Safety of Nuclear Power Plants: Design and then by SSR-2/1 (Rev. 1 [17]. In evaluating the differences between older and newer requirements and identifying potential safety improvements, regulatory bodies may specify the use of NS-R-1 as a comparison if NS-R-1 was used for the initial design of the plant.

IAEA Safety Guides related to safety improvements

Safety Guides provide recommendations on how to comply with the safety requirements, indicating an international consensus on the measures recommended. The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. Several Safety Guides are directly applicable to States' consideration of adopting safety improvements at existing nuclear power plants, as noted below:

- IAEA Safety Standards Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [20]; Published April 2013, this Safety Guide provides recommendations on the conduct of a PSR for an existing nuclear power plant. Global assessment of safety may lead to identifying and resolving safety issues, including safety improvements. Using PSRs or equivalent approaches to identify safety improvements at existing nuclear power plants is described in Section 3.1.2 of this TECDOC.
- IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [21]; Published November 2018, this Safety Guide provides recommendations to operating organizations on implementing and improving ageing management and on developing a programme for safe long term operation of nuclear power plants, including both physical and non-physical measures. Section 3.1.4 of this TECDOC discusses ageing management as a source of safety improvements at existing nuclear power plants.
- IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [22]; Published 2019, this Safety Guide provides recommendations on accident management programmes aimed at prevent and to mitigate the consequences of severe accidents (core melt).
- IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [23]; Published 2019, this Safety Guide provides recommendations and guidance for designers, operating organizations, regulatory bodies and TSOs on performing deterministic safety analysis (DSA) that applies to nuclear power plants. It also provides recommendations on the use of DSA in demonstrating or assessing compliance with regulatory requirements and identifying possible enhancements of safety and reliability.
- IAEA Safety Standards Series Nos SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [24], and IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants [25]; Published in 2010, these Safety Guides address the necessary technical features of a Level 1 and 2 PSA for nuclear power plants in relation to its application, with emphasis on procedural steps and the essential elements of the PSA. Using PSA to identify design improvements or modifications to operational procedures could reduce the probabilities of severe accidents or mitigate their consequences.

There are several Safety Guides that are relevant for the safety re-assessment of NPPs and for the consideration for improvements. In general, the whole suite of Safety Guides for NPP design considered as far as they already have been revised or newly developed to implement recommendations to SSR 2/1 (Rev.1.) [17]. Additional Safety Guides may be found on the IAEA website, including recommendations for evaluating internal and external hazards.

This list is not intended to provide a complete list of relevant safety standards. Additional Safety Guides related to design of nuclear power plants are under revision. These Safety Guides will include more detailed recommendations for plant designs that meet the requirements of SSR-2/1 (Rev. 1) [17]. Application of these IAEA publications in the countries that were involved in the development of this TECDOC is discussed in more detail in the annexes. For example, the part of Annex I that relate to Finland and Annex II. show how IAEA safety requirements were transposed into obligatory legal documents, and how IAEA Safety Guides and TECDOCs were transposed into discretionary regulatory guides.

2.3.3. Other international standards and guidance

Many organizations have developed collaborative and consensus standards that support regulatory bodies, licensees, designers, and TSOs in identifying and implementing safety improvements. For example, technical standards organizations have produced standards on instrumentation and controls, engineering design, and PSA that have supported identification and implementation of safety improvements. In addition, international and regional organizations such as the NEA, the Western European Nuclear Regulators Association (WENRA), the World Association of Nuclear Operators (WANO) and Institute of Nuclear Power Operations (INPO), and others have developed relevant documents and publications that are inputs to national regulatory frameworks and/or licensee practices.

For example, detailed objectives for topics including operations, design, PSRs, and protection against hazards are included in the WENRA Safety Reference Levels for Existing Reactors [26].

2.3.4. Technical documents and publications

Detailed technical documents and publications (including this one) provide a broad underpinning for the regulatory framework in Member States. These technical documents and publications include research reports, “lessons learned” reports, surveys of specific technical topics, and many more types of documents and may be produced by national bodies or any of the international organizations referenced above. In general, these reports are not specifically endorsed by regulatory bodies as acceptable or preferred approaches. They do, however, provide useful information in identifying and implementing safety improvements. Specific examples are not presented here given the variety of topics and scopes; however, IAEA’s TECDOC website⁶, the U.S. NRC’s NUREG website⁷, and the NEA Publication website⁸ are examples of information sources.

2.4. INITIATORS OF NATIONAL FRAMEWORK CHANGES

The frameworks described above have evolved in response to multiple external triggers and regulatory bodies’ identification of improvements. Some examples are highlighted below:

- **Experience feedback**, including identification of generic insights through inspection, and comprehensive safety assessments such as PSA, in the frame of PSRs, stress tests—

⁶Information on the IAEA TECDOCs is available in the following link < <https://www.iaea.org/publications/search/type/tecdoc-series>>

⁷ Information on the NUREG-series Publication is available in the following link < <https://www.nrc.gov/reading-rm/doc-collections/nuregs/>>

⁸ Information on the NEA publications and reports is available in the following link <<https://www.oecd-nea.org/pub/>>

for example, requirements and standardized approaches for severe accident management that arose from insights gained from nuclear power plant accidents in the 1970s and 1980s.

- **Lessons learned from events (especially from significant events⁹)**—for example, new requirements in many Member States associated with loss of safety functions after extreme external events.
- **Insights from scientific or technological research** including experiments that could identify phenomena and areas inadequately or not currently addressed in regulation, or revisions that might be needed—for example, development of new emergency core cooling system design requirements to address fuel performance research or behaviour of different materials under some specific accident conditions.
- **Updated licensing framework**—for example, development in some Member States of revised licensing processes for new reactors.
- **Technological improvements**—for example, regulations have been revised to address obsolescence or improved equipment reliability. Some Member States have developed requirements or policies on safety features that may be appropriate in new/advanced reactors, as well as encouragement to develop new technology to further enhance level of safety.
- **Lessons learned from licensing**—for example, insights gained when licensing a new type of nuclear power plant can identify a need for changes in national requirements, gaps in the regulatory framework, or an outdated requirement format.
- **International collaborations and peer reviews**—for example, European regulators participating in WENRA update their regulations to incorporate the revised Safety Reference Levels (2014) referenced above, The European Union and some neighbouring countries conduct topical peer reviews every 6 years under the EURATOM Nuclear Safety Directive, and many Member States have updated their regulatory frameworks in response to peer reviews such as IRRS.
- **Legislative changes and public interest**—legislative changes and public interest may lead regulatory bodies to require additional safety improvements at nuclear power plants.

A significant programme of updates of Safety Guides (that are subsequently being incorporated in the regulatory frameworks in individual Member States⁹) is that undertaken by IAEA following the accident at the Fukushima Daiichi NPP. As described in the preface of SSR-2/2 (Rev. 1) [18], the IAEA's review covered, among other topics, regulatory structure, emergency preparedness and response, and 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

⁹ There is no universally agreed definition of what constitutes a significant event. Clearly, the accidents at Three Mile Island NPP, Chernobyl NPP, and Fukushima Daiichi NPP would be considered significant events given the severity of the accidents and the magnitude of the international response that followed. These events triggered worldwide, coordinated programmes to draw, and implement solutions to, all the lessons that could be learned from them. However, other events that might not have led to accidents have also triggered international lesson-drawing initiatives, which in turn have resulted in strengthening of specific areas of nuclear safety at many installations. Such events are also included in the category of significant events in this document. For example, the beyond design basis earthquake that affected the Kashiwazaki-Kariwa nuclear power plant in Japan damaged many non-safety-related SSCs and led to prolonged shutdown of the seven units at the site, even though all safety-related SSCs survived undamaged and the plants were successfully brought to safe shutdown (for more detail, see Japan Annex (2)). In addition, during the electrical event at Forsmark nuclear power plant, a short circuit in the 400kV switchyard disabled half of the emergency core cooling systems and half of the information systems in the control room and demonstrated that the surviving half was also susceptible to the same failure mode.

gases, the behaviour of nuclear fuel and the safety of spent fuel storage). The Commission on Safety Standards (CSS) endorsed revisions to GSR Part 1, GSR Part 4, SSR-2/1, and SSR-2/2, and the publication of IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [27] published in 2019. In 2014, the CSS chair confirmed that a small set of amendments was needed “to strengthen the requirements and facilitate their implementation.” The CSS further highlighted that revisions are not be limited to lessons from the accident at Fukushima Daiichi NPP, but also to include other operating experiences and advancements.

The topic of design extension conditions is most relevant to this TECDOC, as these conditions are drivers of potential safety improvements at existing nuclear power plants. Requirement 20 of SSR-2/1 (Rev. 1) [17] states:

“A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions are to be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”

The primary changes introduced by SSR-2/2 (Rev. 1) [18] were in the following main areas:

- PSRs and feedback from operating experience;
- emergency preparedness;
- severe accident management;
- fire safety.

Notably, the accident management provisions in Requirement 19 of IAEA Safety Standards Series No. SSR-2/2 (Rev. 1) [18], Safety of Nuclear Power Plants: Commissioning and Operation were revised to add more specificity on the management of severe accidents. Regulators may consider these new provisions in adjusting their regulatory frameworks for accident management—both plant modifications and operational changes.

More information about severe accident management hardware updates can be found in the parts of Annex I that relate to Finland, Russian Federation, as well as in Annex II. .

2.5. IMPOSITION OF NEW OR REVISED REQUIREMENTS

As frameworks evolve, the regulator will determine whether and how to impose newly adopted or adapted requirements to existing nuclear power plants. The existing state of the plant is carefully considered and evaluated to determine the safety benefit of safety improvements resulting from the new requirement. Different specific national approaches, including different roles of regulatory bodies and licensees, to this process exist in different countries, but most weigh safety benefits, practicality of improvements, and (in some cases) costs.

Requirements and resulting improvements that add, significantly strengthen, or make independent layers of defence in depth may, in general, be more likely to be applied to existing nuclear power plants than those that provide a more marginal benefit. Key features of defence in depth include a robust design to prevent accidents, containment features to prevent and

mitigate significant radioactive release, and emergency preparedness to protect the public. The five levels of defence in depth described in SSR 2/1 (Rev. 1) [17] are summarized below:

1. Prevent deviations from normal operation and the failure of items important to safety;
2. Prevent anticipated operational occurrences at the plant from escalating to accident conditions;
3. Provide inherent and/or engineered safety features, safety systems, and procedures that are capable of (a) preventing damage to the reactor core or preventing radioactive releases requiring offsite protective actions and (b) returning the plant to a safe state;
4. Prevent the progression of accidents and mitigate the consequences of a severe accident;
5. Mitigate the radiological consequences of radioactive releases that could potentially result from accidents.

For example, the part of Annex I that relates to Germany describes in detail how active engagement in the development of IAEA Safety Standards and WENRA Reference Levels has helped the regulator to be informed about most recent developments. These insights are being used to continuously benchmark the national regulatory framework to identify possible improvements and consequently implement them.

3. GENERAL APPROACH FOR INITIATION OF SAFETY IMPROVEMENTS

3.1. INTRODUCTION

The safety of nuclear power plants is not static but is a continuous process of improvement. Member States identify and implement safety improvements throughout the life time of nuclear power plants to:

- Maintain the licensing basis of the plants, including ALARA (for example, safety improvements to address the adverse effects of ageing).
- Increase the level of plant safety beyond the current licensing basis (for example, to address safety shortcomings identified through investigation of significant events such as Fukushima Daiichi NPP).

The approach to safety improvement at any existing nuclear power plant will depend on the overall regulatory framework. For example, Member States issue licences with terms of different length and with different conditions. In some cases, authorization for an additional licence term depends on conduct of a systematic safety assessment such as a PSR. In other cases, licence changes are made on an ongoing basis separate from the duration or extension of the licence. In addition, the approach to safety improvement at any existing nuclear power plant depends on the siting of the nuclear power plants; the design type of the facility, plant age, and history; and other conceptual, technological, regulatory, and plant-specific considerations. Common to these approaches is a general strategy, which includes a number of triggers that initiate a process to identify, select, plan and implement safety improvements.

This chapter presents various triggers, including licensees' ongoing programs (e.g. ageing management, obsolescence, operating experience, benchmarking, and internal evaluations), integrated safety reviews (such as PSRs, LTO, and external evaluations), and other specific safety reviews. Some of these triggers are continuous in nature while others are periodic or occur in response to a significant event (e.g. TMI NPP, Chernobyl NPP, Fukushima Daiichi NPP). In response to these triggers, regulatory bodies and licensees may identify potential safety improvements that could further contribute to prevention or mitigation of accidents, as discussed in the following chapters.

Safety improvements strengthen the defence in depth, which is fundamental to nuclear safety. In determining what can be done to further prevent and mitigate radioactive releases, the licensees generally consider all levels of defence in depth that are within their responsibility. Licensees explore measures to enhance the robustness of different levels of defence in depth, including maintaining the independence among levels. Where improvement measures at a particular defence in depth level are determined to be not reasonably practicable in a particular case, efforts may be made to determine whether additional improvements in other levels of defence in depth could be considered in their place. The regulator may decide whether the overall remaining risk is acceptable or not for continued operation, considering developments in science and technology and current safety requirements.

Figure I-11 in the part of Annex I that relate to Finland illustrates the expected increase in safety level over the lifetime of a nuclear power plant due to continuous improvement activities performed by a licensee. An existing nuclear power plant would have been designed in accordance with the requirements, codes, and standards applicable at the time of design and construction, providing an initial level of safety that is reflected in the licensing basis for the plant. The level of safety is enhanced throughout the lifetime of the plant as a result of the

implementation of safety improvements identified by the use of the triggers mentioned in Fig. I-11 and discussed in the following sections. Those activities can conveniently be categorised in three groups: licensees' ongoing programmes (which address both maintenance of expected safety levels and licensing basis, such as through ageing management, and specific safety improvements), established mechanisms for integrated safety reviews, and specific safety reviews requested by regulatory bodies. It is understood that the use of these activities will vary among Member States, and each State will not necessarily employ each and every activity listed below.

3.2. LICENSEES' ONGOING PROGRAMMES

It is a fundamental nuclear safety principle that the prime responsibility for nuclear safety rests with the licensee. As stated in para. 3.12 of SF-1 [13]:

“Leadership in safety matters has to be demonstrated at the highest levels in an organization. Safety has to be achieved and maintained by means of an effective management system. This system has to integrate all elements of management so that requirements for safety are established and applied coherently with other requirements, including those for human performance, quality and security, and so that safety is not compromised by other requirements or demands. The management system also has to ensure the promotion of a safety culture, the regular assessment of safety performance and the application of lessons learned from experience.”

Management system requirements provide overall direction to the licensee organization for developing and implementing sound management practices and controls for the organization. An effective and well-implemented management system helps to assure the regulatory bodies that licensees will conduct their licensed activities safely.

A successful management system ensures that nuclear safety matters are not dealt with in isolation. They are rather considered in an integrated manner within the context of safety, health, environment, security, quality assurance, human-and-organizational factors, societal and economic elements. The management system will ensure the fostering of a strong safety culture, regular assessment of performance and the application of lessons learned from experience. The knowledge gained from all these activities becomes the driving force for safety improvements: nuclear power plant staff will identify adverse conditions or opportunities for improvement, develop the necessary solutions (e.g. modifications either to hardware, processes, training or procedures) and prepare plans for their timely implementation to address identified safety issues.

The triggers described in the following subsections are generally expected to be part of the overall management system. Relevant requirements are established in GSR Part 2 [15].

3.2.1. Ageing management

Paragraph 1.2 of SSG-48 [21] states:

“Ageing management for nuclear power plants is implemented to ensure that the effects of ageing will not prevent structures, systems and components (SSCs) from being able to accomplish their required safety functions throughout the lifetime of the nuclear power plant (including its decommissioning) and it takes account of changes that occur with time and use. This requires addressing both the effects of physical ageing of SSCs,

resulting in degradation of their performance characteristics, and the non-physical ageing (obsolescence) of SSCs (i.e. their becoming out of date in comparison with current knowledge, codes, standards and regulations, and technology).”

When ageing is taken into account at each stage of a plant’s lifetime, that is, during design, construction, commissioning, operation (including long term operation and extended shutdown), and decommissioning, it helps ensure that SSCs important to safety remain capable of performing their required safety functions. Effective ageing management programmes, therefore, support the safe and reliable operation of nuclear power plants.

An ageing management programme will often give rise to the need for plant modifications or modernisation programmes that also provide an excellent opportunity to enhance safety by replacing and improving aged SSCs. Examples include emergency diesel generators, reactor coolant pumps, I&C components, and obsolete components (see for example the part of Annex I that relates to Finland).

Approaches

An overview of approaches typically used by Member States to identify safety improvements using this trigger is as follows:

- Member States typically review the ageing management programme for consistency with the nine generic attributes of an effective ageing management programme listed in Table 2 of SSG-48 [21], as follows:
 - Scope of the ageing management programme based on understanding ageing
 - Preventive actions to minimize and control ageing effects
 - Detection of ageing effects
 - Monitoring and trending of ageing effects
 - Mitigation of ageing effects
 - Acceptance criteria
 - Corrective actions
 - Operating experience feedback and feedback of research and development results
 - Quality management.
- The programme includes definition of the scope of SSCs to include in the ageing management programme, identification of degradation mechanisms, establishment of monitoring programmes and inspections, and periodic review of the programme to identify new degradation mechanisms through operating experience feedback and research and development. In some Member States the programme is established as an integrated set of programmes and activities, interfacing with other existing programmes such as the maintenance, chemistry, surveillance, fitness for service, and safety analysis programmes (e.g. an integrated ageing management programme, discussed further in Section 3.3.2).
- Criteria are established to ensure that corrective actions are taken before a degradation mechanism in any of the monitored SSCs progresses to an extent that could be adverse to safety. Activities include re-assessment, re-analysis, repairs, replacements, or plant modifications.
- The effectiveness of the programme is assessed periodically and revised as needed.

- The results of the programme are documented in reports, which are then used for multiple purposes. For example, the licensee may develop appropriate strategies to control current conditions of the plant components and to inform the regulatory body as needed.
- Each licensee has its own programme for ageing management with relevant processes. These processes are not necessarily the same as those depicted in Section 5 of this TECDOC; however, these have been demonstrated to be equally effective.

See some IAEA references related to ageing management are as follows:

- IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [18]
- Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, SSG-48 [21]
- Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), IAEA Safety Reports Series No. 82 [28]
- Proactive Management of Ageing for Nuclear Power Plants, IAEA Safety Reports Series No. 62 [29]

3.2.2. Obsolescence

Obsolescence of SSCs refers to their becoming out of date in comparison with current knowledge, standards and technology. As the state of the art in component technology, regulations, codes and standards and scientific knowledge can move rapidly in comparison to the lifetime of a nuclear power plant, many SSCs might suffer from obsolescence during the plant lifetime. Therefore, management of obsolescence is a specific element of the ageing management process and is usually incorporated in the ageing management programmes of nuclear power plants. It consequently provides the same opportunity to enhance safety by enhancing and improving aged or obsolete SSCs by identifying the following:

- Components, or spare parts for components, that are no longer manufactured or are no longer available from their original manufacturers;
- Components, systems, and concepts that have to comply with applicable regulations, codes, and standards;
- Components that are no longer qualified in accordance with applicable regulations, codes, and standards;
- Components that are out of date with respect to current knowledge and scientific research.

As mentioned above, management of obsolescence is a specific element of the overall ageing management process and recommendations on the management of technical obsolescence are provided in SSG-48 [21]. In addition, SSG-25 [20] on PSRs specifically refers to management of obsolescence in its safety factor 2 dealing with the actual conditions of SSCs important to safety, and in safety factor 8 on safety performance. Additional information can be found in Ref. [28].

Approaches

The information below provides an overview of the method typically used by Member States to identify safety improvements that are needed due to obsolescence:

- Licensees systematically identify SSCs important to safety that are obsolete or are expected to become obsolete in the foreseeable future and prioritize the SSCs on the basis of their safety significance.
- Licensees then develop effective replacement solutions. Equivalencies can be established, or similar components can be used as replacements as long as the components' safety function and capacity (safety limits) are maintained or improved in accordance with their safety classification.
- In certain extreme cases, licensees may make significant modifications to address obsolescence.
- Each licensee has a maintenance programme that includes the management of obsolescence. Obsolescence is addressed by most Member States under the ageing management programme. Processes under these programmes are not necessarily the same as those depicted in Section 5; however, these have been demonstrated to be equally effective. Note that Member States may give special attention to obsolescence during implementation of a long term operation (LTO) programme.

3.2.3. Operating experience feedback

One prerequisite for the safe and reliable operation of nuclear power plants is that lessons are learned from the experience accumulated from the operation of the licensee's own nuclear power plant(s) and from other nuclear power plants. This activity is generally referred to as operating experience feedback. The purpose of this activity is to prevent accidents and other events adverse to safety by identifying and eliminating circumstances leading to failures, flaws and non-conformities, and by underlining sound procedures proven at both own and other nuclear power plants. (In this publication, response to significant events is addressed separately from operating experience feedback; see Section 3.4 on specific safety reviews.)

Operating experience feedback is monitored and assessed by all Member States in order to enhance safety. Safety-significant operational events are evaluated for the purpose of identifying the immediate and underlying causes as well as defining and implementing corrective and preventive actions.

Operating experience feedback covers all significant flaws, observations, and good practices in terms of the design, construction, technical implementation, operation, and decommissioning of nuclear power plants, and the management systems covering the said stages. Furthermore, operating experience feedback encompasses the experiences gained from the activities of key suppliers and contractors. Additionally, licensees benefit from the operating experiences accumulated at other nuclear power plants.

Approaches

To make efficient use of operating experience, the licensee will generally identify, analyse, investigate, and report events related to the operation of nuclear power plants. Operational events here refer to such developments, failures, flaws and problems that are of relevance in terms of nuclear or radiation safety. Operational failures are identified and corrected by the licensee, and causal analysis may be conducted as appropriate to identify lessons learned or corrective actions. These automatically become internal operating experience. Based on safety significance of an event or the generic applicability of the findings of a root cause analysis, the experience may be communicated more widely such that it may become external operating experience.

In addition to national databases of operational events, all Member States have access to IAEA databases such as the Incident Reporting System (IRS) and International Nuclear and Radiological Event Scale (INES). Other internationally maintained sources of operating experience are available to licensees in Member States, such as the WANO database and the owner's groups associated with a specific vendor or reactor type. An additional source is generic communications of nuclear authorities or technical bulletins from the manufacturer.

The lessons learned from an event in one nuclear power plant may be applied to other nuclear power plants with similar designs or operating approaches, and licensees can use them to avoid event repetition and improve safety. Examples of safety improvements that have been implemented in various Member States based on operating experience include:

- Improvements to containment recirculation sump performance;
- Provision of backup power capability based on multiple operating events;
- Correction of open phase electrical circuit vulnerabilities at certain facilities based on operating events;
- In-vessel melt retention or ex-vessel corium cooling capabilities;
- The addition of containment filtered venting capability.

Requirement 24 of SSR-2/2 (Rev. 1) [18] addresses feedback of operating experience, and further recommendations are provided in IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [30]

3.2.4. Comparative evaluations (benchmarking)

The nuclear industry places great importance on continuous learning. A comparative evaluation of the approaches taken by other relevant organizations (commonly referred to as benchmarking) is one of the most powerful tools employed. When used properly, a self-critical benchmarking exercise can highlight opportunities to strengthen nuclear safety and can also reveal nuclear safety weaknesses in an organization.

More specifically, benchmarking is comparing the safety performance and practices of one organization (nuclear power plant, engineering division, research and development division) with another comparable one that has demonstrated leadership in a particular area. In this way, the licensee learns how well the safety performance and practices compare between the two organizations and why some of these are recognised as leaders in the field.

The scope of benchmarking is very wide. It can be used in a comparison of focused topics (e.g. small modifications, comparison of codes, comparison of methodology) or in large scope projects such as preparation for LTO. When applied to nuclear power plants, benchmarks usually correspond in methodology to peer reviews (see Section 3.3.3). In addition, operating experience, as discussed in Section 3.2.3, can be instructive for licensees wishing to benchmark their selected approaches against activities previously undertaken by their peers.

Benchmarking is a great approach to enhance safety by learning best practices from leaders in the industry. Furthermore, regulatory bodies can use benchmarking to evaluate the effectiveness of their national frameworks, safety criteria, and requirements.

Annex II. shows an example of international benchmarking through cooperation, common analysis, and similar solutions regarding severe accident management.

3.2.5. Internal evaluations and self-assessments

Internal evaluations are a common practice in the nuclear industry, aimed at evaluating the compliance of licensee staff with internal processes, and at finding opportunities for strengthening programmes, processes, procedures, and practices at the nuclear power plant being reviewed. Internal evaluations also provide insights on how to improve these processes and procedures. These are a form of safety improvement as they can improve human performance and support a positive safety culture.

It is possible to consider internal evaluations conducted in four levels:

- Self-assessment completed by the employee or group itself;
- Peer reviews typically performed by qualified personnel in the subject matter and from another group within the same branch of the organization;
- Audits that consist of independent assessments on behalf of upper management by an internal independent team from a separate branch of the organization;
- Independent nuclear safety oversight provided on a continuing basis by a dedicated group of highly qualified independent evaluators.

Generally, evaluations and self-assessments reveal opportunities for improving operational safety practices and enhancing operational safety at the reviewed nuclear power plant. The safety improvements resulting from these triggers are more likely to be related to organizational changes, procedural adherence, or human factors, rather than safety improvements through design modifications.

Approaches

The method used by Member States to identify safety improvements using these triggers is determined by each licensee's management system.

Reports are generally issued to document the results of audits and self-assessments and may lead to improvement activities, as necessary. The regulatory body is not involved in these activities but may be interested in the reports produced.

Each licensee has processes in place to perform these activities. These processes are not necessarily the same as those depicted in Chapter 5; however, these have been demonstrated to be equally effective.

3.3. INTEGRATED SAFETY REVIEWS

3.3.1. PSRs and alternative arrangements

Many Member States require that PSRs be performed. As already explained in Section 3.1, in some States, the term of an operating license is aligned with the PSR periodicity (typically 10 years), so that the PSR is part of the process for renewal of the operating license. Furthermore, in some countries, the process for justifying and licensing the extension of the lifetime of a nuclear power plant makes use of the PSR process, typically with some adaptations and extensions of the scope, for example to re-evaluate the safety analyses of SSCs with time-limited assumptions and identify the life-limiting features of the plant so that suitable modifications, refurbishments and replacements can be made to ensure the safety of the plant

for the duration of the planned extension (see for example the part of Annex I that relates to Argentina for practical use of PSR in the frame of periodic license renewal and LTO).

In some Member States the requirement to perform PSR is included in the national legislation or is part of the regulatory system, although the scope, content and implementation methodology may vary depending on national regulations. For example, the EURATOM Nuclear Safety Directive requires a systematic re-assessment of the safety of nuclear installations every 10 years, with the objective of ensuring compliance with the current design basis and identifying further safety improvements by taking into account ageing issues, operating experience, most recent research results and developments in international standards.

PSR is a systematic, integrated review that considers both operational performance trends and design safety, assessing the cumulative effects of plant ageing and modifications, as well as site-specific and organizational aspects. The scope of the review is comprehensive, covering all relevant safety issues and all facilities and SSCs on the site covered by the operating license (including, if applicable, waste management facilities, on-site simulators) and their operation, together with the staff and the operator's organization. As all reasonably practicable improvement measures are taken, PSRs not only confirm the safety level but improve it.

When performing PSR, the initial step of the review is the agreement between the regulator and licensee with regard to the scope and timing of the review and the codes and standards that will be used (as discussed, for example, in the part of Annex I that relates to Netherlands). Typically, this agreement is compiled in a "basis document" that governs the conduct of the PSR and the regulatory review of the PSR results. The content of this document includes: the scope and methodology of the PSR, major milestones, including cut-off dates (beyond which changes to codes and standards and new information will not be considered), the safety factors to be reviewed, the structure of the documentation, and the applicable national and international standards, codes and practices.

It is also a common practice to include in the basis document the methodology for the global assessment of the findings raised during the review in view of the importance of this activity to the success of the PSR. The objective of the PSR global assessment is to arrive at a judgement of the nuclear power plant's suitability for continued operation on the basis of a balanced view of the findings from the reviews of the separate safety factors. This judgement takes into account the safety improvements considered in the global assessment as necessary (which may relate to the plant, or to the operating organization) together with any positive findings (strengths) identified in the safety factor reviews. The global assessment evaluates the impact on safety based on the findings from all the separate safety factors and so needs to be performed after completion of all the individual safety factor reviews.

The rationale for a safe continued operation until the next PSR cycle takes account of safety improvements considered in the global assessment as necessary (which may relate to the plant or to the operating organization) together with any positive findings (strengths) identified. There is a need for the regulatory body to assess the risk associated with negative findings both in the short term prior to the implementation of identified safety improvements and in the long term if the global assessment concludes that addressing some of the negative findings is not reasonable and practicable.

A method for assessing, categorizing, ranking and prioritizing safety improvements to address negative findings needs to be also established prior to performing the global assessment. The method is based on the safety significance of each proposed improvement and then applied to

all the improvements proposed within the global assessment. The approach adopted could be based on deterministic analysis, PSA, engineering judgment, and/or risk analysis or a combination thereof.

Paragraph 6.10 of SSG-25 [20] states:

“As part of the global assessment, the following matters should be examined:

- The time necessary for implementing corrective actions and/or safety improvements. Considerations need to be given to the actual benefit to safety that the corrective action will achieve and the duration of the benefit (the remaining planned lifetime of the plant). Alternatively, depending on the safety significance of the safety improvement and the remaining planned lifetime of the plant, adequate interim measures could be implemented. If a modification is necessary on the grounds of unacceptable risk, then relevant operations need to be halted until after the modification has been implemented or adequate interim measures implemented and, where required by regulations, approved by the regulatory body.
- The use of PSA to estimate the risk posed by a negative finding. Such estimates need to be included in the review for the PSA safety factor (safety factor 6). However, while PSA can provide useful insights into relative risks, help judge priorities and compare options, a decision-making process that is solely based on numerical risks is not appropriately robust or reliable and not adopted.
- The total effect of the negative findings, safety improvements and positive findings (strengths) identified in the PSR are examined using deterministic methods to ensure that the overall level of plant safety is adequate.”

Paragraph 6.11 of SSG-25 [20] states:

“The global assessment reviews the extent to which safety requirements relating to the concept of defence in depth and the fundamental safety functions (reactivity control, core cooling and the confinement of radioactive material) are fulfilled.”

Based on the findings of the PSR and considering the global assessment, the licensee develops the proposed Integrated Implementation Plan (IIP) that is submitted to the regulatory body for approval. The approach to address any improvement in plant safety level needs to be demonstrated and the remaining risk needs to be as low as reasonably practicable.

For a comprehensive PSR, SSG-25 [20] recommends 14 safety factors, grouped in five areas to use for subdivision of the review scope, as shown in Table 1.

TABLE 1. PSR AREAS AND SAFETY FACTORS [20]

| Areas | Safety Factors |
|--|---|
| Plant | Plant design |
| | Actual condition of SSCs |
| | Equipment qualification |
| | Ageing |
| Safety analysis | DSA |
| | PSA |
| | Hazard analysis |
| Performance and feedback of experience | Safety performance |
| | Use of experience from other plants and research findings |
| Management | Safety management systems and safety culture |
| | Procedures |
| | The human factor |
| | Emergency planning |
| Environment | Radiological impact on the environment |

The number of safety factors and/or their grouping may be different depending on the specific needs of the operating organization and the particulars of the nuclear facility. When the concept of safety factors or the number of the safety factors is different, the comprehensiveness of the review is ensured by other means.

It is recognized that some Member States prefer alternative arrangements to a PSR. For example, some Member States apply routine comprehensive safety assessment programmes that deal with specific safety issues, significant events, and changes in safety standards and operating practices as they arise. Such programmes can, if applied with appropriate scope, frequency, depth, and rigour, achieve the same outcomes as the PSR process.

Approaches

The method typically used by some Member States to identify safety improvements through PSRs include the following:

- Licensees will generally perform a systematic review of the facility’s design, condition, and operation in different safety areas to support continued safe operation. The safety areas adopted by Member States are typically the 14 safety factors in SSG-25 [20].
- The safety factors in SSG-25 [20] are used to conduct a comprehensive evaluation of the design and operational safety of the plant against modern codes, standards and best industry practices.
- The configuration management of the plant is reviewed and updated, as necessary. This is an important aspect for ensuring the continued safe operation of the plant in compliance with the licensing basis.
- Upon completion of the PSR, gaps between the facility’s licensing basis and updated requirements and/or modern codes and standards are identified, and their safety significance is evaluated.
- Potential safety improvements are identified to cover significant gaps.
- From the potential safety improvements identified, those that are reasonably practicable to implement are selected, since full compliance with updated requirements and/or modern codes and standards might not be practicable. The process for selecting safety improvements to be implemented is discussed further in Chapter 5.

- Some countries use PSA insights to select safety improvements, and/or to prioritize them. PSA as a tool is discussed in Chapter 4. For additional suggestions on how to prioritize, consult Chapter 5.
- PSR reports document compliance of the design, condition and operation of the facility against the current licensing basis. PSR reports also document identified gaps compared to updated requirements and/or modern codes and standards, including practicable safety improvement activities to bridge those gaps that are of importance to safety.
- The outcome of the PSR is communicated to the regulatory body, which provides feedback. The licensees revise the PSR outcome until the regulatory body is satisfied with the practicable safety improvements identified to bridge those gaps important to safety together with a time frame when these improvements need to be implemented. Remaining gaps are also justified because safety improvements to address these gaps are not practicable or do not provide a demonstrable safety benefit.

3.3.2. Long term operation, lifetime extension and licence renewal

Many nuclear power plants worldwide have been in operation for more than 30 years. Since the nuclear power plants represent significant financial investments, licensees in many Member States are giving a high priority to continuing the operation of nuclear power plants beyond the time frame originally anticipated (e.g. 30 or 40 years).

Member States adopt different approaches with respect to the duration of nuclear power plant operating licences. In some Member States, operating licences are limited in time, with the time limit in some cases aligned with the design lifetime of the plant. In this case a formal licence renewal process is needed to continue operation beyond the license duration.

In some other Member States, licences are granted without any fixed time limitation. In these Member States, operation of the installation may continue for as long as it can be demonstrated to be safe. The PSR is typically the tool used to make the necessary demonstration. Still, the notion of the plant design-lifetime is important and the PSR that would justify continued operation beyond the original design lifetime would typically include some adaptations and extensions of the scope, for example to re-evaluate the safety analyses of SSCs with time-limited assumptions and identify features of the plant that might limit its safe lifetime so that suitable modifications and replacements can be made to ensure safety for the next period of operation.

When selected, the PSR option incorporates an integrated safety review of the plant to confirm safety of ongoing operation and to identify safety improvements judged to be practicable to support the period of long term operation. This option uses the process of endorsing long term operation as an opportunity to increase safety margins beyond their current level or to reduce risk from plant operation as far as practicable. It also seeks to apply improvements in technology and methods to correspondingly improve plant safety as part of its assessment of long term operation.

There are also some member states that issue operating licenses with a time limit shorter than the plant design lifetime. These licenses are typically valid for 10 years and are aligned with the PSR periodicity, so that each PSR is part of a formal license renewal process (see for example the part of Annex I that relates to Argentina).

LTO, or lifetime extension, of a nuclear power plant is the operation beyond a pre-established time frame, which has been justified by appropriate safety assessments, with consideration given to life limiting processes. In some Member States with time-limited licenses, LTO is often referred to as license renewal.

Member States apply LTO concepts in a variety of processes adapted to the national regulatory framework. Preparations for LTO typically include demonstration of the plant's ability to operate safely for the envisaged period of lifetime extension. PSRs may be one of the elements of these programmes. Plant modifications or modernisation programmes may also be necessary for operating beyond the initially anticipated lifetime. These also provide licensees and regulatory bodies with an excellent opportunity to identify reasonably practicable improvements to the safety of the installation in line with updated standards (see for example the part of Annex I that relates to Finland).

The LTO guidance from the IAEA Refs [31–33] is used by regulatory bodies, who may also have additional provisions in their national legislation or other regulatory requirements and guidance. These could include comparing the nuclear power plant with the latest safety standards.

Significant safety improvements are often linked to the case for pursuing LTO, as the potential for return on investment during the extended operating period and the extensive preparatory works that are often necessary can be a strong motivation, for the regulator and licensee alike, for enhancing safety while continuing safe operation beyond the originally projected design life.

Approaches

The management of ageing processes of SSCs is an important element in the safety assessment that may be used to justify life extension of an operating nuclear power plant, and licensees have implemented a number of ageing management-related programmes (see also section 3.2.1).

Some Member States and international organization have developed an Integrated Ageing Management (IAM) programme, integrating other relevant programmes and processes (e.g. inspection and maintenance) and using international standards such as Refs [31–33], SR-2/2 (Rev. 1) [17] and SSG-48 [21].

Some Member States are also making use of specific programmes as described in Ref. [28] and based on Ref. [34]. This includes, for SSCs important to safety:

- A collection of proven ageing-management programmes using the same nine generic attributes from IAEA Safety Series No. SSG-48 [21] and providing more information on specific degradation mechanisms of SSCs.
- A collection of typical time-limited ageing analyses that may be used as needed to analyse identified degradation with respect to projected operational time.
- This may include equipment requalification on harsh environment conditions.

The safety justification for long term operation may take place within a broader regulatory process, such as a specific license renewal process or a periodic safety review (see Section 3 of SSG-25 [20] and SSG-48 [21]).

If the PSR process is used, the safety factor related to ageing may be expanded to include evaluation and update of the safety analyses that have time-limited assumptions and assessments of ageing affects.

The method typically used by Member States to identify safety improvements include the following:

- A comprehensive review and assessment is completed and used to demonstrate the necessary functionality of SSC for the time-extension period needed.
- The outcome may be used to justify safe operation of the plant or to justify the residual lifetime of major components during the life extension period.
- Feasibility studies and pre-conditions assessments are typically completed prior to assessing LTO.
- Licensees will generally identify SSCs that need to be assessed and determine the scope of the related assessment. They will then prepare a list of equipment to replace or modify. Replacement and/or modifications may have already been implemented as a result of ageing management, may be targeted for future implementation, or may already have been implemented based on the outcome of a recent PSR.
- New or revised safety analysis for SSCs reflecting their aged conditions may need to be performed.
- Some Member States use PSA insights for scoping and screening out SSC. PSA as a tool is discussed in Chapter 4.
- LTO has its own process that might not be the same as the process in Chapter 5; however, the process used for LTO has been demonstrated to be equally effective.
- The outcome of the LTO is communicated to the regulatory body, which provides feedback. The licensees address comments from the regulatory body to ensure that the nuclear plant continues safe operation for the life extension period and that any identified reasonably practicable safety improvements are implemented.

3.3.3. External evaluations and peer reviews

External evaluations or peer reviews are a common practice in the nuclear industry. They represent a technical exchange of experiences and practices at the working level, aimed at finding opportunities for strengthening programmes, procedures and practices at the nuclear power plant that is being reviewed.

Peer reviews are conducted by international experts under the auspices of IAEA, WANO, and other peer groups. The focus of these reviews is determined by the organization to be evaluated.

The safety of nuclear power plants depends on factors such as design; management; policies, procedures, processes and practices; personnel; accident management and emergency preparedness; and resources. These factors are assessed collectively to determine the nuclear power plant's safety performance.

A review mission can also be focused on a single element of the organization, such as practices used in outage management.

Most of the evaluations have a strong technical focus (e.g. emergency preparedness and response, accident management, the use of PSA to assist with safety management), however the expert reviewers also identify organizational issues such as safety culture challenges.

Generally, the evaluations provide the host country and the relevant institutions—plant and utility management, the regulator, and other governmental authorities—with an objective assessment of the operational safety at the reviewed nuclear power plant. However, the results of some evaluations are tightly controlled in the interests of ensuring very open discussions between the reviewers and utility management and staff in order to obtain the best possible evaluation of nuclear power plant operation.

Approaches

External evaluations are undertaken following an established and systematic process described by each external organization. The evaluations may result in observations, suggestions, and recommendations for improvement that may lead to the need to implement safety improvement measures.

The method used is determined by the external organization and might not be the same as in the process presented in Chapter 5.

3.4. SPECIFIC SAFETY REVIEWS AFTER SIGNIFICANT EVENTS

Regulatory bodies have various means for responding to a significant event, including meetings with plant management, letters, technical bulletins, enforcement activities and licence conditions. An initial regulatory reaction to an event may be a request for an immediate review of the existing safety case for a nuclear power plant and its capability to respond successfully to similar events. After a thorough review of the event and the safety case, a regulatory body may require the implementation of solutions to strengthen the resistance of a nuclear power plant to a similar event.

For significant events, the regulatory body may request the industry to perform a prompt assessment and response, potentially with immediate corrective actions and long term safety improvements.

In the past, accidents with severe core degradation have initiated international and national activities to further enhance nuclear safety. Depending on the accident and its main root causes, different aspects have been the focus of safety improvements. In the following paragraphs, the main lessons identified from the accidents at Three Mile Island NPP, Chernobyl NPP, and Fukushima Daiichi NPP are summarized.

The accident at Three Mile Island Unit 2 (TMI-2) in 1979 was due to an effect unknown to the operating personnel (formation of a large steam bubble in the reactor pressure vessel), procedural deficiencies, and inadequate use of operating experience feedback. This severe accidents occurred at nuclear power plants and caused changes in severe accident management regulatory requirements in several States (see for example the parts of Annex I that relate to Finland, Germany, and Sweden) and an improved ability to manage severe accidents at operating nuclear power plants. This event also led to the establishment of the international reporting system (IRS), national systems of operating experience feedback in many member states, and the cooperative organization Institute for Nuclear Power Operations in the United States. This accident highlighted the importance of knowledgeable operating personnel and

appropriate operating infrastructure and led to the development of symptom-based emergency operating procedures, simulators, and training. Gauges to monitor the water level inside the reactor pressure vessel have been added to pressurized water reactors. Information and alarms displayed to the control room staff have been reconsidered under ergonomic aspects.

In 1986, an accident occurred in the Soviet Union at Chernobyl NPP Unit 4. The main reasons for the accident were inadequate safety culture and core design issues. In the aftermath of the accident, the importance of safety culture and safety management, including operator training and simulator use, was emphasized and improved in nuclear power plants worldwide. In addition, this accident revealed the importance of an effective operating experience feedback programme, as reactivity feedback issues had already occurred in the Leningrad nuclear power plant, as well as during commissioning in the Ignalina nuclear power plant. This accident triggered many modification programmes in nuclear power plants, such as implementing diverse shutdown systems, supplementary control rooms, and several accident management measures. Furthermore, PSRs have been initiated to regularly review and assess nuclear safety with the aim to further enhance safety. On an international level, the CNS [2] was established as well as the International Nuclear Event Scale (INES) to communicate the severity of nuclear events to the public, and WANO was established.

In 2011 an accident occurred at Fukushima Daiichi NPP Units 1 to 4. This accident was initiated by a combination of two external hazards (earthquake and consequential tsunami) and affected several units on site and several nuclear power plants in the vicinity at the same time (multi-unit, multi-site). In many Member States re-evaluations of external hazards and the robustness of the plant against hazards exceeding the design basis events have been initiated. Again, this event led to an enforcement of accident management measures in many plants, in particular in those plants that had not yet made major safety improvements after the event at TMI NPP and Chernobyl NPP. Furthermore, this event led to further consideration of accidents simultaneously affecting several facilities on the site leading to higher autonomy requirements for the safety features of individual facilities. A further lesson was that identified safety improvements have to be implemented in a timely manner.

Approaches

The information below provides an overview of the method used by Member States to identify safety improvements from significant events. While methods vary, Member States generally take both immediate and long term actions to such events.

Immediate response:

- The goal of the immediate response is to identify as soon as practicable whether immediate corrective actions are necessary to ensure the safety of the public—including whether plants need to continue to operate—in light of the significant event that has occurred.
- Immediate corrective actions for some licensees may include a temporary shutdown, or a request by the regulatory body to take a specific action, such as the actions after the accident at Fukushima Daiichi NPP to procure additional and appropriate external storage for the additional emergency response equipment.
- In many cases, nuclear power plants might not need immediate corrective actions because the safety of the plant is acceptable, though long term response may be warranted.

Long term response:

- A long term response is generally pursued to identify further safety improvements. The goal of long term actions is to ensure that necessary safety improvements identified as a result of the event are implemented at nuclear power plants.
- The regulatory body may request in-depth assessment of the safety of the nuclear power plant against previously unforeseen hazards or hazard levels similar to those experienced in the event in order to determine whether new hazard(s) are to be included in the design basis.
- The process entered to identify areas for enhancing safety at the plant using the lessons learned from a significant event vary among member states. Generally once safety improvements are identified, action plans are developed. In these cases, depending on the scope of changes being contemplated, the regulatory bodies may need to agree to the action plan and schedule for implementation.

3.4.1. Examples of Regional Approaches to Safety Reassessment

This section provides two regional approaches to safety reassessment that have been implemented by Member States with nuclear power plants in operation following significant events. Additional national experiences under the Regional Approaches are described further in the annexes. The parts of Annex I that relate to Finland and Germany describe actions undertaken to improve severe accident management after the TMI accident and how these activities were intensified after the accident at Chernobyl NPP. These early reactions to improve severe accident management meant that in response to the demands of the Regional Approach (European Union), only minor improvements in this area were needed after the accident at Fukushima Daiichi NPP. Further examples of improvements made after the Fukushima Daiichi NPP accident can be found in many of the national Annexes. In addition, many Member States conducted safety assessments (commonly referred to as “stress tests”). Two regional intergovernmental approaches are described in the subsections below. Other Member States including the Russian Federation (see the part of Annex I that relates to Russian Federation) independently conducted stress tests.

European Union stress tests

In response to the accident at the Fukushima-Daichi nuclear power plant, the European Council of 24/25 March 2011 requested that the safety of all European Union nuclear plants be reviewed, based on a comprehensive and transparent risk and safety assessment (“stress tests”). These “stress tests” were targeted reassessments of the safety margins of nuclear power plants, organised by the European Nuclear Safety Regulators Group (ENSREG), including the European Commission.

The safety assessment considered mainly extreme natural hazards like earthquakes and floods and initiating events potentially leading to multiple loss of safety functions requiring severe accident management. All the operators of nuclear power plants in the European Union had to review the response of their nuclear plants to those extreme situations.

The stress tests were conducted in accordance with the technical specifications prepared by the European Nuclear Safety Regulators Group (ENSREG) and covered the following topics:

- Initiating events;
- Loss of safety functions;
- Severe accident management.

The safety assessments performed for each nuclear power plant resulted in reports submitted to the respective regulatory body. These reports were first reviewed by the national regulatory bodies, who then prepared summary national reports. The national reports were reviewed by peer-review teams from countries involved in the drafting of this TECDOC as well as other stakeholders and the public. The review also included visits to selected nuclear power plant sites in each participating country. These activities resulted in several recommendations for safety improvements, both for nuclear power plants and for the regulatory framework in each country. On the basis of these recommendations, the national regulatory bodies in all participating countries developed National Action Plans.

This approach was implemented by all European Union countries with operating nuclear power plants, as well as Armenia, Switzerland, and Ukraine.

Extensive information is provided on the ENSREG website.¹⁰

Ibero-American stress tests

Once the causes of the Fukushima Daiichi NPP accident were properly understood, the Ibero-American Forum of Radiological and Nuclear Regulatory Organizations (FORO) launched stress tests in the nuclear power plants of the FORO member countries. The FORO is an association of Ibero-American radiation and nuclear regulatory bodies, which was created in 1997 with the objective of promoting radiation, nuclear safety and security at the highest level in the Ibero-American region. At present, the FORO is integrated by the regulatory agencies of Argentina, Brazil, Chile, Colombia, Cuba, Spain, México, Paraguay, Perú and Uruguay.

In a first phase, the stress tests (which were formally termed “Evaluation of Resistance of Nuclear Power Plants”) were carried out by the four FORO countries with operating commercial nuclear power plants, namely Argentina, Brazil, México and Spain. In a second phase, a process of “cross-review” of the outcome was carried out involving additional FORO’s members to verify the technical consistency and coherence of the testing as well as to allow a fruitful exchange of experiences.

A final document, available at the FORO website¹¹ (<http://www.foroiberam.org>), presents the test results and the proposed improvements, which are carried out under the FORO supervision. The results were presented at the Fukushima Ministerial Conference on Nuclear Safety, held in Japan on December 15 and 17, 2012.

¹⁰ For more information on European Nuclear Safety Regulators Group, please visit <http://www.ensreg.eu/EU-Stress-Tests>

¹¹ For more information on European Nuclear Safety Regulators Group, please visit <http://www.foroiberam.org>

The FORO stress tests were recognized and highlighted in the Declaration adopted by the Heads of State and Government at the Ibero-American Summit that took place in Cádiz on November 17, 2012.

3.5. THE ROLE OF THE REGULATORY BODY

The regulator also has an important role to play in the initiation of safety improvements.

The regulator, as part of its regulatory oversight activities, develops and implements a programme of inspection for nuclear power plants to confirm compliance with regulatory requirements, with any conditions specified in the licence, and with the licensing basis. It needs to be noted that regulatory inspection does not diminish the prime responsibility for safety of the licensee, and cannot substitute for the control, supervision and verification activities conducted under the responsibility of the licensee.

The regulator, in the conduct of its inspection activities, may identify areas for safety improvement additional to those identified by the licensee. The regulator also monitors international developments in the area of nuclear power plant safety, in particular safety improvements relevant to the nuclear power plants in the Member State. All considerations for safety improvement are brought to the attention of the licensee who is then responsible to review them and propose safety solutions to the regulator.

4. ASSESSMENT OF SAFETY IMPROVEMENTS

The different national approaches, in general, provide for a framework in which both the licensee and regulatory body continuously assess safety improvements.

Many safety improvements over time have resulted from a combination of the triggers discussed in Chapter 3. Major accidents such as the Fukushima Daiichi NPP accident have served as additional motivation for Member States to implement safety improvements in nuclear power plants.

This chapter provides a brief description of the main methodologies used to assist in the identification of potential safety improvements. These methodologies also serve to assess the overall safety contribution and practicability of identified safety improvements. This information is a valuable input to the overall decision-making process for safety improvements discussed in Chapter 5.

In most Member States the following three methodologies are typically used to assess safety improvements:

- Engineering assessments

Engineering assessments are used to review a wide range of topics related to nuclear power plant safety. Examples include the ageing of SSCs, comparison and assessment of the actual status of the plant with modern safety requirements, design codes, and standards (conducted under a reassessment of safety, a PSR), stress tests such as those conducted after the Fukushima Daiichi NPP accident, mechanical analyses, thermal-hydraulic analyses, reliability analyses, and engineering judgement. This review is performed based on the plant documentation of the current licensing basis for SSC important to safety. Typically, the review covers the range of operating duties and loads within normal operation, design-basis accidents (DBAs), and design extension conditions, application of Defence in Depth concept and considering relevant operating experience. Operating experience feedback is another trigger for engineering assessments. For example, the occurrence of unplanned events can indicate that the original engineering assessment was not sufficient to recognize and take account of the full range of potential operating situations. Specific design requirements regarding engineering assessment are established in SSR-2/1 (Rev. 1) [17]. Recommendations and guidance relevant to engineering assessment can be found in the IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [35]. Refs [36–40] provide additional information regarding engineering assessments.

Relevant examples of engineering assessments can be found in the parts of Annex I that relate to Finland, Russian Federation, and in Annex II.

- Deterministic safety analysis (DSA)

DSA is used during the licensing process to demonstrate whether the plant design can meet the prescribed regulatory limits, including ALARA, for the limiting safety parameters, and for radioactive releases and radiation doses resulting from normal operation, anticipated operational occurrences and accident conditions. Subsequently, during the operating life of the plant, DSA is repeated in whole or in part, as necessary, to confirm to the satisfaction of the regulatory body that plant safety measures remain adequate (the plant continues to

be in compliance with its licensing basis) taking into account operating experience and the actual status of the plant. In addition, the DSA is reviewed in the frame of PSRs for the assessment of proposed safety improvements such as plant modifications. DSA cases may also need to be rerun following significant plant upsets that resulted in reactor shutdown via the safety systems, when this action by the safety systems was not previously predicted. The new analysis may show that safety improvements are necessary to ensure the continued validity of the licensing basis.

DSA is used to justify the safety case by demonstrating compliance with deterministic acceptance or safety criteria. The selected approach depends on the scope and objective of the analysis. For example, DSAs performed for DBAs typically use a conservative approach, while DSAs used for DECAs use a less conservative approach, as indicated in SSR-2/1 (Rev. 1) [17]. Because the more realistic “Best Estimate” analysis does not use the same level of conservatism as “Conservative” analysis, there is higher level of confidence that the “Conservative” analysis predictions will bound the consequences than the “Best Estimate” predictions. As a result, “Best Estimate” is not used for Design Basis Accident analysis. Requirements on DSA are established in GSR Part 4 (Rev. 1) [16]. SSG-2 (Rev. 1) [23] provides recommendation and guidance on the use of DSA. Complementary information on DSA is available in Refs [41–43].

- Probabilistic safety assessment (PSA)

PSA is a systematic and comprehensive methodology utilizing fault trees and event trees to evaluate the risks associated with the operation of a nuclear power plant. A PSA typically includes initiating event analysis, accident sequence analysis, systems analysis, analysis of dependent failures, analysis of common cause failures, and human reliability analysis. Importance analysis, sensitivity analysis and uncertainty analysis are additional components that are essential to put the PSA results in proper context.

PSA provides qualitative and quantitative insights into the safety of a nuclear power plant and it can be useful in ranking the PSR findings or proposed plant modifications in terms of their safety significance. Other PSA applications include the identification of systems and components important to safety.

PSA and its results are powerful tools to determine whether planned safety improvements and design provisions will be effective in stopping or managing the progression of a severe accident, and the effectiveness of mitigating strategies. An example of the practical use of PSA for design changes can be found in the part of Annex I that relates to Hungary.

Member States have recognized the need for the evaluation of a site risk in an integrated way. For example, the Fukushima Daiichi NPP accident demonstrated the potential for an accident involving nearly concurrent core damage at multiple reactor units and spent fuel pools. The evaluation includes consideration of the potential for accidents involving multiple installations concurrently, and in an appropriate way, the aggregation of the various risk contributions from different sources, diverse hazards and plant operating states.

Member States have efforts underway to collect best practices on this topic. For example, Korea, Canada and the USA have already prepared multi-unit Level 1 and Level 2 PSAs, and to some extent, Korea performed a multi-unit Level 3 PSA. The IAEA has started studies to determine a suitable approach to carry out multi-unit PSA for the different

designs of nuclear power plants. Requirements on PSA are established in GSR Part 4 [16]. Recommendations on the use of PSA are provided in IAEA Safety Standards Series Nos NS-G-1.7, Protection Against Internal Fires and Explosions in the Design of Nuclear Power Plants [44], and NS-G-1.11, Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants [45]. Complementary information on PSA is available in Ref. [46].

5. INTEGRATED DECISION MAKING

5.1. PROCESS OVERVIEW

This Section presents the overall decision-making process for safety improvements. Specifically, the addition of a safety improvement to an existing nuclear power plant is a decision based on a solid analysis including considerations such as: the reduction in radiological risk, the gains in nuclear safety, the complexity of the proposed safety improvement, cost of its implementation (including procedures), maintenance and testing.

When the existing reactors were commissioned, their original level of safety met the required safety level based on the regulatory requirements that were in force at that time. Measures to maintain or exceed the original level of safety at the nuclear power plant were gradually taken in accordance with new knowledge and experience that has emerged from analysis of operating experience, newer nuclear power plant design concepts, research and development, and the refinement of safety assessment methodologies such as DSA and PSA, as discussed in Chapters 3 and 4.

Significant events that have occurred at nuclear power plants, such as TMI NPP, Chernobyl NPP or Fukushima Daiichi NPP, have also influenced these measures. Modernization programs and power uprate programs are another important mechanism to identify safety improvements.

In many Member States, comprehensive and systematic safety assessments are carried out for existing installations throughout their lifetime. A PSR aims at ensuring compliance with the existing licensing basis and applicable requirements and identifying safety improvements by reviewing current (modern) requirements and designs, as well as identifying further safety improvements (e.g. by taking into account developments in science and technology). Those safety improvements necessary to maintain the licensing basis have been regarded as mandatory, whereas safety improvements that enhance and increase safety to a level that approaches that of new reactor designs, are implemented on a “reasonably practicable” basis.

Utilities follow a process of continuous improvement that has been applied since the commissioning of the existing reactors. This process is based on the identification, selection, prioritization, development and implementation of safety improvement. Figure 2 provides an overview of the process. These steps are consistent with the recommendations provided in IAEA Safety Standards Series No. NS-G-2.3, Modifications to Nuclear Power Plants [47], and take into account that the activities conducted do not always follow a linear process.

Identification of issues and corresponding safety improvements is the initial stage of the process. The objective of this stage is for the licensee to identify an issue of concern and determine a list of potential safety improvements that would address the issue. Chapter 3 lists and explains a number of processes that initiate this identification. Chapter 4 provides information on methodologies used for the assessment and identification of safety improvements initiated by these processes.

The regulator may also identify issues of concern through its regulatory oversight program. These are brought to the attention of the licensee who is then responsible to assess them and propose safety improvements to the regulator.

Selection of safety improvements and associated proposals for solution is the second step of the process. The objective of this step is to evaluate the range of solutions that can address the issue of concern and select specific solutions for further review that best address the constraints under which the licensee is operating. The licensee will assess which solutions to eliminate and which to define in detail and to optimize for implementation.

Prioritization is the third step of the process. Its objective is to determine the prioritization of improvements to implement at the facility on the basis of criteria that determine their importance. These criteria may include safety significance, regulatory requirements, level of documentation or approval (including environmental assessment) needed, and financial and technical feasibility.

Detailed design is the fourth step of the process. Its goal is to produce the detailed and optimized design for every solution that has been chosen to cope with the selected safety improvements. Technical and economical optimization of the solutions is important, as is the timely implementation of safety improvements. Different factors are considered in a detailed way in this step of the process.

Implementation is the final step of the process, where the means to realize the solution chosen for the safety improvement is planned in detail, following approved procedures. (The actual realization of the solution is beyond the scope of this TECDOC.¹²) Implementation is expected to include commissioning (SSCs) or validation and verification (procedures, software) sufficient to demonstrate to the extent practicable the safety claim made regarding the installed safety feature.

At each step of the process, continuous communication between the licensee and regulator may be needed, depending on the significance of the project. The objective of this communication is the overall safety enhancement of the plant, taking into account different inputs explained in Section 5.6. This communication is also useful for the licensee to keep the regulator informed regarding any delays in implementing agreed modifications.

¹² Also, not addressed in detail in this TECDOC are situations in which a licensee determines for economic reasons that implementing a particular modification is not feasible and a decision is taken to discontinue operation.

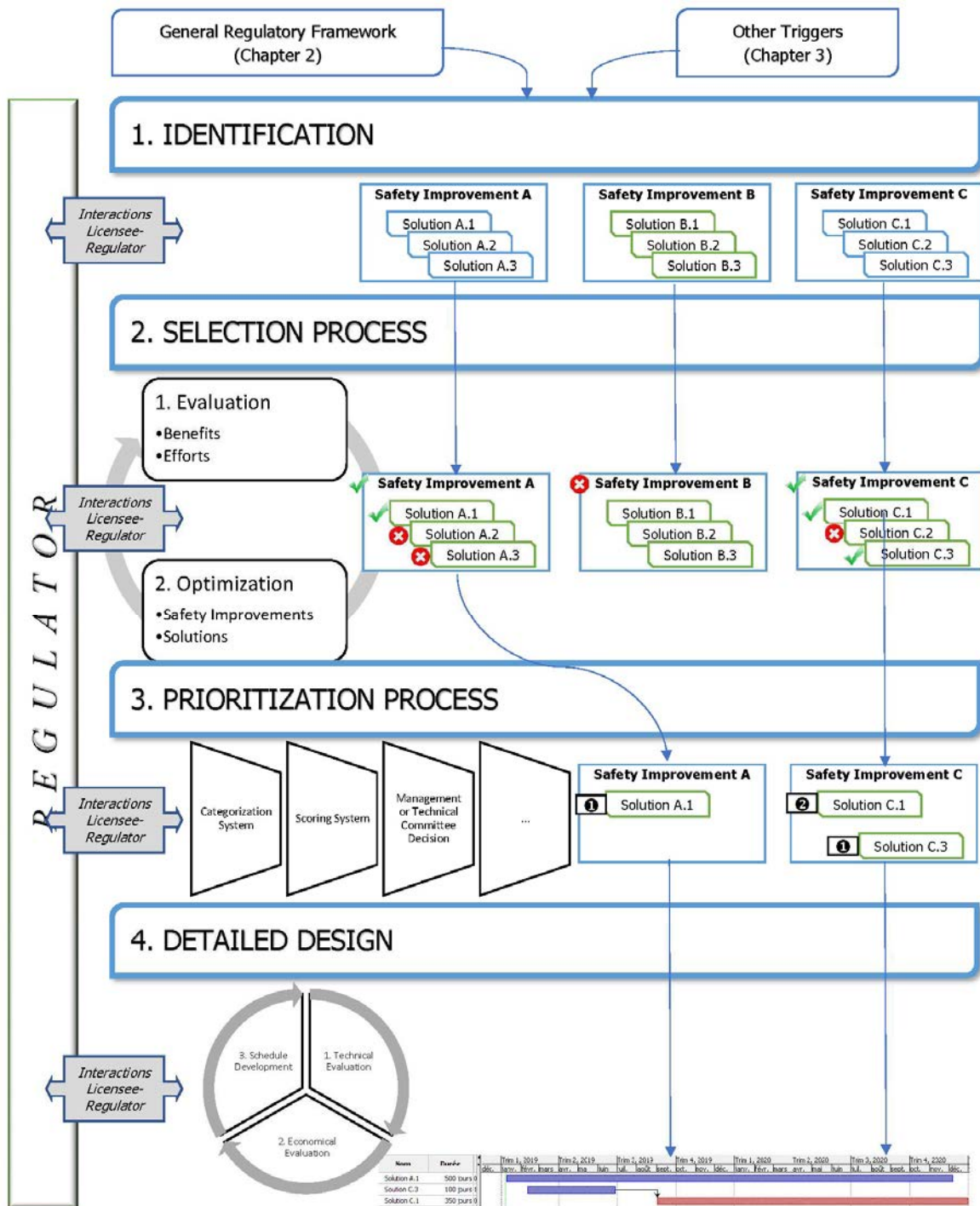


FIG. 2. Summary graphic showing process steps related to continuous improvement.

5.2. IDENTIFICATION

Identification of issues of concern and corresponding safety improvements and associated proposals for solutions is the initial stage of the whole process. The objective of this stage is for the licensee to determine a list of potential safety improvements that address an issue of concern and to decide (in agreement with the regulator, as needed) those that could be implemented on site. The overall safety of the nuclear power plant is considered in identifying the range of possible solutions.

The safety improvements can be general, such as improving one or more safety functions, or detailed, such as modifying a specific system or implementing a new one. The possible solutions to address the issue of concern can be, as described in NS-G-2.3 [47], either modification of structures, systems or components, or operational limits and conditions, or procedures, or software, or the management systems and tools for the operation of a nuclear power plant.

As described in Chapters 2 and 3, there are basically two different sources from which safety enhancements can be identified:

- The first one corresponds to changes made in the national regulatory framework, potentially as a result of changes to international standards (see Chapter 2). Changes to the framework itself would be led by the regulator.
- The second one is related to the analysis, assessments, or other well-established processes performed routinely under the scope of the current licensing basis for a given facility (see Chapter 3). This source is mainly driven by the licensee.

As noted in Chapter 3, issues of concern and corresponding safety improvements can result from licensees' ongoing programmes (e.g. ageing management), integrated safety reviews (e.g. PSRs), and specific safety reviews after significant events, and input from the regulator.

The expected output of this stage is a list of potential safety improvements or gaps identified with different proposals for a solution. Normally, in this step of the process, different solutions are categorized (see NS-G-2.3 [47]) and globally designed, which means that they are designed with a low level of detail and are not optimized. In the next step of the process, the licensee will select, based on several parameters, which of these potential safety improvements will be optimized and which of the safety improvements are not to be pursued.

In proposing a modernization project, the licensees consider safety impacts. Undertaking a large scope modernization may provide opportunities to enhance safety at the same time. For example, some regulatory approaches may apply new or updated requirements or guidance to the review of licensee requests for modernization, providing the opportunity to enhance safety through imposition of these provisions. Furthermore, licensees search for and avoid negative safety impacts (unintended consequences) of modernization.

The specific area for safety improvement at any existing nuclear power plant will depend on the overall regulatory approach, the location of the nuclear power plant, the design type of the facility, plant age and history and other technological, regulatory, and plant-specific considerations. There is no standard set of specific engineering or operational improvements that will be appropriate for all reactors and operational regimes, though it is common practice for licensees to also look at what others have done. To identify different solutions for each safety improvement, the licensee will take into account the relevant state-of-the-art of technology. Tools that are low in complexity (compared to those used in the “detailed design and optimization” step of the process) are used to generate a quick, global design.

Interaction with the regulatory body in this initial stage of the process could be necessary depending on the type of triggers and on the specific national regulatory framework. In case of non-compliances, the licensee has to restore compliance, which can include plant modifications or modifications of the licensee's activities. Regulators can also approve exemptions from certain requirements, as allowed under the national regulatory framework.

5.3. SELECTION PROCESS

The objective of the selection process is to determine, among the potential solutions identified in the previous step, which to consider for further optimization, design, and implementation and which to eliminate. In this step it is also very important to take a holistic approach because there might be potential solutions or smart combinations of solutions that cover more than one safety improvement or gap.

The selection step is carried out taking into account the prevailing circumstances, namely, which safety improvements are reasonably practicable or achievable for particular plant design or facility. As in the identification stage, these will depend on: the overall regulatory approach; the site of the nuclear power plants; the design of the facility; the plant age and history; and other technological, regulatory, and plant specific considerations. What is achievable for a certain plant might not be reasonably achievable for another plant.

As the result of the selection process, there will be two categories of outcomes from the analyses performed:

- (1) Selected solutions or combinations of solutions for further development;
- (2) Rejected solutions.

Licensees may implement some improvements without a further selection process, such as when they represent a standardized industry practice or respond to a common experience. An example of these improvements commonly implemented is the symptom-based approach for severe accident management. In addition, if the regulator requires that a specific means of safety enhancement be implemented, no further selection process is needed, and the licensee can directly transition into the detailed design stage (see Section 5.5). Nevertheless, potential solutions might be optimized or smartly combined with other solutions.

Except for these specific types of improvements there will be always a set of ideas (mostly on a generic level) that could potentially improve safety; for instance, the amount of coolant for core cooling or new strategies to achieve ultimate heat sink. It is the licensee's duty to analyse those ideas and select such improvements that are achievable for the particular plant status and could be implemented in a timely manner.

To select "reasonably practicable or achievable" safety improvements and associated solution(s) that could be implemented in a timely manner, the licensee evaluates, at least globally at this stage, the benefits given by every solution and the efforts needed to design, optimize, and implement them, as well as any disadvantages associated with their implementation (e.g. increased worker doses, generation of radioactive waste). The licensee chooses which elements it wants to evaluate, including:

- Benefits: prevention and/or mitigation improvements, safety margins, radiation risk reduction, radioactive dose reduction;
- Efforts and disadvantages: costs, collective dose, time schedule, technical complexity.

Other factors can be taken into account, such as the remaining plant lifetime (see for example the part of Annex I that relates to Lithuania) and whether the solution would be first-of-a-kind. Normally, and because the process is continuous, the parameters that are evaluated here are, at least, the ones that are used in the step “*prioritization*” (see Section 5.4) and in the step “*detailed design*” (see Section 5.5). The level of detail and precision are different between the evaluations in this current step and the other ones.

The tools to identify gaps and propose solutions include, for example: PSA and DSA results (as discussed in Chapter 4), cost/benefit analysis, experimental studies, draft schedule. With those tools, the licensee determines the potential safety improvements that are reasonably practicable or achievable and those that are not. At the same time, the licensee prepares justifications for its choices because, in certain Member States, the selected and dropped solutions have to be presented for approval by the regulator. Some regulatory bodies have a process in place to decide what the licensee needs to submit for approval. Some examples of this process are shown in the parts of Annex I that relate to Romania and Sweden.

Before the discussion starts between the regulator and the licensee, licensees develop their proposals, including what the licensee plans to improve and what the resulting safety benefit will be. The outputs would support regulatory acceptance of the selected improvements, as well as any that were not selected even though they would address relevant guidance or expectations (e.g. an evaluation of new safety standards). The understanding between the regulator and the licensee is important and it is in general beneficial that it is established before the selection process is finished.

For example, the part of Annex I that relates to Germany provides insights on how safety improvements after the accident in Chernobyl NPP were selected by the German advisory committee RSK. Namely, within RSK, experts from vendors, utilities, expert organizations, as well as scientists, agreed by consensus on the selected safety improvements to be implemented in German nuclear power plants.

5.3.1. Optimization of reasonably practicable or achievable approaches

A key element of the discussion between the licensee and the regulator is the concept of “reasonably practicable” or “reasonably achievable” and how this concept has been used and applied in the selection process.

This concept is an important element of the Principle 2 of the Vienna Declaration of Nuclear Safety and is part of Article 8a of the EURATOM Nuclear Safety Directive. This concept is analogous to the optimization principle referred to in SF-1 [13], where it is more broadly applied to all aspects of nuclear safety.

Optimization also means that the selected option would be considered the best under the prevailing circumstances, consistent with SF-1 [13]. Paragraph 3.21 of SF-1 [13] states:

“The safety measures that are applied to facilities and activities that give rise to radiation risks are considered optimized 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 utilization.”

The term “optimization” corresponds to the best compromise between risk reduction and efforts needed to achieve this reduction.

Because the selection of safety improvements that are reasonably practicable or achievable corresponds to a compromise, it is not possible to define in advance a single point that could be called “optimum.” In practice, it is possible to represent it as an area as shown in Fig. 3. This explains that some point may exist for which improving safety measures might not be reasonably achievable. At this point, the licensee can demonstrate to the regulator, as necessary, that the risk reduction is disproportionate to the efforts needed to achieve such a reduction. Furthermore, some residual risk remains after making the maximum reasonable effort to reduce risk. Figure 3 shows an illustration of those notions:

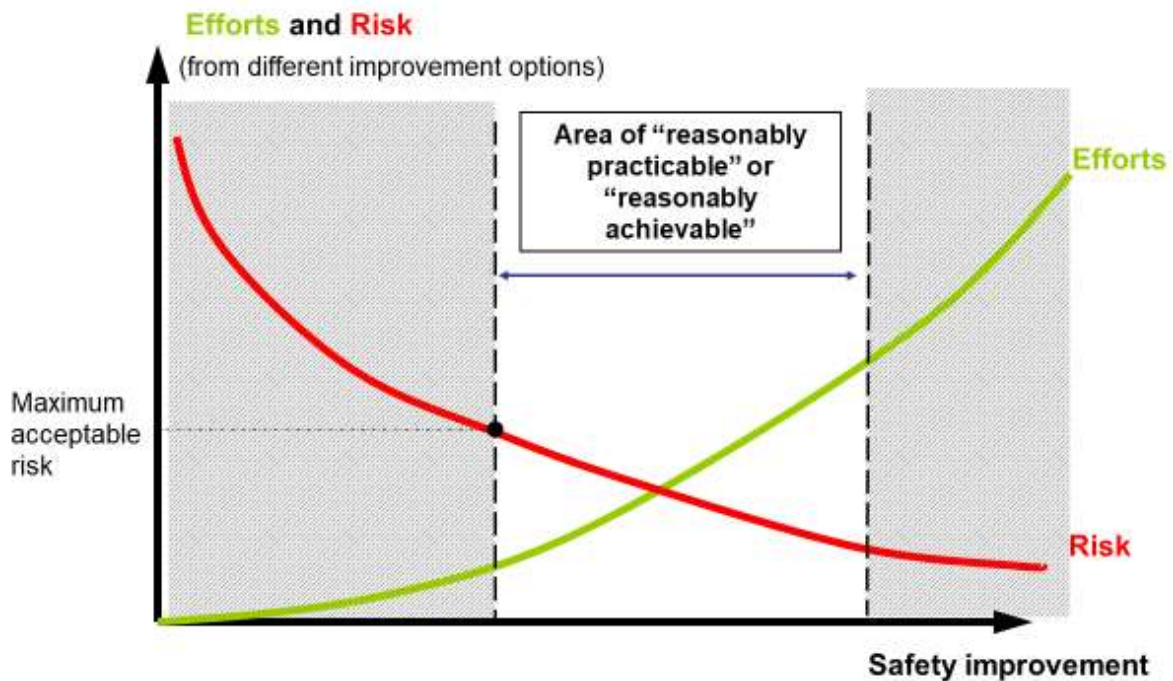


FIG. 3. Conceptual view of the “reasonably practicable or achievable” area.

Factors such as costs, time, production restraints and occupational exposures are limiting factors in licensees’ ability to implement more and more safety improvements. At a certain point, assuming that a specific improvement being considered by a licensee is not a regulatory requirement, the improvement could need so many resources that it could be viewed to unduly limit the utilization of the facility (e.g. if the generation of nuclear electricity becomes unfeasible) and no longer be pursued.

Most regulatory frameworks do not prescribe a systematic approach for assessing what is reasonably practicable or reasonably achievable. Therefore, the process is normally considered on a case by case basis, in part by using engineering judgement. Since the responsibility for safety lies only on the licensee, it is the licensee’s responsibility to justify and convince the regulator that additional measures are either justified or not and that the available options are optimized.

Safety research and advances in science and technology, as well as revisions to international safety standards, support decisions on a specific solution as evaluated by the licensee. Insights from PSAs and PSRs, for example, may also bring new insights for safety improvement needs when looking at 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. As discussed in Chapter 4, PSA 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 need to be considered (e.g. the challenges of implementing a solution that is unproven or with which there is no experience). Licensees can compare the costs of the plant modification to the gained safety improvement and for example propose alternative solutions based on the PSA results and overall safety of the plant. It is important to have an overall picture of the plant safety.

5.3.2. Reasonably practicable or achievable: examples

Significant limitations leading to a conclusion that a particular solution is not “reasonably practicable” include the following:

- Technical infeasibility of implementing a solution (e.g. major plant layout changes);
- Permanent worsening of operability of the plant (significantly longer outages, increase of collective and individual effective doses, decrease of robustness of existing barriers in defence in depth);
- For safety improvements that are not mandatory, efforts and implementation time to implement a safety improvement (e.g. feasibility to recover costs in the remaining plant lifetime) are not justified by the magnitude of the safety improvement that would result.

5.4. PRIORITIZATION PROCESS

The objective of the prioritization process is to determine the order of implementing various improvements at a given facility on the basis of criteria related to the significance of the improvement to nuclear safety, relevant regulatory requirements, financial impacts of the project, and technical feasibility of the improvement. The solution(s) identified to cope with the selected safety improvements, at the level of the basic design, will be, after prioritization, incorporated into the schedules for their implementation, where they will go through the procurement phase and elaboration of a detailed design.

The prioritization begins with the selected safety improvements and their selected solution(s) at the level of basic design coming from the selection step of the process. The output of the process is a sequence of implementation, taking into account the criteria discussed above and others as applicable. Projects have defined technical specifications at the level of basic design sufficient for procurement or preparation of detail design.

A method for prioritizing safety improvements is established at nuclear power plants typically prior to their implementation. The method is generally based on the safety significance of each proposed improvement and associated solution(s), and then applied to detailed design later in the development process. The approach adopted could be based on DSA, PSA, engineering judgement, cost-benefit analysis, or a combination of analytical methods.

Experts participating in the development of this TECDOC provided as examples several different approaches for the prioritization of modifications: categorization methods, scoring methods, and management or technical committee decisions.

Categorization

In some countries that shared their experiences (see part of Annex I that relates to Netherlands), regulatory requirements and a cost-benefit analysis are used to categorize modifications. When a cost-benefit analysis is used, the licensee may consider safety benefits (e.g. safety margins, prevention and mitigation improvements), availability benefits (e.g. power level, operating time, thermodynamic efficiency), and the time schedule (including availability of human resources, needed equipment, and administrative delays).

Depending on the results of that cost-benefit analysis, a licensee may elect to implement faster modifications with a smaller increase of the safety level (quick wins), potentially while the licensee is also in the process of planning a longer-term modification that might significantly increase safety.

Scoring

Some Member States apply a scoring approach based on the allocation of points from individual experts of nuclear facility departments. They have a number of points that correspond to the level of impact on nuclear safety, work safety and security, reliability, economy, and technical feasibility. The licensee may use specific categories to evaluate a particular project in terms of safety, regulatory, and business considerations, including nuclear safety significance, the complexity of approvals that would be needed, operational support, equipment reliability, and economic or strategic value.

Each category includes attributes that divide categories as needed to achieve a more accurate evaluation. Approved projects determined to be mandatory go to the top of the project list where they receive top priority and no further ranking is necessary. Non-mandatory projects address safety issues that do not pose an imminent threat to plant operation. This is the case for virtually all projects considered in long term planning. Thus, these projects are prioritized based on the number of points they receive.

All scores in each of the categories are summed to provide a true understanding of the significance of the project. For an example, see the Annex II.

Management or technical committee decision

This approach is based on requirements or proposals for design change. After an evaluation of the change by the designers, the change request is reviewed and approved by the technical committee. The technical specifications or basic design for the modification are developed in line with the expected financial expenses of the project. A final decision is made by the company's management and by its technical committee, which set the priority and include the project in the nuclear safety enhancement programme for investment and implementation, subject to approval by the regulatory body, as necessary. In general, it is also very important to have the involvement of the regulator before the final decision of the management for significant modifications.

5.5. DETAILED DESIGN

At this step of the process, the licensee has a ranked list of solutions that can be implemented in order to cover the selected safety improvements, but those solutions are not completely designed and not optimized from a technical and economic point of view. The goal of the following step of the process is to produce the detailed and optimized design for every solution. In practice, the licensee will:

- Define precisely either technical solutions (such as system or materials modifications) or programmatic solutions (such as procedure modifications);
- Define precisely the processes to produce the detailed design and define the methods and tools they use, how the methods and tools are used, and the input data and the parameters;
- Define precisely the time schedule for the necessary studies and verify that the time schedule is in accordance with the objectives fixed by the national and/or local legislation, as applicable.

The licensee uses methods, tools, and data with a suitable level of detail to be able to implement the selected solution, taking into account any agreed-upon deadlines for the whole process.

At the end of this step, as necessary, the regulator formally agrees on the solutions that will be implemented. To do this, the regulator needs, for every modification, enough information to verify that the proposed solutions will meet applicable requirements within the regulatory framework; accordingly, the licensee produces sufficient information to justify its choices and the implementation approach and schedule to the regulator.

In this step of the process, the licensee tries to find the best compromise between the technical and economic elements for the selected solutions. To design and optimize one or more technical solution(s) to cover the safety improvements, often the licensee has firstly to determine precisely what are the safety and technical requirements (whether regulatory requirements or self-imposed standards) that the solutions have to respect and what guidance or other technical documents exist to support licensees in determining how to meet the requirements. In some Member States, the detail of design may progress further following regulatory approval, depending on the level of detail required by the regulatory body.

Design activities may be ongoing at the same time as the development of national safety and technical requirements. Therefore, the licensee may need to study different options for every safety improvement at the same time, to take into account different possibilities.

The following aspects need to be considered for every option that the licensee evaluates:

- The safety benefits (e.g. safety margins, prevention and/or mitigation improvements, PSA improvements, impact on radiation protection);
- The availability benefits and costs (e.g. power level, operating time, thermodynamic efficiency)
- The costs and other disadvantages (e.g. workforce costs, materials and equipment to buy, impact on the lifetime of the nuclear power plant, worker dose)
- The risks (e.g. technical risks, acceptability of the solutions) associated with any uncertainties (given the limited time dedicated for completing the studies);
- The schedule for implementation, considering the availability of human resources, equipment needed for implementation and administrative delays.

The optimization phase tries to find the best design that takes into account the benefits and efforts based on those parameters.

Safety benefits are evaluated using techniques including:

- Engineering judgement and DSA;
- PSA (applied with different frequencies in various Member States);
- Benchmarking with other countries.

The availability benefits and costs are always evaluated using classical tools and methods (e.g. thermodynamics software, engineering judgement). Those methods and tools can be developed by the licensee or appropriate commercial products may also be used. In every Member State, the tools and methods used are to be fully qualified for use in the nuclear industry. That qualification is in general defined by the regulator.

The costs for every option are evaluated in every Member State because this is a key element for the final decision. Even when the safety improvement has to be done whatever the associated cost, the licensee still evaluates costs in order to determine the solution that can provide the necessary safety improvement for the lowest cost. Some Member States use methods to evaluate the costs, but some others use specific approaches such as ‘design to cost’ methods or ‘design to value’ methods.

For Member States performing PSRs, it might not always be possible within the PSR schedule to complete full-scope studies that support the detailed design of every option. As a result, uncertainties may remain at the end of the process. Some Member States use specific approaches, including risk-informed approaches, to take account of these uncertainties in the prioritization process.

5.6. INTERACTION WITH THE REGULATORY BODY

Depending on the regulatory system in each State and the origin of the safety improvement, the interaction between the licensee and the regulator and the information that is produced by the licensee may vary.

The regulator may impose a new requirement for a safety improvement. The requirement may be prescriptive, particularly if it is focused on a specific design or site.

There may be a need for approval of the schedule of implementation to ensure the improvement is made in a timely manner. Otherwise, inspection of the implementation may be the primary means of regulatory confirmation of compliance—either planned or unplanned inspections. The regulator may also define an inspection programme to verify the adequacy of the process and confirm that implementation conforms to the approved proposal and plan.

If a regulatory requirement is more general or performance-based, discussion between the licensee and regulator may be needed to determine an appropriate and timely approach. The licensee retains the primary responsibility for safety, and the regulator’s role is to verify that the approach taken meets relevant requirements and agreed timing.

If the licensee proposes to make a safety improvement, the regulatory structure may enable the licensee to make certain changes without regulatory approval (and may necessitate regulatory approval of more significant changes) prior to implementation. If the safety significance is such

that regulatory approval is needed, timely review by the regulator will contribute to timely safety improvement. This process may be iterative to ensure that relevant safety requirements are met under the licensee's proposed approach. If the licensee has the ability to make the change without prior regulatory approval, inspection may be warranted to verify continuing compliance with regulatory requirements. Generic issues identified during such inspections may be triggers to adjust the regulatory framework accordingly. In some cases, licensees may make safety improvements that exceed current requirements. If new or updated regulatory requirements in that area become necessary, the regulator may use these safety improvements as a reference point in establishing an updated licensing basis for the plant.

For the larger safety reassessment projects like PSR or LTO, these interactions might take place at multiple stages, including the following:

- Before the execution of the project, the licensee may need to provide documentation about the scope, process, timing, references, and evaluation methods used to find the gaps or opportunities for improvement.
- During the evaluation phase, there may be progress meetings, regulatory review of interim products (e.g. in the case of PSR there might be a document for each safety factor), or inspection (e.g. on the quality of the process).
- At the end of the evaluation phase (before the start of the integrated decision-making): the evaluation report, containing all the results of the evaluation process with the findings and related potential improvements (alternatives) may be reviewed by the regulatory body.
- Before the start of the integrated decision making, the licensee may present its process to the regulator for approval (alternatively the regulator might have regulations or guidance that governs the process). Elements that might be included are (i) a risk matrix using e.g. deterministic criteria, probabilistic criteria, radiation criteria, (ii) how costs are weighted, and (iii) how integration took place (e.g. one modification covering several identified safety issues)
- After decision making by the licensee, the results and (in a transparent way) the steps to reach the results may be sent to the regulator for final approval and agreement (this might be after several iterations of a draft version with the regulator). This could be in the form of a conceptual design or integrated improvement plan (this does not yet contain planning or prioritization), which could also include information on the predicted global safety impact and how balanced the final safety results will be in respect of risk contributors and/or improvement of the defence in depth barriers.
- After the agreement on the improvement plan, the licensee may provide a detailed plan of implementation (or integrated implementation plan), with a schedule and priorities for approval by the regulator; this approval may also be subject to periodic progress reporting. Elements of planning could include a licensing plan, progress steps to implement the modification, schedules (design, manufacturing, construction, installation, commissioning). The regulator, assuming there are applicable requirements driving the change, would determine whether the plan, priorities, and modifications are appropriate. This might take several iterations of draft versions. The regulator may require the licensee to accelerate its implementation, take alternative approaches, or even shut down the facility until the modification is complete.
- For each modification itself, some regulatory structures may require additional detailed information, depending on the safety class of equipment to be affected. The licensee and regulator might discuss preliminary versions.

- Periodic progress meetings will be held throughout implementation, and the regulator may inspect the implementation, depending on its safety significance, including such aspects as planning, technical specifications, safety requirements, and documentation. If the information from the proposed documents changes during the implementation of the modification, then all the affected modification documentation may need to be reissued and reapproved by the authority.
- Depending on the regulatory requirements or agreements the licensee may produce a final report to the regulator for approval, stating the full implementation of the improvement plan, which could include a determination of the realized safety impact.

6. EXAMPLES OF COMMON SAFETY IMPROVEMENTS

The strategies, tools, and methodologies described in this TECDOC have been used by various countries involved in the development of this publication to identify safety improvements at existing nuclear power plants. A list of examples is presented below as a summary of the outcomes that have resulted in the past from the processes described in this TECDOC. These need to be viewed as examples, not as a complete list of safety improvements that have been (or need to be) implemented. The examples might be applicable only in certain regulatory frameworks or for certain designs.

Safety improvements can be organized into various categories such as: (i) those that are oriented toward preventing accidents (e.g. through maintaining criticality control, heat removal, and barrier integrity), and (ii) those that are oriented toward mitigating the consequences of an accident.

On the subject of accident prevention, some examples of topical areas that have been sources of safety improvements in multiple Member States are as follows:

- Provision of additional water sources for core and spent fuel cooling, as well steam generator makeup, including hardened installed equipment and non-permanent equipment using appropriate connections. A detailed example for an additional hardened water supply can be found in the part of Annex I that relates to Switzerland.
- Provision of additional power sources (e.g. hardened installed equipment and mobile generators with appropriate connections).
- Provision of backup capability for the ultimate heat sink (e.g. air-cooled measures).
- Addressing natural external hazards, including seismic strengthening (e.g. of tanks and passive flow paths) and protection against flooding events (e.g. raising of dykes and banks and enhancement of leaktight barriers), as well as human-induced hazards (e.g. industrial hazards, aircraft crash). Several examples can be found in part of Annex I that relates to Japan.
- Addition of spent fuel pool instrumentation to monitor temperature and level.
- Development of features to protect reactor system integrity, such as improved reactor coolant pump seals to limit coolant losses in station blackout events.
- Improvement to emergency response and accident management capability, such as procedures for depressurization to enable water injection, or a rapid response force for dispatching offsite mobile equipment.
- Expansion or enhancement of suction strainers (e.g. suppression pool, containment sump).
- Identifying and reducing key risk contributors (e.g. enhancing equipment or system availability and reliability, considering shutdown and transitional risk).

On the subject of accident consequence mitigation, some examples of topical areas that have been sources of safety improvements in multiple Member States are as follows:

- Hydrogen control, such as passive autocatalytic recombiners or hydrogen igniters;
- In-vessel retention of molten core material through various design solutions or improvements in the ex-vessel core melt retention concept;
- Pressure control features for both the reactor system and containment;
- Filtering and/or scrubbing of releases through dedicated systems;
- Qualified instrumentation for monitoring severe accident conditions and the effectiveness of mitigation actions;

- Improvement of emergency preparedness and response capabilities (increasing the robustness of emergency control centre, enhancement of communication system, drills);
- Enhancements to procedures and training (e.g. severe accident management guidelines).

The annexes to this report provide additional detail of individual participating countries implementation of safety improvements in many of these areas.

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ANNEX I. NATIONAL EXPERIENCES ON IMPLEMENTING SAFETY IMPROVEMENTS AT EXISTING NUCLEAR POWER PLANTS

This annex collects experiences from individual Member States with respect to their national regulatory frameworks and the timely implementation of reasonably practicable safety improvements (either completed or planned) at existing nuclear power plants, as well as additional information on collective efforts and supporting documentation. These contributions were developed independently based on the topics of particular interest to the contributing Member State, within the general subject of safety improvements at existing nuclear power plants. No attempt was made to harmonize the format or content; rather, the annexes reflect the diversity of safety-focused approaches and experiences across the global nuclear community. References within the main body of this TECDOC to specific annexes are intended to provide selected examples of interest regarding topics discussed in the TECDOC, not an exhaustive cross-reference. Inclusion of selected examples does not imply that a topic is not addressed in other annexes or considered by other Member States.

I-1. ARGENTINA

I-1.1. Introduction

This deals with experiences on implementing safety improvements at existing nuclear power plants. The purpose of this Annex is to describe the experience and position of the Argentine Nuclear Regulatory Authority (ARN) on this issue following the format established by the Secretariat.

I-1.2. Regulatory framework

This section describes the Argentine regulatory framework and ways in which it expresses fundamental safety objectives and expectations in relation to the purpose of this publication. It also addresses international treaties and cooperatively developed approaches of which Argentina is part as well as how the Argentine regulatory framework adopts or considers the IAEA safety standards.

It is to be underlined however that the relevant international legally binding obligations and political commitments on nuclear safety, international nuclear safety standards and national nuclear regulations have all been developed mainly for prospective planned situations, namely for the introduction of nuclear power plants including their siting, design and operation. They are not generally tailored to impose legally binding obligations to retrospective situations such as implementing safety improvements at existing nuclear power plants. Notwithstanding, when there are elements specifically introduced for dealing with retrospective gaps towards updated standards, they are framed on the achievement of safety improvements as far as practicable.

With the above provision, the overall framework is described hereinafter.

I-1.3. Relevant national legal framework: the Argentine regulatory structure

The Nuclear Regulatory Authority (Autoridad Regulatoria Nuclear, ARN) is the competent Argentine national agency for regulation of radiological and nuclear safety, safeguards, physical protection and nuclear security. The ARN was created in 1997 by the National Law No. 24,804, as an autonomous entity within the jurisdiction of the federal Presidency of Argentina, fully competent for establishing regulatory standards, granting licenses for nuclear power plants, carrying out the regulatory control necessary to ensure and enforce the compliance with standards and regulatory requirements.

In relation to safety improvements at existing nuclear power plants, the legal framework entitles ARN to impose regulatory requirements. However, this needs to be carefully exercised not to create conflicts with already issued licenses.

The licensing framework imposed by ARN for NPPs has the following features:

- The operation licenses are issued for 10 years periods
- The granting of licenses after this period is implemented by the issuance of regulatory requirements and other legally binding instruments.

I-1.4. Relevant legally binding instruments: International Conventions

Relevant legally binding commitments were undertaken by Argentina under the Convention on Nuclear Safety, signed and ratified by Argentine through Law 24776 sanctioned on February 19, 1997. Again, it is underlined that the Convention on Nuclear Safety was developed with a prospective perspective. Only its Article 6 establishes retrospective elements.

Therefore, in relation to implementing safety improvements at existing nuclear power plants, the main obligation undertaken is that Argentina could be construed to be those established in Article 6, namely to take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention had entered into force for Argentina be reviewed and, when necessary in the context of the Convention, Argentina will have to ensure that all reasonably practicable improvements are made to upgrade the safety of its nuclear installation.

Associated obligations under the Convention on Nuclear Safety relate to the assessment and verification of safety and they include, inter alia, that Argentina will take the appropriate steps to ensure that comprehensive and systematic safety assessments are carried out through the life of a nuclear installation. In accordance with these obligations, such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information and reviewed under the authority of the regulatory body.

Compliance with the legally binding obligations of Argentina under the Convention are described in the reports on the measures it has taken to implement each of its obligations, which were submitted for review prior to each meeting of the contracting parties and are made publicly available.

Argentina is also contracting party to all other nuclear Conventions, but their obligations play a minor role in the safety improvements at existing nuclear power plants and therefore they will not be described here.

I-1.5. Relevant political commitments: the Vienna Declaration on Nuclear Safety

A significant political commitment was assumed by Argentina at the Diplomatic Conference of the Convention on Nuclear Safety, which was presided by Argentina and took place on 9 February 2015, in Vienna, Austria. At that Diplomatic Conference the Contracting Parties of the Convention on Nuclear Safety, adopted the *Vienna Declaration on Nuclear Safety* [1] on:

- principles for the implementation of the safety objective of the Convention to prevent accidents and mitigate radiological consequences, and
- decisions on how to make this implementation accountable through the national report to the CNS review meetings.

Argentina has included such information in its latest national report¹³ Agreement on international safety standards: the IAEA safety standards

Argentina is full member of the international organizations that cosponsor the international nuclear safety standards being established under the aegis of the International Atomic Energy

¹³ Please visit, <http://www.arn.gov.ar/es/informes-y-documentos/informe-nacional-de-seguridad>

Agency (IAEA). Argentina has supported the development of these standards and shared their endorsement by the policy making organs of the cosponsoring organizations.

A primary standard is IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [2], which is cosponsored by the European Atomic Energy Community,, the Food And Agriculture Organization of the United Nations, IAEA, the International Labour Organization, the International Maritime Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization, the United Nations Environment Programme, and the World Health Organization. The principles were established by the IAEA’s Board of Governors in September 2006 with the concurrence of Argentina. It follows that Argentina fully adhere to these Principles.

It has been recognized that the Fundamental Safety Principles [2] were developed for prospective planned situations. Some of its principles could be construed as applicable to implementing safety improvements at existing nuclear power plants.

Argentina assigns special relevance to the safety objective established in SF-1 [2], as follows:

- To control the radiation exposure of people and radioactive releases to the environment in operational states;
- To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source, spent nuclear fuel, radioactive waste or any other source of radiation at a nuclear power plant;
- To mitigate the consequences of such events if they were to occur.

Requirement 12 of IAEA SSR-2/2 (rev 1) [3], Safety of NPPs: Commission and Operation states:

“Systematic safety assessment of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant’s operational lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.”

Noteworthy, this requirement on existing plants is not focused on safety objectives.

In terms of the treatment of existing facilities regarding the assessment of improvements, there is relevant work within IAEA Safety Guides: SSG-25 [4] provides recommendations on how to perform Periodic Safety Reviews on nuclear power plants in a systematic and regular basis. Some Member States, including Argentina, use this Safety Guide to support justification for reviewing operation licenses of NPPs periodically and furthermore, for a safe long term operation.

I-1.6. Approaches to the safety of existing facilities

It is a common practice to customize the SSG-25 [4] content to the particular regulatory environment of each country. In this regard, ARN has gained experience and built an understanding about the need of defining a “focus” to overcome to some paragraphs of the guide’s structure. It is presented a methodology of many safety factors (14), and some definitions need to be more accurate in order to avoid overlaps and “space” for exceptions.

One example of the above mentioned possible focus are the sections of SSG-25 [4] that aim at identifying safety improvements to overcome the gaps with respect to current standards and settling priorities for implementation:

- SSG-25 [4] proposes the organization of the implementation of necessary improvements as a project, which is an essential feature for the practical pursuing of safety objectives.
- SSG-25 [4] shows neither explicit nor exhaustive acceptance criteria, which, as mentioned, lie within the legal competence of national authorities.

It is worth recalling that the regulatory approach applied to the end of the expected “Lifetime” of an NPP is different for each country. The cornerstone is the concept of “licensing basis,” as the main element of the “demonstration of safety” at the design stage, and is projected to the requirements on construction, commissioning, operation:

- There are countries that use the concept of “License Renewal,” based on keeping for an installation the same licensing basis (*) of the construction license, identifying potential replacements needs by “aging programs” and “evaluation of actual condition” of equipment and components. The safety improvements might be implemented in a framework of “reasonably practicable.” This may be understood as “Extending the Life” by “Extending the initial license.”
- There are countries that use the concept of “Long Term Operation”, based on the review (update to standards) of the licensing basis of an installation, identifying the modifications needed to proof that the safety assessment with current standards is successful. These modifications are “added” to the interventions needed for a License Renewal.

Argentina has defined the maintenance of the licensing basis is a permanent goal within two PSRs and requires the review and update of the licensing basis every ten years based on performing a PSR.

Finally, it is worth mentioning that there has been an international effort in the aftermath of the Fukushima Daiichi NPP accident, by the implementation of “stress tests” on existing nuclear power plants. These were programmes for identifying engineering features that would contribute to the mitigation of severe accidents in a specific NPP and also for assessing the applicability of specific plant modifications to incorporate features that had already been identified for other plants. In some cases, these programmes were outlined at a regional level, as was the case of the stress tests project carried out by the FORO (Forum of Ibero-American Regulators of Nuclear and Radiological Safety).

More details on the approach of ARN on the compliance with safety objectives in existing plants are presented hereinafter.

I-1.7. Identification of safety improvements

I-1.7.1. Driver for the enhancement process

Argentina shares the ample international consensus on advancing to ensure that safety objectives are an essential part of criteria and standards in all member states (MS), and this has been made accountable in the national presentations to the Convention of Nuclear Safety Review Meetings

However, for ruling over **existing facilities** (already licensed) the elaboration of these national criteria and standards cannot be derived from the international safety standards at the level of setting technical requirements. These standards have been developed and established for **prospective** situations. Requirements that affect facilities already licensed would be **retrospective** in nature. They would imply a change in the approach in the established standards.

Nonetheless, ARN understands that there is still an effective framework to pursuing safety objectives in retrospective assessments. This is possible if it is accepted that the focus of the Periodic Safety Reviews (PSR) of existing plants has to be accountably oriented to the safety objectives by a global assessment. ARN understands that it is the Regulatory Body of each country who is responsible for determining how safety objectives are accomplished in existing plants, in terms of procedures, requirements, criteria and standards.

Selection process of safety improvements

ARN view of SSG-25 [4]

The assessment of an existing plant in terms of the compliance with safety objectives requires as an absolute pre-condition that the assessment be of the plant as a whole. I.e. the assessment has to be global and/or integral, and this implies a safety analysis as the one applied to a “new” project and presented in the Safety Analysis Report, plus the assessment of actual condition of systems structures and components.

The possibility of focusing within SSG-25 [4] for producing an integral assessment is reflected in the following paragraphs quoted from the guide:

As defined in SSG-25 [4] INTRODUCTION, the periodic safety review (PSR) is a means:

- “...to assess the cumulative effects of plant ageing and plant modifications, operating experience, technical developments and siting aspects...”;
- “...To include an assessment of plant design and operation against applicable current safety standards and operating practices...”;
- “...To ensure a high level of safety throughout the plant’s operating lifetime...”;
- “...PSR is complementary to the routine and special safety reviews conducted at nuclear power plants and does not replace them...”. SSG-25 [4]

In the section RATIONALE, OBJECTIVE AND GENERAL RECOMMENDATIONS FOR PERIODIC SAFETY REVIEW:

- “A PSR may be used in support of the decision-making process for license renewal or long term operation, or for restart of a nuclear power plant following a prolonged shutdown.”
- “In many States, PSR forms part of the regulatory system, though the scope and content of the PSR, the manner of its implementation and the regulatory activities relevant to the PSR vary depending on national regulations.”

In this sense, ARN applied this approach for granting a license every 10 years and as an input for LTO.

- “PSR provides a mean for regulating the safety of plant operation in the long term and for addressing requests by licensees for authorization to continue plant operation ...”

ARN uses the PSR as a base for facing the LTO.

In section §3.5, the PSR is a means for LTO input:

- “It could be used as support the decision-making process prior to entering long term operation”
- “Detect safety improvements to ensure that the licensing basis remains valid and in accordance to updated standards during the period of LTO”
- “Improvements might include refurbishment, the provision of additional SSCs and/or additional safety analysis and engineering justifications...”

In section 4, REVIEW OF STRATEGY AND GENERAL METHODOLOGY, the global assessment should consider:

- positive and negative findings from the PSR;
- the corrective actions proposed;
- safety improvements proposed;
- the assessment of the overall level of safety.

Where there are negative findings, the global assessment should provide a justification for any improvements that cannot reasonably and practicably be made.

ARN understands that in case of a plant facing the LTO, in addition to the global assessment, the PSR will enlarge the scope of the analysis to other aspects such as ageing and obsolescence.

Paragraph 4.22 states:

“The level of plant safety should be determined by a global assessment reflecting, among other things, the combined effects of all safety factors. It is possible that a negative finding (deviation) in one safety factor can be compensated for by a positive finding (strength) in another safety factor.”

ARN considers that the effectiveness of safety improvements is demonstrated by a documented Global Assessment, later reflected by an updated Safety Analysis Report (SAR). Negative findings in terms of safety shortcomings are to be coped with by engineering, technological or administrative means.

In section 6 GLOBAL ASSESSMENT:

“The objective of the PSR global assessment is to arrive at a judgement of the nuclear power plant’s suitability for continued operation on the basis of a balanced view of the findings from the reviews of the separate safety factors.”

ARN states that the final balance or integral result should always be positive, weighted with consideration of the Defence in Depth levels:

- “The global assessment should consider all the findings (+/-) from the separate safety factor reviews and should consider what safety improvements are reasonable and practicable.”
- “The global assessment should also consider overlaps and omissions between the separate safety factors and determine whether additional or grouped safety improvements arising from more than one safety factor review are also reasonable and practicable.”

Regarding these two quotes, ARN understands that an objective safety-graded classification and prioritization of identified improvements should be developed:

- “Identified safety improvements judged not to be reasonable and practicable should not be pursued.”

ARN understands that the discussion of safety improvement for the mitigation of very low probability scenarios should be produced by accountable judgements.

Regarding classification of outcomes, paragraph 6.7 states:

- “A method for assessing, categorizing, ranking and prioritizing safety improvements to address negative findings should be established,”
- “The method should be based on the safety significance within the global assessment,”
- “The approach adopted could be based on deterministic analysis, PSA, engineering judgement, cost benefit analysis and/or risk or a combination thereof.”

ARN understands that this cost benefit analysis is essentially related with the fundamental radiological protection principles of justification and optimization and should not be confused with a business-oriented approach.

I-1.8. Detailing safety improvements

Regulatory assessment based on SSG-25 [4]

An example of this use for identifying outcomes of safety improvements are the result of the assessment of Atucha I NPP. Atucha I NPP is a PHWR in operation since 1974. A new operating license was issued on 2014. It involved a first use of SSG-25 [4] aiming at a global and integral safety assessment. The assessment of the 14 safety factors gave specific outcomes and updates on the safety report. The outcomes identified of safety improvements are presented in the following section.

All plant modifications may be justifiably described as safety improvements, but a definition is needed on how they could be graded and selected, pointing to implementation.

The first step is to classify each modification based on the specific safety functions it involves and in which Defence in Depth level it has impact.

The prioritization based on:

- Deterministic criteria (comparing plant response with the improvement implemented or not)
- Probabilistic criteria (risk level improvement – mainly full power PSA level 1)
- Implementation consequences (risk as a subjective probability of incurring a range of dose)

The three bullets above are linked to the DiD levels starting from level 3.

As general remarks to Atucha I PSR:

- The treatment of all gaps is desirable
- For high priority findings, a project for implementation is required in the short term (months, implementation may imply outage).
- For low priority findings, a medium or long term implementation project is acceptable (several months, a few years).
- All shortcomings need to be treated before licenses issuance of LTO or a new license.
- Some findings were also consistent with post-Fukushima stress-tests actions already identified as needed.

I-1.9. Outcomes identified of safety improvements

This section will present a partial summary of the identification of improvements as an outcome of the assessment and the definition of priorities on the outcomes of the PSR on Atucha I NPP performed in 2014 is summarised in Table I-1.

As a first order approach, a **High** or **Low** relevance or priority has been selected.

TABLE I-1. Definition of priorities on the outcomes of the PSR on Atucha I NPP

| Related Safety factor | GAP Description | DID | Priority |
|------------------------|--|-------|----------|
| Probabilistic Analysis | PSA update considering plant modification | 3,4 | H |
| | Fire PSA | 3,4 | H |
| | Low power and shutdown PSAs | 3,4 | L |
| Deterministic Analysis | Update all safety analysis with plant modification | 3 | H |
| Operation | Update of SAMG procedures | 4 | L |
| Design provisions | Review of the methodology for classification SCC's based on safety functions | 3 | H |
| | Review and technical justification of all SCC unavailability time | 1,2,3 | H |
| | Consequential Failure analysis in safety system | 3 | H |
| | Review of the main control room habitability and absence of the emergency control room | 4 | L |
| | Review of the main control room habitability | 3 | H |
| | Filtered venting | 4,5 | L |
| | Strategy for cooling the reactor vessel | 4,5 | L |
| | Improvements in the Fire detection system and mitigation | 3 | H |
| | Improvements needed in the reactor protection system | 3 | H |

I-1.10. Conclusions

In a very short summary, ARN understands that the regulatory treatment of existing NPPs in order to keep an integral assessment in terms of safety goals is viable by an adequate national approach of rules and requirements, within the framework of the IAEA safety standards. This is possible if it is accepted that the focus of the Periodic Safety Reviews (PRS) of existing plants has to be accountably oriented to safety objectives by a global assessment. The Regulatory Body of each country who keeps the responsibility on how safety objectives are accomplished in existing plants, in terms of procedures, requirements, criteria and standards.

It is necessary to make an effort to treat the safety improvements in the frame of a global assessment of the plant as a whole. This is recognized as a “nuclear policy” issue beyond the facts that some tasks may contribute to safety, but only partially:

- Individual modifications can be assessed as safety improvements pursuing safety objectives.
- The use of the PSR guide SSG-25 [4] by assessing separately the 14 safety factors produces useful information on the actual condition of a plant.

Previous information of this annex shows that ARN regulatory approach to existing Nuclear Power Plants is based on a mature use of international standards, complying international legally binding instruments and in line with political commitments.

I-2. ARMENIA

I-2.1. External hazard combination

Following the nuclear accident at the Fukushima Daiichi nuclear power plant on 11 March 2011, the Armenian Government emphasized the need for urgent actions to reassess the preparedness of Armenian Nuclear Power Plant to respond to emergencies. In June 2011, ANRA required to conduct an in-depth reassessment of the ANPP safety in the light of Fukushima Daiichi NPP accident (stress-tests). The stress-test national report covers hazards correlated with earthquake, flooding, extreme weather conditions, Loss of electrical power and loss of ultimate heat sink and Severe accident management as it is recommended in the ENSREG and the EC. The analyses of external hazard possibilities have been mainly performed during design stage (reactor design and fuel storage design) and later in the frame of external hazard PSA implementation. In addition to the stress test analyses Armenian Nuclear Regulatory Authority TSO (NRSC) together with Armenian NPP launched a co-operation project with IAEA aimed to perform a complementary analysis of plant robustness by qualitative assessment of potential impact of external hazards and their credible combinations. Some result of this study was included in the stress-test national report. The Stress Test Peer Review Team characterize this as a strength and recommended to complete the IE of ANPP Level 1 PSA with some hazard combinations.

Systematic assessment of an NPP's response to extreme events, with a focus on long term accident progression and the identification of cliff edge effects with the potential to affect the provision of important safety functions and their associated support functions (alternating and direct current power supply, essential service water), is usually beyond the scope of the licensing basis. Plant systems (both normal operation and safety classified systems) have usually been assessed mainly against design basis accidents, including certain postulated external and internal hazards. However, for certain design extension conditions, success paths to perform safe plant shutdown and maintain the reactor in a safe state may exist due to margins embedded in the design of safety-related systems, structures and components.

To analyse the impact of external hazards and their credible combinations the Fault Sequence Analysis (FSA) method was used. This method makes possible to assess the plant protection against the extreme events including combined hazards and long duration accident sequences using outputs of the existing deterministic and probabilistic analyses and identifies critical combinations of failures that could lead to core damage. The FSA method represents a 'what-if' analysis. It combines the probabilistic safety assessment (PSA) logic (minimal cutsets from Level 1 internal initiating events PSA) and operability limits of systems, structures and components in regard to the impact of external hazards (extreme maximum/minimum temperatures, flood level, seismic capacity) to identify critical fault sequences including components failures and human errors that might be caused by postulated external hazards or their combinations.

The method focuses on the analysis of minimal cutsets generated in an internal initiating events Level 1 PSA using minimal cutsets concept. Therefore, a minimum prerequisite for the use of the FSA method is the availability of a Level-1 internal initiating events PSA of high technical quality and sufficient level of detail. The PSA's logical models is used to analyse the fault sequences that could occur due to extreme event. The method comprises five major analysis steps illustrated in Fig. I-2.

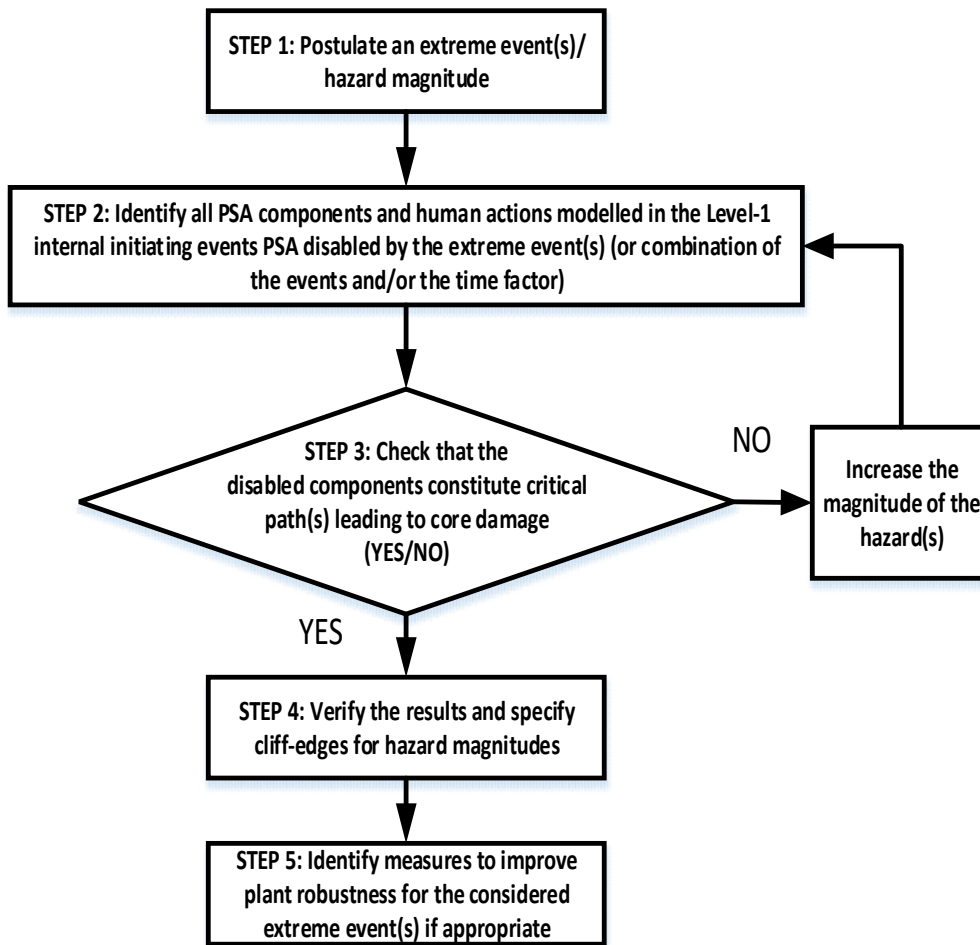


FIG. I-2. Fault sequence analysis steps.[5]

As it could be seen from Fig. I-2, FSA method results in identification of measures to improve the plant safety based on systematic evaluation of the information already available at the plant (in most cases). Although experience shown that implementation of FSA method could highlight necessity for additional analysis that were not performed at the plant before.

The process of FSA method application for ANPP started with the selection of external hazards for further consideration. The selection of hazards was implemented based on already performed studies. After the list of hazards has been identified, the PSA model was checked from the viewpoint of FSA method application. During PSA model verification several deficiencies in the existing ANPP PSA model were revealed that creates obstacles for efficient application of FSA method. Therefore, special efforts were devoted for additional supporting analysis (e.g. elimination of over conservatism in Loss of Offsite Power (LOSP) accident progression model) and corresponding PSA model upgrade. Once the ANPP PSA model was finalized, a list of PSA elements has been completed. The list of PSA elements includes basic events, which represent equipment failures, human errors and Common Cause Failures. The next step implied identification of operability limits for each PSA element in the list. Data collection for operability limits was performed based on plant design documentation, available safety assessments and expert judgment. During the process of data collection, it was identified that some information is not available and additional analytical support is required (analysis of plant ventilation system's failure impact, analysis of feasibility to perform certain human actions in case of external hazards). The additional analyses allowed to finalize the input deck

for Fault Sequence Tool for Extreme Events (FAST-EE) software. The FAST-EE software allows to efficiently utilize the qualitative information obtained from an internal initiating events Level-1 PSA (i.e. minimal cutsets), information on the operability limits of structures, systems and components, and the feasibility of operator actions under different severe conditions caused by extreme events. The input deck of FAST-EE software consists of the following attributes:

- Minimal cut-sets associated with LOSP initiating events
- Assignment of basic events appeared in LOSP related Minimal cutsets to the plant buildings
- Basic events' and plant buildings' operability limits defined for each external hazard

After that several calculations for selected cases were performed. The results of performed analyses allowed to come up with several conclusions and recommendations related to robustness of ANPP against external hazards and their credible combinations using the FSA method and possible measures aimed at enhancing plant protection against extreme events.

Depending on the results of the FSA the following combinations of external hazards could be challenging for ANPP safety.

1) Seismic event during long lasting period with high temperature. Long lasting high air temperature could not lead to significant challenge of ANPP safety if ventilation systems are operable, however in case if at the same time seismic event could affect ventilation systems and lead to temperature increase in switchgear compartments and consequential loss of offsite power. In such a scenario switchgear equipment could become unavailable and even DGs are operable power supply to consumers will not be possible. In such scenario the only possibility to maintain reactor cooling function is to use diesel pump located in boron unit of Unit #1 for feedwater supply and SG safety valves (that have high seismic resistance and could be operated without electrical power supply). Therefore, it is important to assure operability of diesel-pump and SG safety valves in case of high air temperature and seismic event.

2) Seismic event during long lasting period with low temperature. Long lasting low air temperature could lead to loss of offsite power accident with unavailability of emergency cooldown system and diesel driven pump. In such case operability of DG is considered to be critical for maintaining reactor cooling. Low air temperature could not affect DG operability by itself. The most vulnerable DG equipment is the local diesel fuel tank. Freezing of the fuel in the tank or associated pipes could lead to DG failure. Diesel fuel tank is located in DG building where the heating system maintains the air temperature above 50°C. However, if heating system fails the freezing of diesel fuel lines could lead to failure of diesel generators. Heating system failure could occur in case of seismic event. Thus, seismic event during long lasting period with low temperature could be challenging combination of external hazards for ANPP safety. However, it is necessary to mention that such scenario could occur only if a seismic event happened within the short time when the impact of low temperature on the emergency cooldown system and diesel driven pump is not detected. Therefore, the likelihood of such combination is considered to be low.

3) Low temperature and heavy snow load. Long lasting low air temperature could lead to loss of offsite power accident with unavailability of emergency cooldown system and diesel driven pump. In this situation the emergency feedwater system is the only possibility to supply SG and perform reactor cooling function. However, if low temperature will be combined with snow cover formation then the heavy snow load has the potential to affect roof of turbine

building with consequential failure of emergency feedwater system. In such scenario there is still last possibility to maintain reactor cooling function using primary feed & bleed procedure with the electrical power supply from DGs. Hence for this scenario operability of DG is considered to be critical. DG building has high capacity in terms of snow load, therefore it is also important to assure operability DGs in case of extremely low air temperature.

4) High wind associated with the increase of dust concentration. According to the analysis performed, high wind could lead to loss of offsite power. When high wind occurs in the summer time it is possible to have significant increase of dust concentration in the air. Increase of dust concentration in the air could lead to failure of DGs due to blockage of the DGs air intake. Failure of DGs due to wind induced dust will lead to station black-out. In such scenario there is still a possibility to perform reactor cooling function using diesel pump located in boron unit of Unit #1 and SG safety valves that could be operated without electrical power supply. Therefore, it is important to assure operability of diesel-pump and SG safety valves in case of high wind and high dust concentration.

I-3. BELGIUM – CURRENT APPROACH TO DESIGN EXTENSION CONDITIONS’ ANALYSIS FOR EXISTING BELGIAN NUCLEAR POWER PLANTS

I-3.1. Purpose and background

Presentation of approach used by the Belgian utility ENGIE to analyse Design Extension Conditions (DEC) taken into account for existing plants. Approach describes how DEC are defined, identified and analysed.

As defined in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1) [6], Safety of Nuclear Power Plants: Design, DEC are:

“Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best-estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with core melting”.

In addition, Requirement 20 of SSR-2/1 (Rev. 1) [6] states that:

“These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”

I-3.2. Definition of design extension conditions

In Belgium, accident studies that go beyond the design basis of the nuclear power plants and that could be assimilated to DEC, have already been implemented since years i.e. in the framework of the first decennial revision. It includes also severe accident analyses e.g. Belgian NPPs have implemented SAMG since the 90’s as well as Passive Autocatalytic Recombiners (PARs).

Moreover, PSA Level 1 and 2 studies for internal events and internal hazards (fire and flooding) have also been implemented for the reactor core. Procedural as well as hardware modifications on sites have been performed based on those studies as well.

I-3.3. Objectives of DEC’s analysis

There are three objectives as per the WENRA RL 2014 [7] i.e.:

- DEC studies are performed first to assess and verify that features have the capacity to perform in DEC i.e. compare analytical results to defined safety criteria;
- DEC studies will enable to assess that accident procedures are “fit for purpose” in case of DEC;
- DEC studies will provide the DEC in the sense of environmental conditions to which SSC have to be qualified or survive e.g. survivability assessment of existing instrumentations needed to perform in case of DEC B, evaluation of potential on-site and off-site radiological consequences in DEC B.

Recently in the frame of the Stress Tests and dedicated Action Plan, new provisions have also been implemented. Examples of new and coming provisions that do cope with WENRA RL F (list not exhaustive):

- Mobile means in Doel NPP to inject water to the SG, primary circuit and SFP in case of extreme external hazards (seism, external flooding);
- Fixed means in Tihange NPP to inject water to the SG, primary circuit and SFP in case of extreme external hazards (seism, external flooding);
- Wall against external flooding for Tihange NPP;
- Several upgrades done in procedures (EOPs and SAMG) for Doel and Tihange to cope against internal events, fire, internal flooding, external flooding, seismic events and airplane crash including in the case of a severe accident;
- Filtered Containment Venting System (FCVS) installed in Doel and Tihange NPP;
- Alternative sprays installed in Doel NPP and coming in Tihange NPP;
- Direct cavity injection coming in Tihange NPP.

DECs identification and classification

In the past, deterministic criteria for selection of BDBA were used (e.g. SBLOCA with loss of HPSI or LPSI (Class III accident + CCF)). During the Stress Tests, a set of DEC was derived based on engineering judgements and on analyses of potential cliff-edges for what concerns external hazards.

Currently, based on the WENRA RL 2014, the selection of DEC is mainly based on PSA, probabilistic criteria and expert judgment. Elaboration of a list of credible Combination of Events in DEC applicable to both Tihange and Doel sites is ongoing at ENGIE based on WENRA RL 2014 and associated guidance (Issue T).

Definitions for controlled state and safe state following design extension conditions

In the case of DEC A, the analysis stops when the operator has taken back control of the plant. Typically, the stable and controlled state is the Hot Zero Power. This is not of application for all analyses as some analyses are by e.g. PSA-based, with the success criteria for DEC A being absence of core melt.

The end-state of DEC B is clearly defined by the set of criteria that needs to be fulfilled to declare the end of the severe accident emergency phase and exit the Severe Accident Management Guidance. This set of criteria includes the following parameters that need to remain below a defined threshold that can be unit specific (cf. SAMG): Core Exit Temperature, dose rate to the population and/or activity released at stack, containment pressure and H₂ concentration in the containment.

Approach for identification and definition of DECs

WENRA RL (2014) F2.1 has been taken into account in the methodology to derive a comprehensive DEC list, which used both engineering judgement and probabilistic arguments for establishing the lists for internal events, for external hazards and for combination of events (CoE). In addition, some sequences have been selected deterministically (SFP, CoE, stress tests...) and are considered in the elaboration of the DEC list. Moreover, elements considered in WENRA RL (2014) F2.2 have been taken into account in the methodologies for establishing the lists i.e.:

- Internal events have been taken into account, including CCF, failures on one layer of defence. as the methodology uses PSA as input;
- Combination of events have been taken into account including internal events and hazards as well as external hazards;
- All the relevant POS have been taken into account in particular for internal events and for combination of events;
- A screening of the relevant external hazards has been performed in the frame of WENRA RL (2014) Issue O and a dedicated approach deals with elements associated to WENRA RL (2014) Issue T

Eventually, the different contributions from all those sources have been merged in a single list called Merge of the Lists DEC. Finally, WENRA RL (2014) F2.3 is implicitly taken into account in the elaboration of the DEC B list. The selection of DEC B scenarios uses the PSA level 2 and the stress tests as input, where various systems are assumed to fail leading to core melt scenarios. The retained scenarios are selected based on engineering judgement.

Use of best estimate approach for analysis of DEC

The following approach aims to address WENRA RL (2014) F3.1.

In particular for RL F3.1 (a): “rely on methods, assumptions or arguments which are justified, and should not be unduly conservative”, the following principles will be applied:

- Studies will rely on best-estimate codes and assumptions regarding the physical phenomena considered, taking into account the international REX;
- Regarding the treatment of uncertainties, both conservative and best estimate approaches will be applied. It is the intent to start from studies with enveloping boundary conditions in order to obtain enveloping results

How to deal with uncertainties within the approach

The approaches considered aim to address (requirement relative to) WENRA RL (2014) F3.1(b) “be auditable, paying particular attention where expert opinion is utilized, and take into account uncertainties and their impact”

In DEC A, the proposed approach to deal with uncertainties is to start from the licensing calculation of the parent DBA (the parent DBA is typically the DBA with the same PIE as the DEC under consideration. For example, for the DEC A “SBLOCA with failure of HPSI”, the parent DBA is SBLOCA), but without considering a single failure and partly relaxing the conservatism of some assumptions. These assumptions typically correspond to additional failures with respect to the defined sequences, which are used in DBA analysis. These include:

- The one stuck rod assumption (which penalizes the anti-reactivity insertion);
- Failures of control systems;
- Non-consideration of some SCRAM signals.

Licensing calculations are indeed performed taking uncertainties into account, penalizing all relevant parameters. Sensitivities are performed for parameters of which the impact is not obvious. The licensing case is the most penalizing combination. This approach would therefore correspond to the use of a Best-Estimate code with conservative assumptions on the initial and boundary conditions, which is the main approach used for licensing studies in Belgium. This corresponds to the second possible option in the IAEA SSG-2 [8] guide devoted to DSA5, the so-called “combined analysis”.

However, the set of parameters on which conservatisms are applied in the parent DBA calculation might not be relevant for the corresponding DEC study. In this case, another set of relevant parameters to be penalized needs to be selected. For example, for studies where recirculation is not possible, the level in the RWST is an important parameter that is typically not used in licensing studies, which would have to be penalized.

If the safety criterion is not met with this approach, a more sophisticated approach could then be used, as for example Best-Estimate Plus Uncertainties (BEPU). This corresponds to the third possible option in the IAEA SSG-2 [8] guide devoted to DSA.

In DEC B, the uncertainties are taken into account by voluntarily penalizing the sequences considered to perform the deterministic analysis required by WENRA RL Issue F as for example the calculation of the radiological consequences or the calculation of the time available to install mobile means foreseen in DEC B. This corresponds again to the second possible option in the IAEA SSG-2 [8] guide devoted to DSA, the so called “combined analysis”. This approach has for example been applied for the definition of the design parameters of the Containment Filtered Venting System installed in Belgium (CFVS).

Cliff edge effects assessment within the approach

The approaches considered here below aim to address (requirement relative to) WENRA RL (2014) F3.1(f) “demonstrate, where applicable, sufficient margins to avoid “cliff-edge effects” that would result in unacceptable consequences; i.e. for DEC A severe fuel damage and for DEC B a large or early radioactive release”

For DEC A, the evaluation of the margin to cliff-edge will be based on deterministic evaluation, based on the sequences selected in the DEC list (cfr above so-called “Merge of the Lists DEC”). This will be done for each of the cliff-edges that have been identified i.e.:

- Long term Fuel Assemblies (FAs) uncover (without immediate possibility of recovery);
- Unacceptable rate of Departure of Nucleate Boiling (DNB);
- Return to criticality.

Additional verification is performed for the sequences in the DEC list related to Containment Failure (CF) to fulfil WENRA RL (2014) F4.11.

For each of these cliff-edges, bounding sequences have been selected within the DEC list. However, several sequences may be selected for each criterion as it was not possible to determine the most bounding sequence only based on engineering judgment.

The approach is not the same for each cliff-edge. For instance, DNB is a particular case as it is a short term concern. The cliff-edge in this case would therefore be to have an unacceptable percentage of Fuel Assemblies entering DNB. The sequences for which DNB is a concern are

the Anticipated Transient Without SCRAM (ATWS) sequences, the boron dilution sequences as well as the Steam Line Break (SLB) with failure of Main Steam Isolation Valve (MSIV). The uncertainties would be taken into account following the approach mentioned above. For this particular criterion, the margin to the cliff-edge will therefore be naturally contained in the evaluation of the DNB Ratio (DNBR). The same approach can be used for criticality accidents. The quantitative criterion will be the same as that of the Class IV parent DBA. If this criterion cannot be met, a DEC criterion will have to be proposed in agreement with the safety authorities. In any case, the Class IV criterion bounds the DEC criterion.

For internal events sequences that have been selected based on the core uncover criterion, the margin to the cliff-edges will be defined as the maximum time before the operator starts the Emergency Operating Procedures (EOPs) and still avoids core damage taking into account realistic operator action timing.

A criterion for evaluating “acceptable” grace time could be to use the value of 30 minutes used for DBAs before starting the EOPs. If this criterion cannot be met, a DEC criterion will have to be proposed in agreement with the safety authorities.

For the CF aspect, the same approach as for DNB is proposed, i.e. starting from the parent licensing calculation without considering a single failure. The quantitative criterion will be the same as that of the Class IV parent DBA.

The “approach” is mainly based on the criteria used for the equivalent Class IV accidents. DEC criteria may need to be introduced if the Class IV criteria are too restrictive (i.e. if the Class IV criterion cannot be respected for the DEC analysis). This criterion would be bounded by the Class IV criterion, i.e. respecting the Class IV criterion implies respecting the DEC criterion.

For DEC B, the cliff-edges that will be considered for the analysis performed in the frame of WENRA RL (2014) [7] F3.1(f) are:

- Direct Containment Heating (DCH) leading to CF;
- Steam explosion leading to CF;
- Hydrogen burn leading to CF;
- Long term static pressurisation leading to CF (LTSOP);
- BMMT.

The method proposed here is an extension of what is already used in several beyond design studies worldwide to assess the acceptability of a situation regarding the pressure load in the containment. For those types of studies, it is possible to use the ultimate pressure as a criterion to which loads calculated in a deterministic way are compared and was documented in the “State of the art on hydrogen passive autocatalytic recombiner,” 2002) [9]. This is equivalent to defining an acceptable probability of exceeding the resistance of the containment since the fragility curve is based on the ultimate pressure. For instance, the median containment failure pressure in the “Probabilistic Analysis of Containment Structural Performance in Severe Accidents,” is set at 110% of the ultimate pressure. [10]

In that case, the criterion (resistance of the containment) is of probabilistic nature while the loads are deterministic. The margin can then be calculated by subtracting the load value to the criterion value.

SSCs credited in the analysis, analysis end-state

Approach defined to deal with WENRA RL (2014) [7] F3.1(i) "define an end state, which should where possible be a safe state, and, when applicable, associated mission times for SSCs":

For DEC A, the end-state depends on the scenario, as the available SSCs depend on the scenario. The possible end states are the following for the reactor core:

- Cold shutdown with RHRS connected;
- Long term Safety Injection (SI) recirculation;
- Intermediate shutdown (RHRS connecting conditions) with long term SG cooling.

For the SFPs, the possible end states are the following:

- Long term cooling by Pool Loop (PL) pumps;
- Long term cooling by feed and bleed of the SFP (e.g. in case loss of integrity of the SFP).

Different SSCs are needed to reach the defined end-state. For each sequence, the list of required SSCs is established.

A distinction between the SSCs needed to reach the end state and those needed to maintain the end-state is made. The mission time depends on the category of the considered SSC.

Example of application - DEC A

An example of DEC A sequence is the "Feedwater Line Break Inside Containment (FWLBIN) with failure of Feedwater/Auxiliary Feedwater/Emergency Feedwater (FW/AF/EF) and Residual Heat Removal System (RHRS)". Since the RHRS and the SGs are lost, the only way to avoid core damage is to use feed and bleed of the primary circuit. The associated end state is long term recirculation.

The required SSCs for this strategy to succeed are SI and pressurizer Relief Valve (RV) (as well as their support systems).

I-4. BELGIUM – ENGIE ELECTRABEL MITIGATION STRATEGY REGARDING SEVERE ACCIDENT (DEC B) FOR THE BELGIAN UNITS

I-4.1. Purpose and Background

The focus for the Mitigation Strategy is to identify the required means and actions to be compliant with IAEA NS-G-2.15¹⁴ and new WENRA RL 2014 [7], which can be considered as a pragmatic approach.

The main objectives of this Mitigation Strategy are the following:

- minimize the releases towards the environment/population
- achieve long term stable state

The translation of this objective into operational parameters is to meet the current requirements of the EPP as defined for the Belgian NPP (i.e. evacuation/sheltering till 10 km; KI tablet distribution till 20 km), even in case of a fast-evolving SA.

In order to even decrease the burden that is associated to the use of a Containment Filtered Venting System (CFVS), strategies that allow to avoid, delay or end the use of the CFVS shall also be developed, and their practicable character shall be evaluated.

After having implemented the EBL Mitigation Strategy, the robustness of both Doel and Tihange NPP against SA (also called Design Extension Conditions (DEC) B scenarios in WENRA 2014) will be further increased and the impact on the environment/population of such an event drastically decreased (for compliancy to WENRA 2014 RL F [7]).

I-4.2. Strategy

Table I–4 highlights the major elements of the Mitigation Strategy for Doel and Tihange NPP along with the means to cope with the different containment failure (SA) modes:

¹⁴ INTERNATIONAL ATOMIC ENERGY AGENCY, Severe Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.15, IAEA, Vienna (2009).

TABLE I-4. MITIGATION STRATEGY FOR DOEL AND TIHANGE NPP

| Important containment failure modes (as highlighted in Belgian PSA2 studies) | Mitigation Strategy | |
|---|--|---|
| | Doel | Tihange |
| Molten Core Concrete Interactions (MCCI) / Basemat Melt Through (BMMT) | <p>In Vessel Melt Retention (IVMR) by Ex-Vessel Cooling (EVC) excludes MCCI/BMMT</p> <p>IF IVMR by EVC fails, strategy transposed to wet cavity i.e. quenching of corium by water present in cavity at moment of vessel failure</p> <p>Use of Alternative Spray System and/or gravitational drain of RWSTs along with cavity flooding device (all means already available)</p> | <p>Dry cavity (before vessel failure) along with top flooding of the corium</p> <p>Use of Direct Cavity Injection System (DCIS)</p> <p>Middle/Long term: Recover water from sumps above 'Very High Level Containment' for the cooling of corium when stopping DCIS (i.e. to avoid WWCCCL) - use of a connection sumps-cavity at Very High Level Containment</p> |
| Slow containment pressurization | <p>Use Alternative Spray System (ASS)</p> <p>(Use of DCIS Containment (CNT) has positive impact on this threat as DCIS injection flow rate is higher than the flow rate relative to the decay heat)</p> <p>Use of Containment Filtered Venting System (CFVS) as ultimate mean</p> <p>Concurrent use of Alternative Spray System and CFVS to have a maximum positive effect on radiological releases</p> <p>Evacuate the residual heat with a dedicated SA mean w/o opening the containment</p> | |
| Ex vessel steam explosion | <p>IVMR by EVC excludes steam explosion</p> <p>IF IVMR by EVC fails, threat is anyway reduced (strategy AFARA)</p> | <p>Dry cavity (before vessel failure) excludes steam explosion</p> |

The list below highlights the means to implement on sites as a prerequisite of the application of the ENGIE Electrabel Mitigation Strategy, along with complementary suggested actions. Items are categorized in three groups i.e. if they are common to both sites or if they are applying to Tihange or Doel.

Means and actions, which are common for both sites:

- Increase reliability of the RCS depressurisation;
- Install Alternative Spray System along with adequate auxiliaries (e.g. qualified containment water level measurement and indicator);
- Install CFVS along with adequate auxiliaries (e.g. qualified containment pressure measurement and indicator);
- Launch a pre-feasibility study to evaluate the necessity to install a dedicated mitigation system to enable (long term) evacuation of the decay heat, without opening the containment;
- Launch a feasibility study to analyse the possibility to recover (non) conventional SSC for (long term) evacuation of the decay heat without opening of the containment.

Specific means/actions at Tihange:

- Install Direct Cavity Injection System (DCIS) along with adequate auxiliaries – including following actions:
 - Create a connection between sumps – cavity at ‘Very High Containment Level’;
 - Evaluate precisely the maximum admissible containment water level.
- Prevent water from entering the reactor pit before vessel failure.

Specific means/actions at Doel:

- Analyse the applicability and feasibility of IVMR by Ex-Vessel Cooling – participate into R&D Project NUGENIA IVMR to confirm this strategy is a good candidate for Doel;
- Ensure timely filling of the reactor cavity.

I-5. CANADA

In Canada there are three nuclear utilities: Bruce Power and Ontario Power Generation (OPG) in the Province of Ontario, and New Brunswick Power (NBP) in the Province of New Brunswick. Bruce Power operates two four-unit stations, Bruce A and Bruce B, located on the shore of Lake Huron. OPG operates the Pickering and Darlington stations located on the shore of Lake Ontario. The Pickering station includes six units, while Darlington is a four-unit station. NBP operates the single-unit Point Lepreau station. Units are all CANDU[®] design.

I-5.1. Licensing basis of existing nuclear generating stations

The Canadian approach to reactor safety, while benefiting from approaches elsewhere, has developed independently. The characteristics of the CANDU[®] reactor relevant to severe accident are set first by the inherent properties of its design, and second by the Canadian Safety & Licensing approach. The licensing basis for Canadian NGS was established based on requirements set by the Atomic Energy Control Board (AECB). The first AECB regulatory documents were developed in the 1970s and determined the safety design approach for Canadian NGS, strongly influenced by the lessons learned from the NRX accident in 1952.

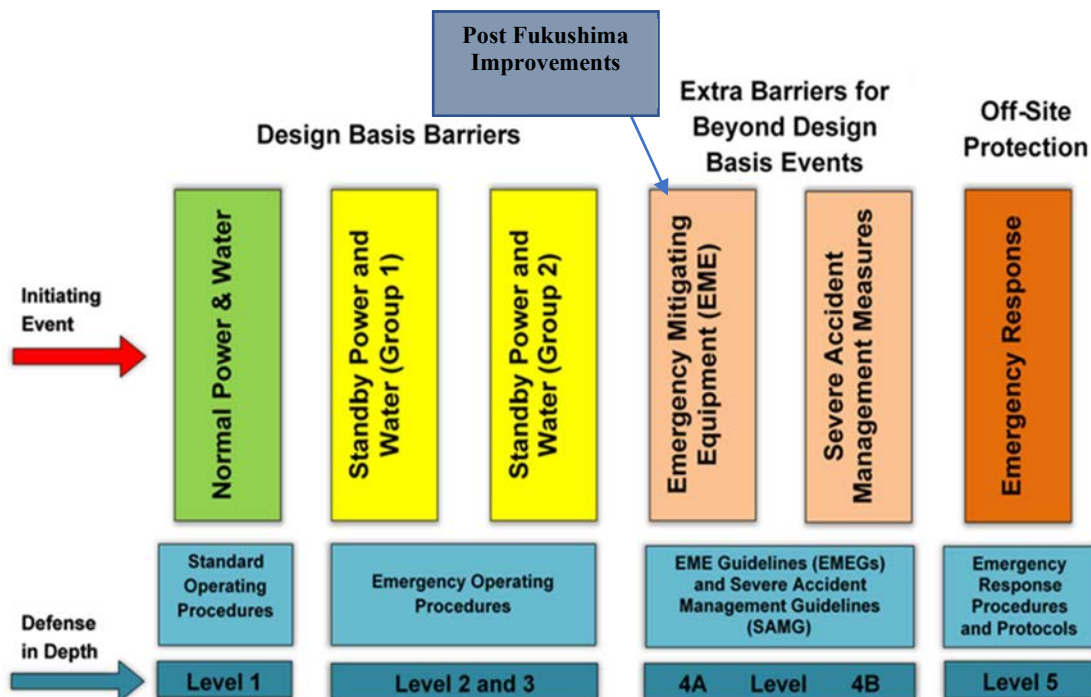


FIG. I-5. Canadian approach to reactor safety. (Courtesy of OPG)

Systems were made sufficiently independent, diverse, redundant, and separated in accordance with the two-group philosophy. Targets were established for the reliability of systems to reduce event frequency. With these provisions, the frequency of a severe accident was made acceptably low (See Figure I-5).

Adherence to deterministic safety principles in both its design and operation is the underlying basis for the robustness of the safety case of each Canadian NGS against accidents. This includes the application of the fundamental concept of defence in depth, which consists of five levels of engineered, administrative and people-based barriers to (1) prevent abnormal operation and failures, (2) control abnormal operation and detection of failures, (3) control accidents within the Design Basis, (4) control severe plant conditions to prevent accident progression and mitigate consequences of severe accidents and (5) mitigate radiological consequences of significant releases of radioactive materials.

The wide array of accidents considered in the Canadian NGS design were analysed with conservative methods and assumptions to confirm that consequences are kept within regulatory dose limits. To demonstrate that the risk to the public is reasonably low, accident-dependent public dose limits were prescribed: the more serious the consequences, the lower the tolerable frequency of occurrence. The prescribed dose limits are from the Siting Guide (1972) and from the Consultative Document C-006 (1980), a modernized version of the Siting Guide. Regulatory documents did not include reliability targets for process systems; therefore, in the 1970s, Safety Design Matrices were used as a design tool to provide insights on the safety level of the plant.

TABLE I-5-1. REGULATORY DOSE LIMITS TO BE MET BY DESIGN FOR EXISTING NPPs

| Event frequency (event/r-year) | Regulatory Dose Limits to be met by Design for Existing NPPs | |
|-----------------------------------|--|---|
| | Darlington (from the Consultative Document C-006 Rev. 0) | Pickering, Bruce A and B and Point Lepreau (from the Siting Guide) |
| | Event Class / Individual Whole Body Dose Limit | Classification / Individual Whole Body Dose Limit |
| $\sim 10^{-1}$ to $\sim 10^{-2}$ | Class 1 / 0.5 mSv | Single Failure / 5 mSv (serious process equipment failure) |
| $\sim 10^{-2}$ to $\sim 10^{-3}$ | Class 2 / 5 mSv | |
| $\sim 10^{-3}$ to $\sim 10^{-4}$ | Class 3 / 30 mSv | |
| $\sim 10^{-4}$ to $\sim 10^{-5}$ | Class 4 / 100 mSv | Dual Failure / 250 mSv (equipment failure plus failure of any safety system) |
| $\sim 10^{-5}$ to $\sim 10^{-7}$ | Class 5 / 250 mSv | |

The Canadian Nuclear Safety Commission (CNSC) regulates Canadian NGS under the Nuclear Safety and Control Act since 2000. The new CNSC regulatory framework includes 14 Safety and Control Areas. Canadian utilities are required to have implemented programs to meet regulatory requirements in each of the 14 areas. By maintaining these programs, a utility always ensures both plant operation within its licensing basis and with low risk to the public.

Emergency preparedness

The Consolidated Nuclear Emergency Plan documents the concepts, roles, and resources required by any NGS to maintain an emergency response capability to protect the public, employees, and the environment in the event of a nuclear emergency. Its main objectives are to deal with releases of radioactive materials from fixed facilities, provide a framework for interaction with external authorities, and define the NGS' commitments under their respective Provincial Nuclear Emergency Response Plan. In the unlikely case of a nuclear accident, radiation surveys are performed on-site and off-site to estimate the source term. Survey results are then used to assist the shift organization in determining radiological hazards, and on-site protective actions, and to assist the Province in understanding requirements for off-site Provincial protective actions. The Action Levels typically specified in the emergency plans are as shown in Table I-5-2.

TABLE I-5-2. PROVINCIAL NUCLEAR EMERGENCY RESPONSE PLAN SURVEY RESULTS

| Protective Action | Whole-body dose | | Thyroid dose | |
|-------------------|-----------------|-------------|--------------|-------------|
| | Lower level | Upper level | Lower level | Upper level |
| Evacuation | 10 mSv | 100 mSv | 100 mSv | 1,000 mSv |
| Sheltering | 1 mSv | 10 mSv | 10 mSv | 100 mSv |
| Thyroid blocking | | | > 50 mSv | |

I-5.2. Activities following the Fukushima-Daiichi NPP accident

Immediately after the accident, reviews were conducted by all utilities, which included a re-examination of the safety case for each NGS to confirm the adequacy of their response to successfully manage events such as those that occurred at Fukushima-Daiichi NPP. This included the review of the effective implementation of the defence in depth concepts with respect to external hazards such as seismic, flooding, fire and extreme weather events, the measures for prevention and mitigation of severe accidents with existing Severe Accident Management Guidelines (SAMG) and emergency preparedness. Reviews also included the station blackout, and the equipment diversity needed to mitigate external hazards.

The reviews confirmed that each existing Canadian NGS had a robust safety case against hazards beyond those explicitly accounted for in its Design Basis. This is a direct result of their design (independent, separated and diverse safety systems), the high reliability of their Special Safety Systems, the multiple physical and administrative barriers, the multiple and large inventory of water surrounding the core, the many hours of passive cooling for the fuel, the unique in-ground spent fuel pools and a detailed emergency preparedness.

I-5.3. Safety improvement post Fukushima-Daiichi NPP accident

Multiple Barriers and the IVR (In Vessel Retention) Strategy For Beyond Design Basis Event Mitigation

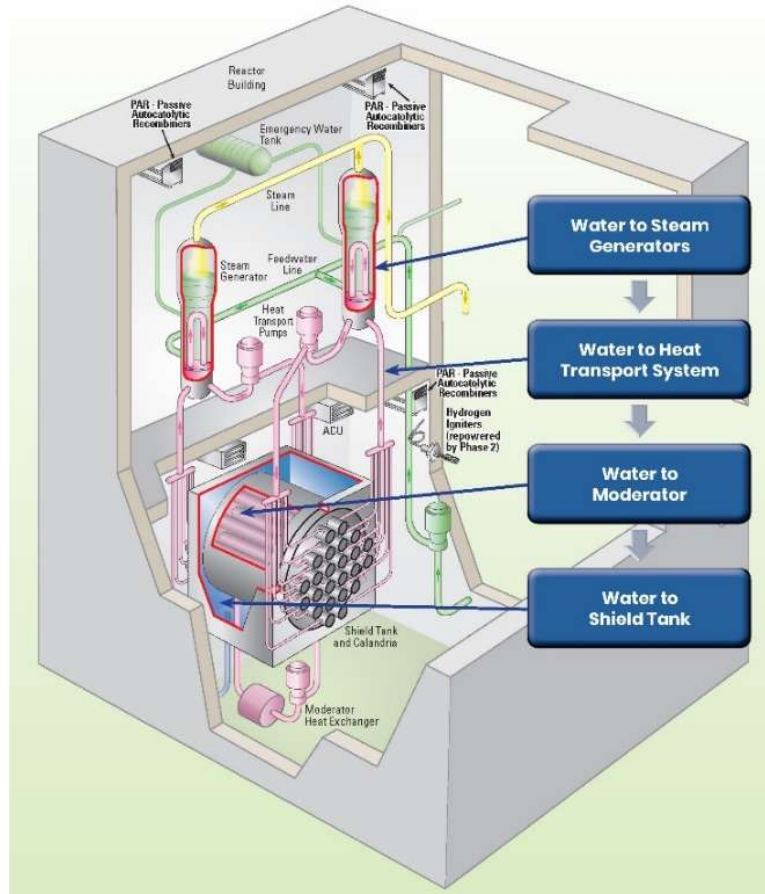


FIG. I-3. Multiple barriers and the IVR (In Vessel Retention) strategy for beyond design basis event mitigation. (Courtesy of OPG)

Following the Fukushima-Daiichi NPP accident, CNSC recommended improvement opportunities and raised associated actions as outlined in the 2012 *CNSC Action Plan: Lessons Learned from the Fukushima Nuclear Accident*. Canadian utilities responded to these actions and implemented all lessons learned. Responses reflected a common philosophy adopted by Canadian utilities to have actions and defences focused on stopping accident progression prior to a severe accident. This is accomplished by maintaining multiple and flexible barriers to severe event progression such as the In-Vessel Retention (IVR) strategy whose primary focus is to provide means of cooling to avoid challenges to Containment integrity. In the extremely unlikely case that IVR were not successful, SAMG strategies would ensure Containment is protected.

Emergency Mitigating Equipment (EME) was acquired with the objective of providing cooling water to the core through a variety of means:

- Depressurize the steam generators to inject cooling water;
- Depressurize the Heat Transport System to inject cooling water;
- Inject cooling water to the Moderator ;
- Inject cooling water to the Shield Tank.

Cooling water is injected using portable pumps while small portable generators provide power to critical monitoring equipment. EME implementation included adding quick-connects to engineered systems, refuelling capability, debris removal equipment and adding monitoring equipment to the spent fuel bays. A few utilities added larger mobile generators to repower hydrogen igniters and the Filter Air Discharge system, and a second backup air system for Containment airlocks.

Additional equipment installed by other initiatives include a portable and completely independent second emergency telecommunication system for each site, Passive Autocatalytic Recombiners, Containment Filtered Venting system and a Near Boundary Gamma Monitoring system.

I-5.4. Canadian experiences on implementation

EME implementation aimed at ensuring that the equipment is reliable, readily available, and easily deployable. EME is stored in large, flexible tents where it is fastened to the ground. To minimize the dependence on immediate external help, a strategy of on-site EME refuelling capability with a fuel truck and multiple totes was adopted. EME's reliability is supported by testing. Emergency Mitigating Equipment Guidelines (EMEG) were also developed and implemented and are an integral part of the training of NGS personnel to respond to an accident. The on-site Emergency Response Team oversees EME and have been trained and qualified to operate the equipment. The effectiveness of EMEG is constantly tested and evaluated with drills and exercises in each NGS. Scalability of the response to a beyond-design basis event has also been addressed by developing inter-utility collaboration and support agreements that will allow loaning both equipment and staff to an NGS in need.

Communication to the public was also improved: a new public alert system has been implemented that uses sirens in the neighbourhoods of the NGS and radio and TV to notify residents of accidents and it has the capability to send text messages to mobile phones. Also, the public at distances between 10 and 50 km from the NGS received or can readily receive KI pills upon request. Information to prepare for emergencies is also readily available.

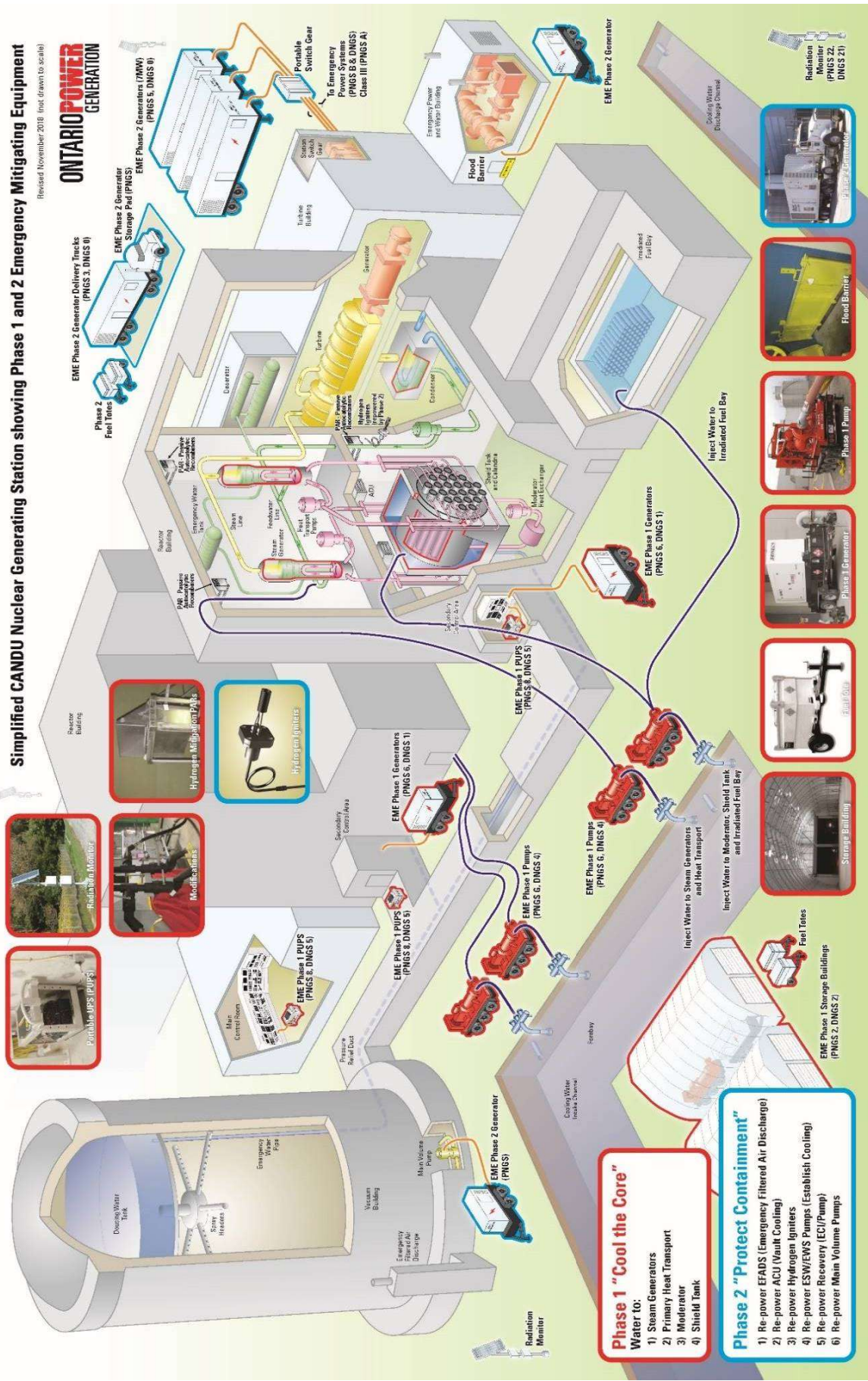
An Emergency Response Projection (ERP) computer code used to provide detailed venting projection of radioactive releases was also updated to include multi-unit severe accidents. This code is used to share needed information with Municipal, Provincial and Federal agencies to coordinate the appropriate response to an event. Large scale drills and exercises with multiple external agencies are also executed periodically. They represent great opportunities to rehearse the EMEG and SAMG as well as the updated ERP code.

In conclusion, all safety enhancements implemented in response to the 2012 CNSC Action Plan have strengthened Canadian reactors' defence in depth. This action plan and responses integrated in each organization's management system is now part of the Canadian NGS licensing basis.

Also, each Canadian utility has processes in place to identify opportunities for safety improvement as part of its commitment to continual improvement under its nuclear management system. CNSC also recently developed modern standards (i.e., REGDOC) that consider best regulatory practices from other countries and international standards, such as those of the IAEA, and incorporate lessons learned from Fukushima-Daiichi NPP in accordance with its 2012 action plan. While these modern CNSC regulations apply to new facilities, they are used during Periodic Safety Reviews to identify practicable safety enhancements to support continued safe operation of existing Canadian NGS.

Simplified CANDU Nuclear Generating Station showing Phase 1 and 2 Emergency Mitigating Equipment

Revised November 2018 (not drawn to scale)
ONTARIO POWER GENERATION



Phase 1 "Cool the Core"
 Water to:
 1) Steam Generators
 2) Primary Heat Transport
 3) Moderator
 4) Shield Tank

Phase 2 "Protect Containment"
 1) Re-power EFADS (Emergency Filtered Air Discharge)
 2) Re-power ACU (Vault Cooling)
 3) Re-power Hydrogen Igniters
 4) Re-power ES/WES Pumps (Establish Cooling)
 5) Re-power Recovery ECI(Pump)
 6) Re-power Main Volume Pumps

FIG. I-5-1. Simplified CANDU nuclear generating station showing phase 1 and 2 emergency mitigating equipment. (Courtesy of OPG)

I-6. CANADA – CANADIAN APPROACH TO SAFETY REASSESSMENT

I-6.1. Immediate

In response to the Fukushima Daiichi NPP accident, the Canadian Nuclear Safety Commission (CNSC) issued an order requesting all licensees of Class I nuclear facilities, under subsection 12(2) of the *General Nuclear Safety and Control Regulations*, to re-examine the safety cases of their nuclear power plants (NPPs) [11]. The World Association of Nuclear Operators (WANO) Significant Operating Experience Report (SOER) 2011-02 [12] also directed all Canadian NPP operators to undertake such reviews to confirm the safety of their plants, and to identify potential vulnerabilities to beyond design basis events. Canadian NPP licensees immediately initiated this re-examination of their safety case based on the initial lessons learned from the Fukushima Daiichi NPP accident.

The preliminary re-examination confirmed that Canadian reactors have a robust safety case to successfully manage events such as those that occurred at Fukushima Daiichi NPP, which confirmed that their safety cases include protection and mitigation against events beyond those assumed in their design basis. It also confirmed that the risk related to operation of their NPP continued to be acceptably low as documented in their licensing basis. No significant issues requiring immediate corrective or compensatory measures were identified.

During this re-examination of their safety cases against the initial lessons learned from events at the Fukushima Daiichi NPP, licensees also identified areas for further study, follow-up, and potential areas for improvements opportunities. The schedule for finalizing safety improvements already in progress before the events at Fukushima Daiichi NPP was expedited, such as installing passive autocatalytic recombiners and implementing severe accident management guidelines (SAMGs). Potential improvements for longer-term were investigated in an expeditious manner. While licensees and industry partners worked aggressively on follow-up actions related to the events at Fukushima Daiichi NPP, a Canadian Utilities working group was formed in March 2011 to work on common issues for the CANDU reactor fleet in Canada.

Also, in April 2011, the CNSC established the CNSC Fukushima Task Force (Task Force) to review licensees' responses and to evaluate the operational, technical, and regulatory implications of the Fukushima Daiichi NPP accident for Canadian NPPs to identify findings and recommendations. On September 11, 2011 the Task Force concluded that Canadian NPPs are safe and pose a very small risk to the health and safety of Canadians and the environment.

In line with licensees' efforts to identify areas for improvements to make their plants more robust to Fukushima Daiichi NPP-like events, the Task Force presented, in September 2011, thirteen recommendations to further enhance the safety of NPP in Canada.[13]¹⁵

¹⁵ [CNSC Fukushima Task Force Report](#), INFO-0824, issued October 2011.

Of importance are also the conclusions reached by the follow-up mission of the IAEA Integrated Regulatory Review Service (IRRS) that was conducted during this period (Fall 2011). The IRRS final report concludes that the CNSC response to the Fukushima Daiichi NPP accident was prompt, robust, comprehensive, and that Canada has an effective and pragmatic framework in place to implement the lessons learned from the accident, and to ensure the continued safety of Canadian NPPs.[14]

I-6.2. Longer-term

To address the Task Force recommendations the CNSC developed a draft Action Plan and embarked on a series of consultations with licensees, the public and other stakeholders, seeking their input in addressing the Task Force recommendations. Opportunities for improvements were adopted in 2012 and documented into the *CNSC Action Plan on the Lessons Learned from the Fukushima Nuclear Accident* (CNSC Action Plan). This established a four-year plan, for both licensees and the CNSC staff, to strengthen reactor defence in depth, enhance emergency response, improve regulatory framework and processes, and enhance international collaboration.[15]

Recognizing the high level of public concern following the Fukushima Daiichi NPP accident, Canadian NPPs developed a set of principles to guide their response, to ensure a consistent approach and to reassure the public. Each Canadian NPP agreed in 2013 to principles for beyond-design-basis events, with the objective of practically eliminating the potential for societal disruption due to a nuclear incident by maintaining multiple and flexible barriers to severe event progression.[16]

The set of nine principles guided efforts in ensuring each NPP has in place multiple and flexible barriers that act to prevent a severe event in the first instance, and further will mitigate the consequences if such a very unlikely event occurs. Reasonable and practical modifications were implemented such as necessary portable equipment readily available and deployable to provide emergency cooling and power to prevent progression to severe accident conditions. Permanently installed equipment is also in place such as PARs, hydrogen-igniters, and containment filtered venting to protect containment and minimize radioactive releases along with other engineered systems if an event progresses to severe accident conditions. SAMGs were updated to provide strategies to mitigate the consequences of a severe accident and return the plant as soon as possible to a long term safe stable state.

All Fukushima Action Items raised on Canadian licensees by the CNSC Action Plan were closed by end of 2015. Safety enhancements were integrated within each licensee's management system by making incremental changes within its governing document framework. Actions from the CNSC Action Plan on provincial and federal jurisdictions including the CNSC were also completed. The CNSC Action Plan and resulting safety enhancements in place at utilities are now integral to the Canadian NPP's licensing basis. As such, the integration of lessons learned within each licensee's management system has improved the accident management framework in place to maintain a long term stable state if an accident were to occur, including severe accidents.

I-7. CHINA

After the 2011 Fukushima Daiichi nuclear power plant accident happened in Japan, under the unified plan of the state council, China National Nuclear Safety Administration (NNSA) together with the National Energy Administration and the China Seismological Bureau and other relevant departments, have carried out a comprehensive inspection for all nuclear power plants (NPP) operating and under construction in mainland of China between March and December in 2011. The inspections were mainly based on China's current effective nuclear safety laws, regulations and technical standards, as well as the latest safety standards issued by IAEA and the preliminary experience and lessons exposed by Fukushima Daiichi nuclear power plant accident.

The comprehensive inspection included 11 key areas, which were as follows:

- The suitability of external events assessed during site selection;
- Assessment of flood control plan and flood control capability for nuclear facilities;
- Assessment of earthquake resistance plan and seismic capacity for nuclear facilities;
- The effectiveness of quality assurance system of nuclear facilities;
- Inspection of fire protection system of nuclear facilities;
- Prevention and mitigation measures of accidents induced by multiple extreme natural events;
- The analysis of Station Black-out accident, the availability of additional power supply after the loss of emergency power and emergency plans;
- Severe accident prevention and mitigation measures and their reliability evaluation;
- Plans coping with group incidents;
- The effectiveness of environmental monitoring and emergency system;
- Other possible weakness.

The inspections were mainly carried out by means of scheme evaluation, document review, power plant self-inspection, site investigation, records inspection and technical evaluation. For operating nuclear power plants, the inspections were focusing on the assessment of the capability to withstand extreme external events, severe accidents, emergence responses, and of the availability of nuclear power plant management procedures, operating procedures and emergency plans. For nuclear power plants under construction, the most recent safety standards were adopted.

The comprehensive nuclear safety inspection lasted more than 9 months. The overall conclusion is that, the nuclear power plants in operation and under construction in China mainland basically meet the requirements of nuclear safety regulations of China and the latest IAEA safety standards. The management during site selection, design, manufacture, construction, installation and commissioning is effective, and the quality assurance system is under normal operation. The engineering construction meets the design requirements, which has certain capability to prevent and mitigate severe accidents. The safety risk is under control and the safety is guaranteed.

However, some problems that might affect the construction quality and operation safety of China's civil nuclear facilities have also been found during the nuclear safety inspection. Referring to China's current nuclear safety regulations, the latest IAEA safety standards and lessons learned from the Fukushima Daiichi nuclear power plant accident, the main problems include: the problem of prevention and mitigation of severe accidents, the problem of design basis flood level of Qinshan Nuclear Power Plant, the problem of the tsunami impact on China's nuclear power plants and the earthquake resistance problem of high flux engineering test reactor.

NNSA has put forward the requirements of improvement actions for post Fukushima Daiichi nuclear power plants, to cope with the problems found in the comprehensive safety inspection, combining the experience feedback from Fukushima Daiichi NPP accident and the improvement work that can further improve the safety level of the NPP, and considering the importance of safety improvement and feasibility of implementation procedure.

Safety improvement requirements for NPP under construction and operating include: 1) Investigation and implementation of waterproof plugging, 2) Installation of facilities such as portable power sources and pumps, 3) Ensuring the effectiveness of the seismic monitoring and recording system of NPP, improving the seismic response capability of NPP, 4) Flood control improvements of NPP, 5) Further evaluation on earthquake and tsunami risks, 6) Perfecting the NPP Severe Accident Management Guidelines (SAMG), improving hydrogen removal facilities if necessary, 7) Increasing the capability of emergency response to nuclear accidents, 8) Enhancing public publicity and information opening, 9) Developing level 2 Probability Safety Analysis (PSA) and external event PSA, 10) Perfecting the analysis and assessment of emergency control centre functions and habitability, 11) Perfecting fire preventing plan and management procedures, in order to improve the ability of early warning and response.

In terms of regulations, NNSA has officially approved and published Code on Safety of Nuclear Power Plants: Design (HAF102-2016) [17] in October 2016. HAF102-2016 is based on IAEA Safety Standards Series no. SSR 2/1 (Rev. 1) [6], Safety of Nuclear Power Plants: Design, and also appropriately takes into account some concepts and contents from HAF102-2004 [18]. The experience of nuclear power design, construction and operation in China are also included, along with experience feedback from Fukushima Daiichi nuclear power plant accident.

I-8. CZECH REPUBLIC

Czech Republic operates two nuclear installations, particularly:

- Dukovany NPP with four reactor units of VVER 440/213.
 - Unit 1 - in operation since 1985
 - Unit 2 - in operation since 1986
 - Unit 3 - in operation since 1987
 - Unit 4 - in operation since 1987

- Temelín NPP with two reactor units VVER 1000/320.
 - Unit 1 - in operation since 2000
 - Unit 2 - in operation since 2002

I-8.1. Regulatory framework

Czech nuclear legislation is fully harmonized with WENRA / IAEA / NSD requirements.

Degree 329/2017 – Requirements for the nuclear plant design

Plant design has to meet the following safety goals:

- To ensure that practically eliminated are:
 - Radiation accidents with insufficient time to implement urgent protective measures for public (early radioactive releases);
 - Radiation accidents requiring urgent protective measures for public, and which could not be limited in area or time (large radioactive releases).

Degree 162/2017 – Requirements for safety assessments

Large early release failure => Release of more than 1% of initial core inventory of Cs-137 earlier than 10 hours after radiation accident was declared.

I-8.2. Identification of safety improvements

I-8.2.1. Post-Fukushima Stress Tests

The objective of the safety assessment was to evaluate the level of robustness and sufficiency of safety margins during exposure to extreme natural conditions (considering the facts of the accident at Fukushima Daiichi NPP), loss of power, loss of ultimate heat sink, and if the event has escalated into a severe accident. A detailed deterministic evaluation was performed to identify the level of defence in depth and the capability to fulfil the fundamental safety functions during the specific initiating events and design extension conditions regardless of extremely low probability of their occurrence. The evaluation was performed for all reactor (and spent fuel pool) operating modes and states, including the case if all site units were affected.

The assessment confirmed for the majority of emergency scenarios that sufficient margins exist and barriers are robust enough to provide defence in depth both in the area of design and in the area of personnel, administrative and technical provisions for accident management.

In spite of considerable robustness of barriers, it was concluded based on results of assessment that opportunities for further safety improvements exist with respect to highly improbable beyond design basis situations. The measures for safety improvements were included in the Post-Fukushima National Action Plan [19].

I-8.3. Post Fukushima Measures

Based on the identified safety margins and general concept of DiD, three-level structure of measures to fulfil specified goals has been adopted. All these three levels are preventive – prevention of an event development into severe accident. If all three levels fail, then the event will progress into severe accident and the last level of DiD – mitigation of consequences of severe accident is applied. This multi-level layout of measures is presented in Table I-8.

TABLE I-8. MEASURES TO FULFILL PLANT DESIGN SAFETY GOALS

| Response Plan | Assumptions | Measures |
|--|--|---|
| Basic (origin design, permanently installed equipment) | All plant design SSCs for DBA available | <ul style="list-style-type: none"> • Permanently installed equipment for DBA (ESF, ER facilities, ...) • Standard personnel staffing • Standard documentation |
| Back-up (new diverse, permanently installed equipment) | Failure of one or more plant design SSCs | <ul style="list-style-type: none"> • Additional permanently installed, diverse equipment • Additional personnel staffing for operation of diverse equipment • Documentation for use of diverse equipment • Provisions for functioning for at least 72 hrs |
| Alternate (new mobile equipment) | Failure of all design and diverse SSCs | <ul style="list-style-type: none"> • Mobile equipment available on site protected against external hazards capable to fulfil the specified functions • Dedicated personnel staffing (on site) for operation of mobile equipment • Documentation for use of mobile equipment <p><i>After 72 hrs: additional resources from offsite until power, water, and coolant injection functions are not restored</i></p> |
| Ultimate (SA consequences mitigation) | Total loss of all capabilities to fulfil explicit function | <ul style="list-style-type: none"> • All equipment used (even beyond design bases) • Equipment dedicated for SA mitigation (e.g. PARs) • All available personnel staffing • Documentation developed for SA mitigation |

Response actions will be prioritized based on available equipment, resources, and time constraints. All response plans can be performed with available site personnel in post-event phase.

The graded approach is used, i.e. for each level of measures the set of different functional requirements is defined, corresponding to significance of risk to meet the desired objective for which the means and measures are proposed. Functional requirements for basic SSCs (existing design systems) are specified by the project. These requirements are the most stringent - based on legislative requirements (e.g. resistance to a single failure, qualified for extreme conditions). For backup and alternate SSCs the less stringent functional requirements are applied (e.g., backup devices might not be a resistant to a single failure, the mobile devices do not necessary meet the qualification requirement on the LOCA, respectively HELB environment).

The philosophy of post-Fukushima safety improvements is shown in Fig. I-8-1.

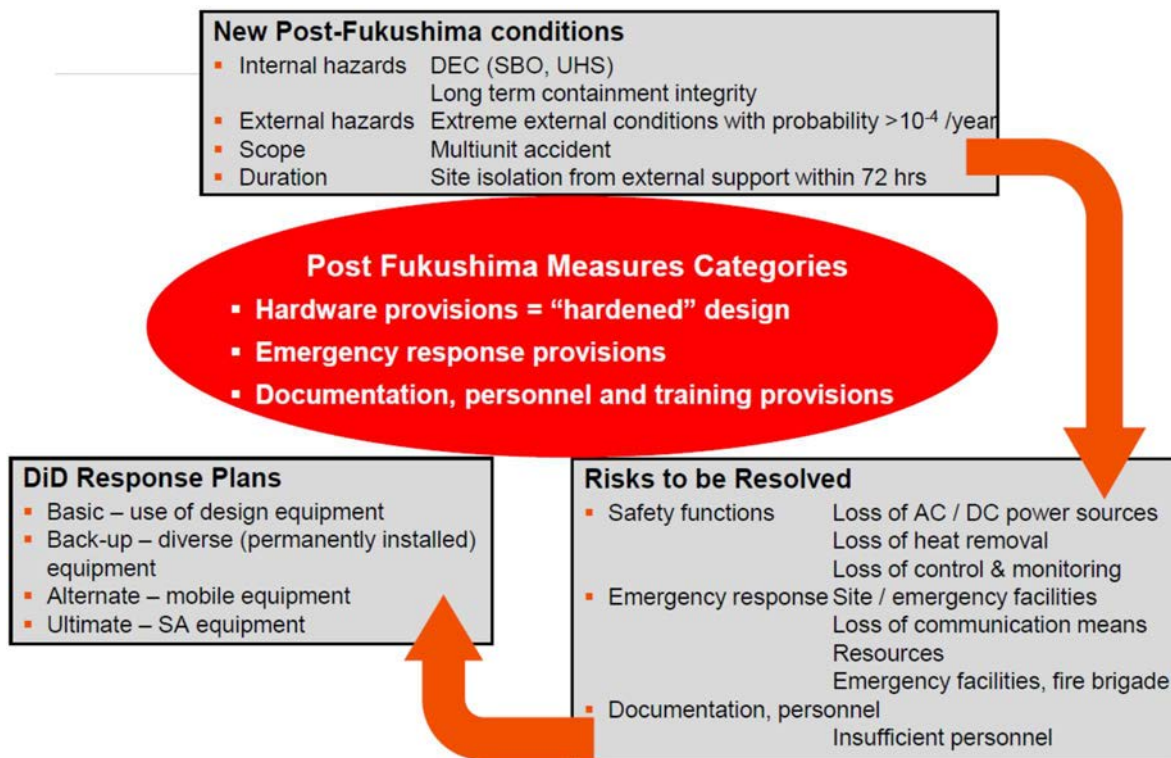


FIG. I-8-1. Philosophy of post-Fukushima measures.

Important safety improvements

Safety improvements comprising modification of certain basic plant design features, complemented by additional back-up, mobile and ultimate means were implemented on units in the Czech Republic as shown in Fig. I-8-2.

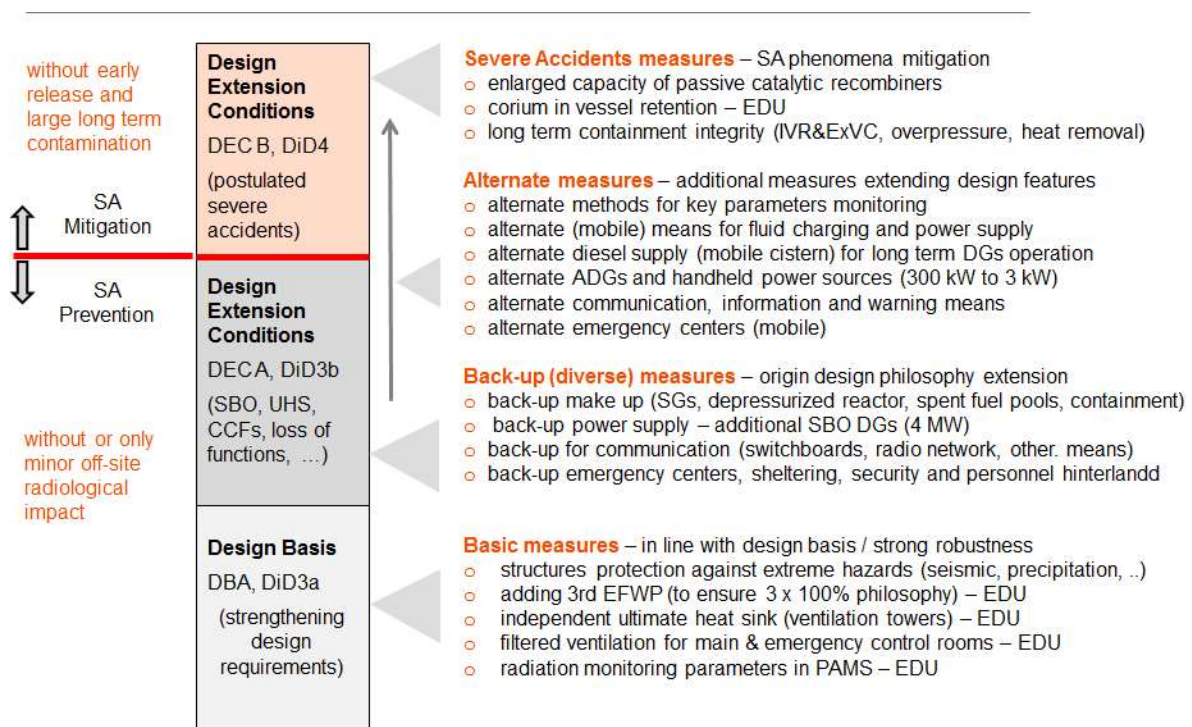


FIG. I-8-2. Safety improvements of certain basic plant design features in Czech Republic.

New emergency response provisions are now available also with the philosophy shown in Fig. I-8-4 and Fig. I-8-5.

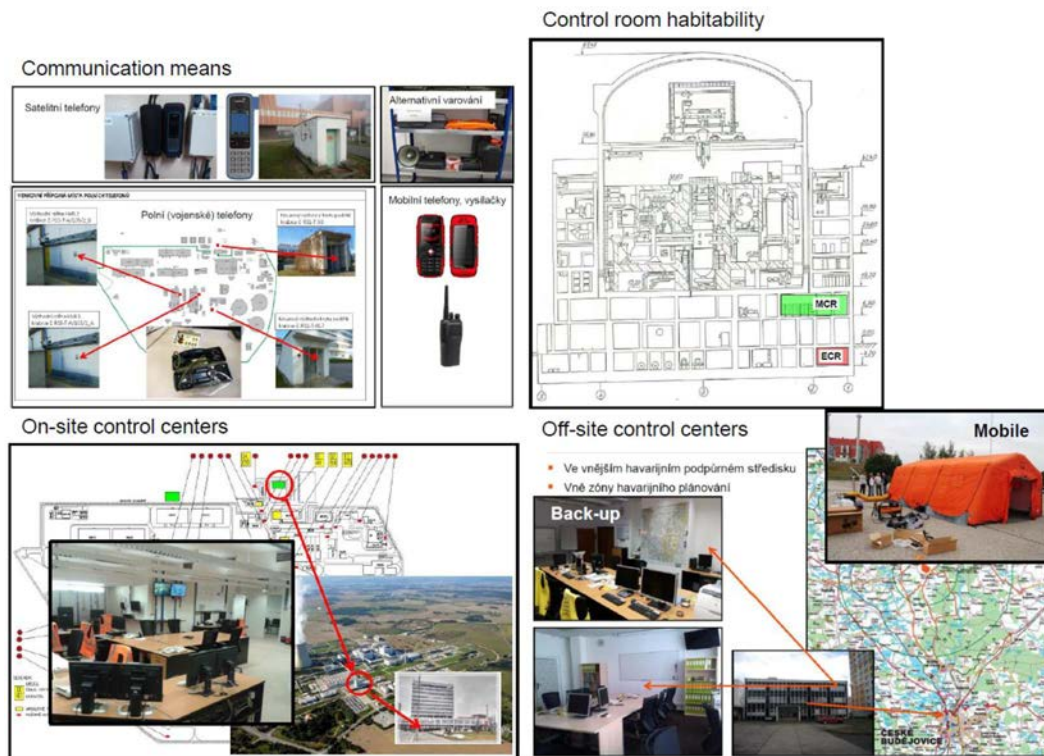


FIG. I-8-4. New emergency response provisions

In case of loss of design equipment functions caused either by loss of site control capabilities or by loss of safety functions, the diverse and mobile equipment supported by dedicated emergency response provisions is used. The corresponding actions are described in new procedures and guidelines that were implemented to extend the existing procedures and guidelines for design extension conditions. The philosophy of new procedures and guidelines dedicated for loss of design equipment functions is shown in Fig. I-8-5.

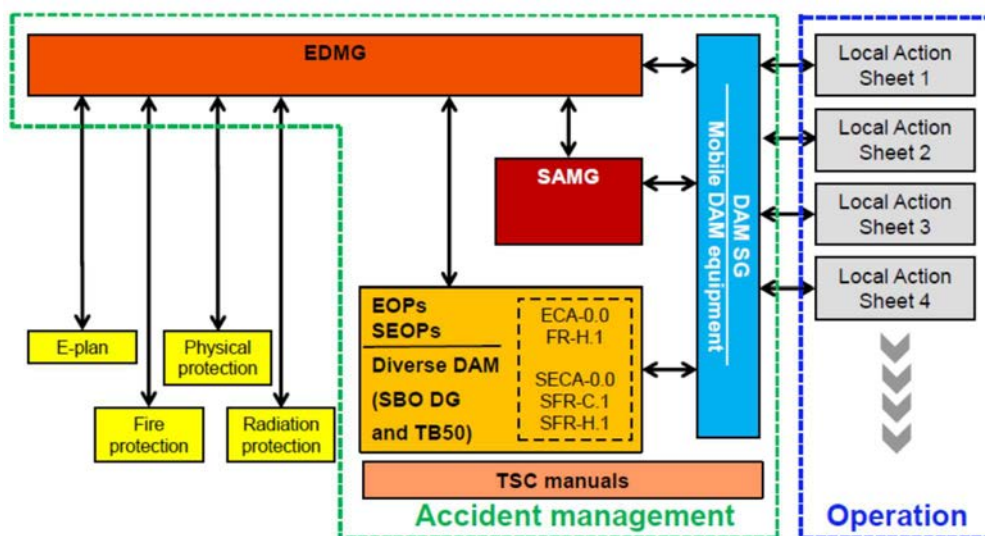


FIG. I-8-5. Philosophy of new procedures and guidelines.

I-9. FINLAND – IMPLEMENTING REASONABLY PRACTICABLE SAFETY IMPROVEMENTS AT THE FINNISH NPPS

I-9.1. Introduction

The general rules for comprehensive and systematic periodic safety assessments at existing nuclear facilities are presented in the Finnish Nuclear Energy Act [20]. 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.”

There are two nuclear power plants operating in Finland: the Loviisa and Olkiluoto plants. The Loviisa plant comprises of two PWR units (pressurized water reactors, of VVER type), and the Olkiluoto plant two boiling water reactor (BWR) units. These reactor units started commercial operations between 1977 and 1980. Several plant modifications have been carried out at the both plants during their lifetime. The most important projects since the commissioning have been modifications made for protection against fires, modifications based on the development of the probabilistic risk assessment (PRA) models, development of severe accident management strategies and implementation of required measures, modifications based on the lessons learnt from the Fukushima Dai-ichi NPP accident, reactor power upratings, and construction of training simulators, interim storages for spent fuel and repositories for reactor operational waste.

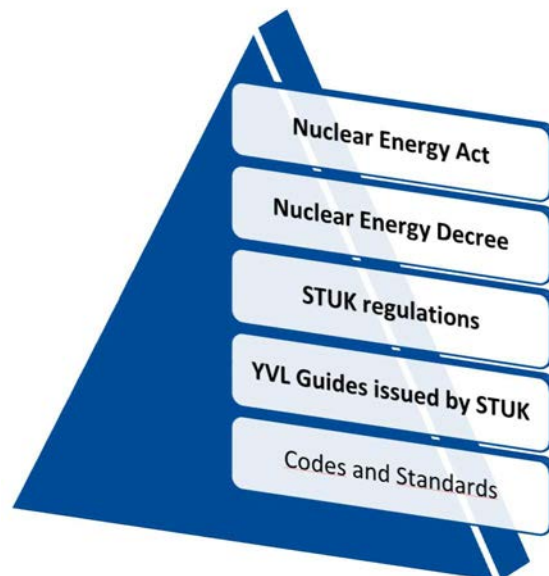


FIG. I-9-1. The structure of Finnish nuclear safety regulations.

I-9.2. Regulatory framework requiring periodic safety assessments and improvement of safety at existing NPPs

The structure of Finnish nuclear safety regulations is shown in Fig. I9-1. 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.

Regular update of regulatory guides setting objectives for new nuclear facilities and the implementation process particularly with regard to nuclear power plants in operation, are unique measures on the international perspective. 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.

For example, requirements related to severe accident management (SAM) and the dedicated SAM systems were introduced 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 NPP accident. Utilities started planning the implementation of the measures in 1980's and first SAM systems were installed at the plants in 1989 (see section on severe accident management strategies and implementation).

Regulatory guides have been regularly further updated. 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 [7] and WENRA safety objectives for new reactors [21]. In addition, the lessons learnt from the Fukushima Daiichi NPP accident were taken into account.

I-9.3. Periodic safety reviews

The implementation of safety improvements has been a continuing process at both Finnish NPPs since their commissioning. Finland has successfully applied periodic safety reviews (PSR) for the operating NPPs. 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 (see Fig. I-9-2). In Finland, this process is covered by the process of implementation decisions of revised regulatory guides written always for new nuclear facilities (see previous section). 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.

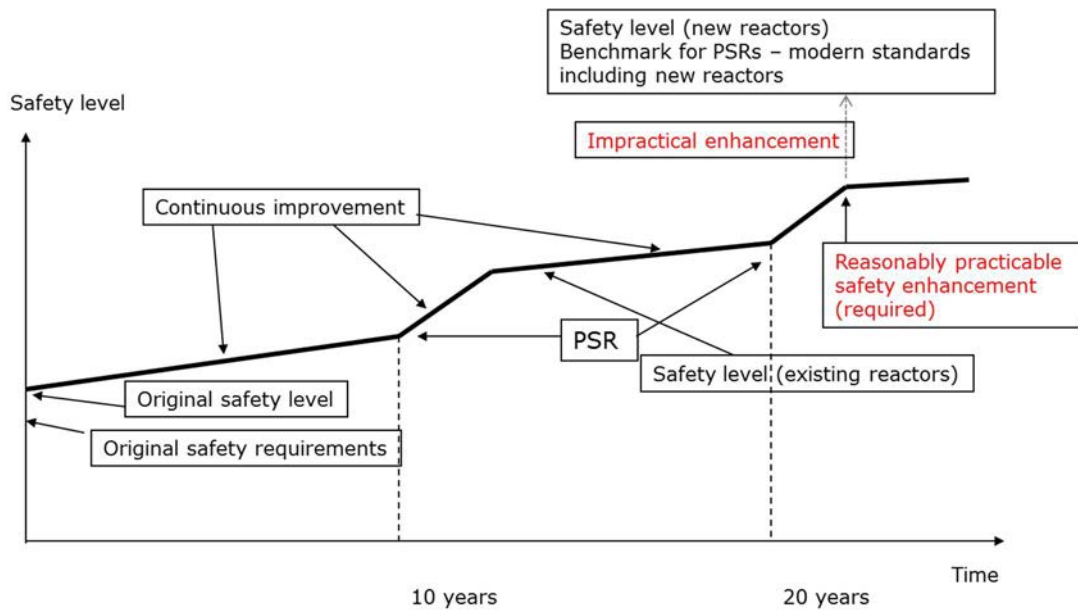


FIG. I-9-2. The concept of continuous improvement [22]¹⁶.

The last PSR of the Loviisa NPP was carried out in 2015-2017, and the Olkiluoto NPP PSR was carried out in 2016-2018. Key issues in the last Loviisa NPP PSR and Olkiluoto NPP PSR have been ageing management, organizational 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 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 PSR has been recently finalized. The specific topics discussed were organizational issues, performing the primary

¹⁶ When the existing NPPs were commissioned, their original safety level met the required safety level based on the safety requirements that were in force then. Safety requirements for NPPs can be updated and new NPPs are designed to meet higher levels of safety than the existing ones. Despite the fact that existing NPPs undergo PSRs as a result of which safety enhancements are implemented, it is likely that there will remain a difference between the safety level of oldest and newest reactors.

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/1/2016 (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, in accordance with 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.

I-9.4. Use of PRA for identifying further safety improvements

Finnish regulations require that licensees maintain an up-to-date and comprehensive plant-specific probabilistic risk assessment (PRA) and that the licensee uses the PRA to enhance nuclear power plant safety, to identify and prioritise plant modification needs and to compare the safety significance of alternative solutions.

In 1984 STUK required that the Finnish utilities shall make extensive probabilistic risk assessments for the Loviisa and Olkiluoto NPPs. The objective of these assessments was to determine the plant-specific risk topographies of the essential accident sequences. Another important objective was to enhance the plant personnel's understanding of the plant and its behaviour in different situations. Therefore, STUK also required that the PRAs are performed mainly by the utility personnel and external consultants are used only for special topics. In 1987 STUK published the Regulatory Guide on PRA. The Guide has been regularly updated and currently the Guide requires a full-scope (including internal events, fires, floods, seismic events, harsh weather and other external events) PRA for power operation and low-power and shut-down states. PRA shall cover the analysis of the probability of core damage (Level 1) and release of radioactive substances (Level 2). PRA shall be updated continuously to reflect plant and procedure modifications and changes in reliability data. The probabilistic safety goals for core damage frequency and large radioactive release apply as such to new NPP units. For operating units, instead of the numerical safety goals, the SAHARA (safety as high as reasonably achievable) principle and the principle of continuous improvement are applied.

Fortum Power and Heat Oy (Fortum), the licensee of Loviisa NPP, provided STUK with Level 1 PRA in 1989. Since then, Fortum has extended PRA by analysing risks related to fires, floods, earthquakes, severe weather conditions and outages, as well as by conducting Level 2 PRA. Until year 2014, PRA was done only for Loviisa NPP unit 1 and the small differences between the NPP units were assessed on case by case basis. Thereafter unit-specific PRA models have been kept up-to-date reflecting the small differences between units 1 and 2. Plant modifications have been carried out continuously at the Loviisa NPP, including safety system improvements, fire safety improvements, implementation of Severe Accident Management systems and a major modernisation programme including power uprate in the mid 1990's. By means of these modifications, risks have been decreased and the risk topography of the plant has been

balanced, as can be seen in FIG. I-9-2. Technical solutions of the modifications have also been often justified with PRA.

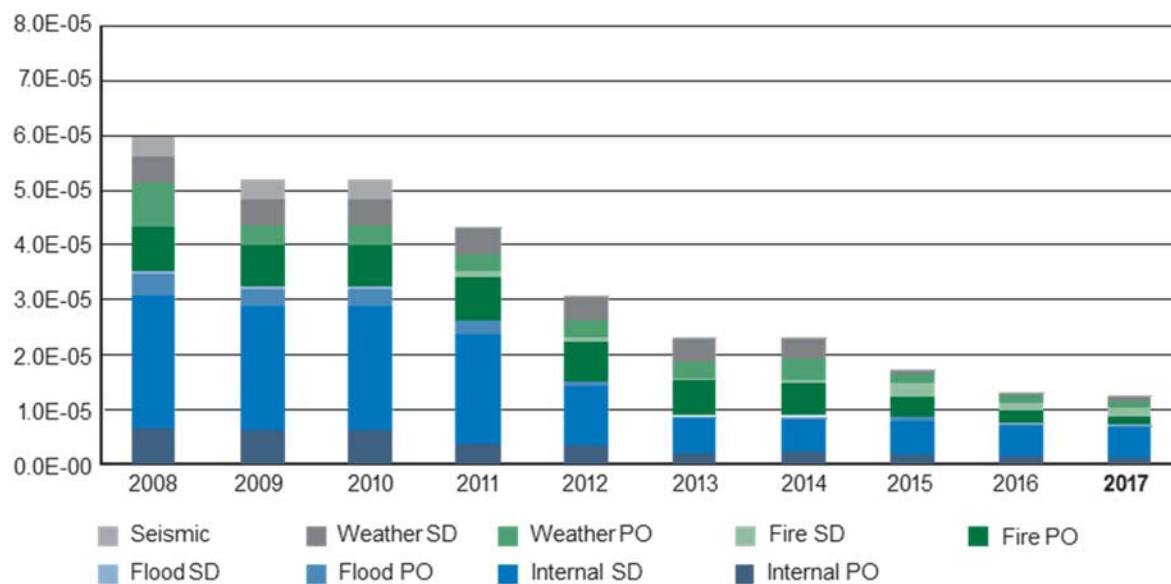


FIG. I-9-3. Development of the estimate of annual core damage frequency of the Loviisa NPP in 2008-2017¹⁷.

Teollisuuden Voima Oyj (TVO), the licensee of Olkiluoto NPP, submitted to STUK the first version of Level 1 PRA of units 1 and 2 in 1989. Since then, the PRA has been updated several times and the scope has been extended. TVO has now practically full-scope PRA covering levels 1 and 2 for full power operation and for low power and shutdown states. After 2013, unit-specific PRA models have been kept up-to-date reflecting the differences between Olkiluoto units 1 and 2.

Annual core damage frequency since 2008 is shown in Fig. I-9-3. In 2014, a new recirculation line modification in auxiliary feedwater system was implemented. The modification reduced the system's dependence on seawater cooling. A similar modification has not yet been implemented at unit 2. Core damage frequency can also increase when updating the PRA model. The risk estimate increase in 2009 is due to a more detailed analysis of the capacity of decay heat removal by diverse systems. The risk estimate increase in 2011 is due to the change of the method used to determine fire ignition frequencies and update of external hazards study that contains a new man-made hazard "marine oil-spill". Risk increase in 2015 estimate is due to more realistic modelling of operator and operating staff actions during shutdown.

¹⁷ The following modifications have decreased core damage frequency: the independent air-cooled cooling units for decay heat removal from the reactor core and from the spent fuel pools, enhanced protection against extreme high seawater level, renewal of auxiliary service water system, renewal of pressuriser overpressure protection valve, renewal of pressuriser spray system and new procedures for sump recirculation in shutdown states.

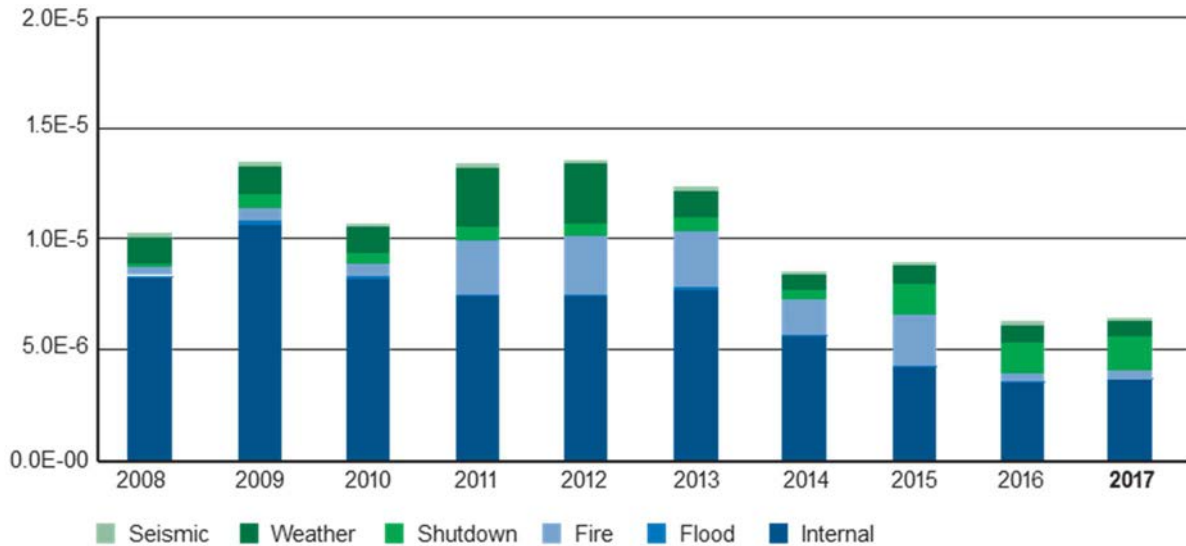


FIG. I-9-4. Development of the estimate of annual core damage frequency of the Olkiluoto NPP in 2008-2017.

I-9.5. Severe accident management strategies and implementation of required measures at existing Finnish NPPs

The current Finnish requirements related to ensuring the containment building integrity state (STUK Regulation Y/1/2016):

- 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;
- 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 presented in the Guide YVL B.6 that, for example, requires that the SAM systems shall be independent from other systems, safety classified, fulfil the single-failure criterion, and qualified for severe accident environmental conditions.

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 NPP and Chernobyl NPP. These strategies are based on ensuring the containment integrity that 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:

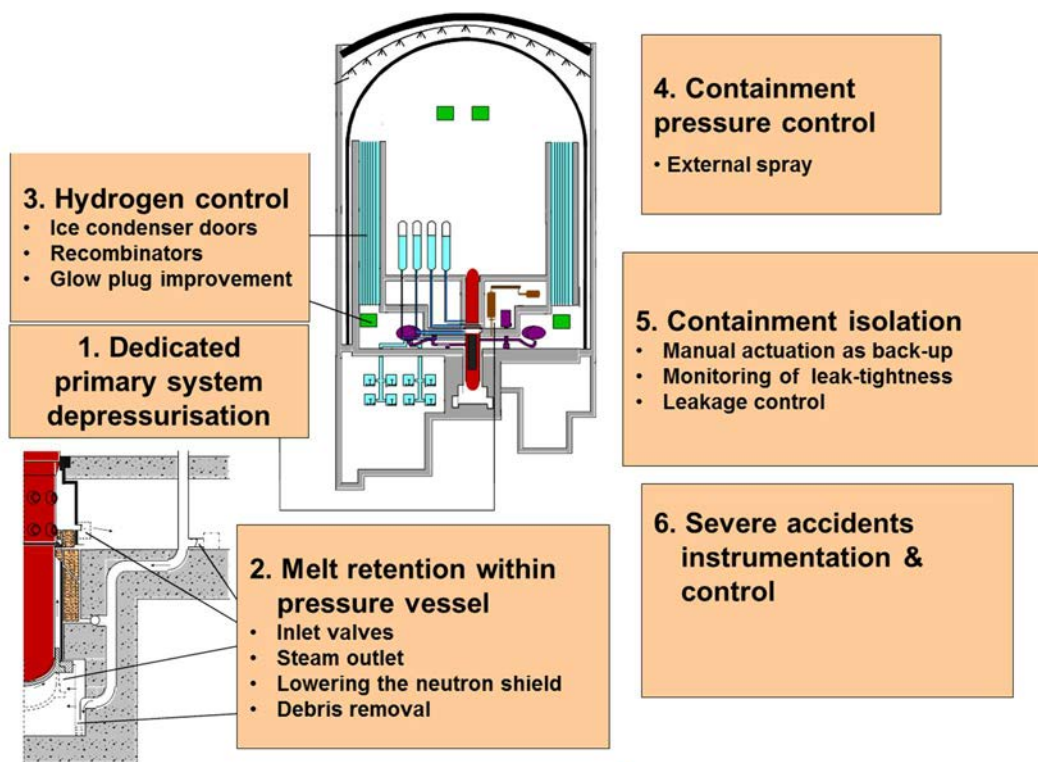
- Depressurisation of the primary circuit;
- Absence of energetic events, i.e. hydrogen burns and steam explosions;
- Coolability and retention of molten core in the reactor vessel;
- Long term containment cooling ensuring subcriticality;
- Ensuring containment isolation.

The developed SAM strategy lead to a number of hardware changes at the plant (see Fig. I-9-5) 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 overpressurization 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 sub-atmospheric 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 that 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.



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FIG. I-9-5. Plant modifications at the Loviisa NPP for severe accident management [Fortum].

The main provisions for severe accident management were installed at the Olkiluoto units 1 and 2 during the SAM project that was completed in 1989. The measures implemented were (see Fig. I-9-6):

- 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;
- 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 might 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 might expose the structures and penetrations that 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.

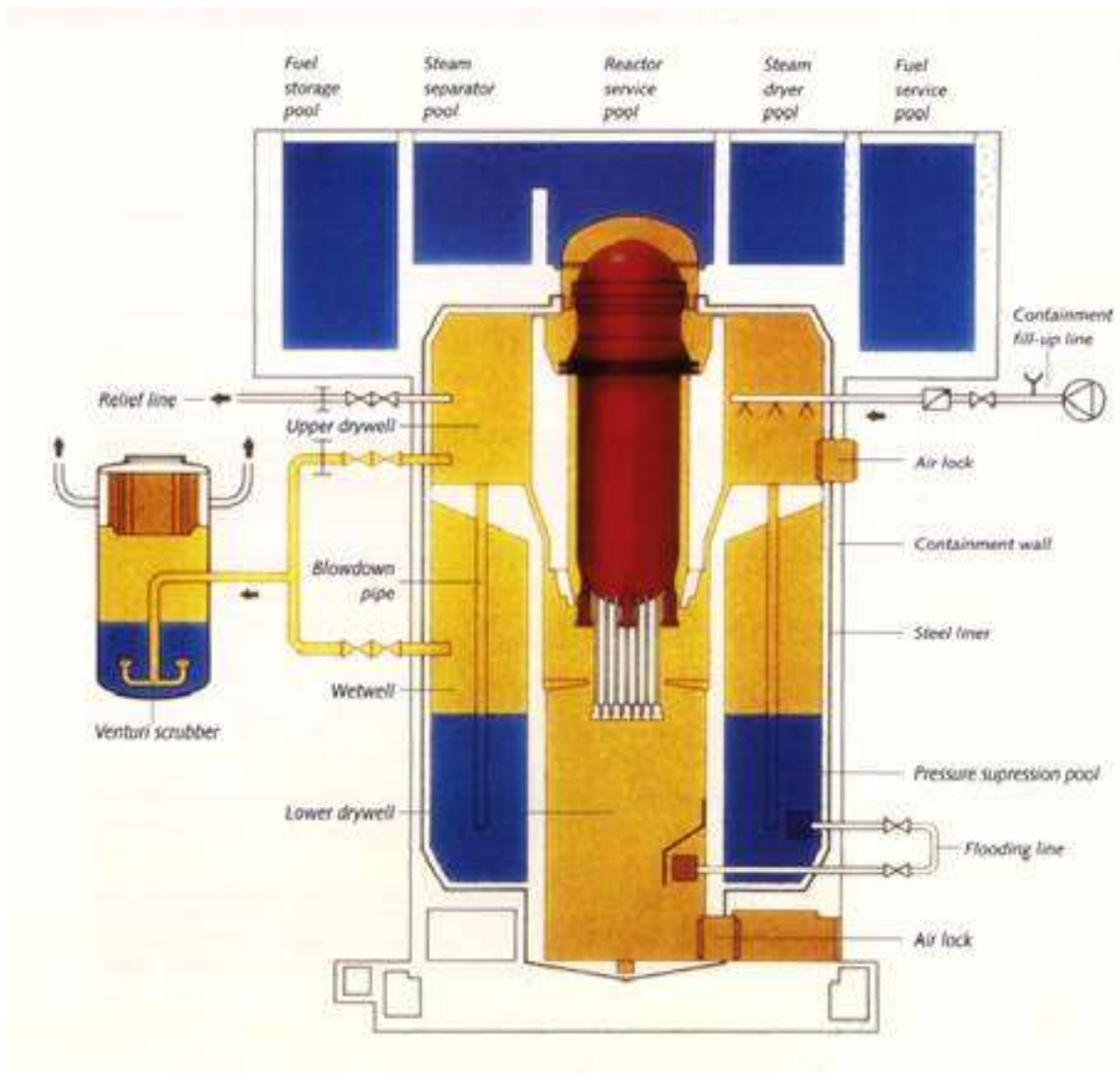


FIG. I-9-6. Plant modifications at the Olkiluoto NPP units 1 and 2 for severe accident management [TVO].

In addition to deterministic requirements to ensure the containment building integrity in case of a severe accident, there are also requirements related to limitation of radiation exposure and releases in accidents. The Finnish Government Decision in 1991 included a requirement that

“there shall be no acute harmful health effects nor long term restrictions on the use of extensive areas of land and water”. The updated Government Decree stipulates that “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.” This wording is very close to the Vienna Declaration on Nuclear Safety published in 2015. Finnish regulatory guide YVL C.3 explains in more detail what is meant by “large scale protective measures.” Analyses have to 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 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.

These probabilistic safety goals 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 NPP 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 NPP 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 NPP 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 in accordance with the SAHARA principle.

I-9.6. Further safety improvements at the Finnish NPPs based on the lessons learnt from the Fukushima Daiichi NPP accident

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 Daiichi NPP accident. Safety assessment was made to study on how the Finnish NPPs have prepared against loss of electric power supply and extreme natural phenomena. The licensees' action plans include 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 Daiichi NPP accident 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 Daiichi NPP 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 Section 5).

The experiences from the Fukushima Daiichi NPP accident were also addressed in the renewed Finnish nuclear safety regulations.



FIG. I-9-7. Independent air-cooled cooling units for decay heat removal from the reactor core and from the spent fuel pools at the Loviisa NPP [Fortum]

I-9.7. Plant modernizations also considered as opportunities to improve safety

Original design lifetime of Loviisa units 1 and 2 was 30 years and currently valid operating licences extended the lifetime to 50 years. The licensee of Olkiluoto units 1 and 2 has applied an operating licence renewal for additional 20 years, which would extend the original design

lifetime of these units from 40 years to 60 years. The expected lifetime of the existing NPPs requires renewal of systems, structures and components and modernization of technologies.

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 may be planned resulting in ageing issues in the remaining parts. This is why improving safety is a continuous process from the start of plant operations and not related only for example for plant's long term operation.

I-9.8. Aspects to be considered when assessing what are reasonably practicable safety improvements and their timely implementation

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 section describing the regulatory framework). Also periodic safety assessments and use of PRA can bring new sights for safety improvement needs when looking the overall picture of the plant safety (see sections on PSR and PRA).

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 Daiichi NPP 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 need to be handled first instead of limiting the assessment only on for example in seismic hazards and flooding.

Previous sections presented the safety improvements carried out at the operating Finnish NPPs that have been considered reasonably practicable. 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. 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, airplane crashes have to be considered. The topic has been handled mainly in security related cooperation activities.

For example, when an enlargement of the spent fuel interim storage at the Olkiluoto site was carried out in 2009-2015, the protection against airplane crashes was required. In the project three additional pools were built, and the storage structures were also modified to comply with the current safety and security requirements. The extension increases the capacity to comply with the spent fuel coming from the Olkiluoto plant units 1, 2 and 3. The main reasoning for requiring the protection against large airplane crash was the spent fuel coming in future also from the plant unit 3 under construction, where the airplane crash protection has been a design basis. New cover slabs were installed for the pools protecting from possibly falling debris and large landfill embankment was built outside the storage protecting from possible direct impact. Because the backfitting measures were done for already existing facility, dimensioning was an optimisation task and acceptability of different design options was not always obvious (e.g. protection against large aircraft vs. small plane). Timely implementation of safety improvements is also an important aspect. The justified safety improvements need to 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). There are examples in Finland of plant modifications where the first design solutions have turned out not to be suitable or have added some additional risks. One example is the diversification of reactor water level measurements at the operating Olkiluoto units 1 and 2. The licensee has studied possibilities to supplement the currently used low level measurement system with another system based on a different measuring principle. The implementation has been delayed several years because of difficulties in finding a suitable technology that can be proved functional in the test facility. Other plant modifications that have been delayed at the Finnish NPPs are related to reducing the risk arising from heavy load lifting at the Loviisa NPP. The topic was recognised in the PSR finished in 2007 and is taken into account in the modernisation of the Polar crane (structural reliability) and the refuelling machine (lifting routes can be changed in connection with the machine renewal). The Polar crane modernisation will be finalised in 2018 and refuelling machine modernisation is now restarting. These projects have not been prioritised at the plant and they have had a lot of project management challenges.

According to a recent WENRA publication [23] concerning reasonably practicable safety improvements, there is no standard set, or tick list, of specific engineering or operational improvements that will be appropriate for all reactors and operational regimes, though it is

expected that licensees will look at what others have done for example to prevent and mitigate radioactive releases to see if it is appropriate for them. If those measures are not appropriate the licensees need to look at what else they could do to achieve a broadly similar outcome. The paper also discusses what is considered being proportionate where one aspect is that the greater the shortfall, the more needs to be done to identify and implement measures to remove or reduce it.

I-9.9. Conclusions

The Finnish nuclear safety regulations include rules for comprehensive and systematic periodic safety assessments at existing nuclear facilities in order to identify reasonably practicable and achievable safety improvements that shall be implemented in a timely manner. The practical tools in Finland are the periodic safety review, use of probabilistic risk assessment and regular updates of regulatory requirements based on which the licensees are required to identify and prioritise the needed plant modifications to improve plant safety.

Several plant changes have been carried out at the Finnish nuclear power plants during their lifetime. Some of the most extensive modifications improving plant safety include the development of severe accident management strategies and implementation of the required measures. The Finnish regulatory guides include requirements for severe accident management systems, which shall be independent from other systems, safety classified, fulfil the single-failure criterion, and qualified for severe accident environmental conditions. These regulatory requirements have been applied also for operating NPPs.

I-10. FRANCE

I-10.1. Introduction

58 nuclear units based on pressurised water reactors (PWR) are operating in France by one operator (*Electricité de France* - EDF). These units have electrical power capacities varying from 900MWe to 1,500MWe and are spread out over 19 sites, with an average age of 33 years.

In 2017, the French NPPs have produced 379 TWh of electricity (~89% of the total electricity production in France).

EDF's PWR model is divided into three series of available electrical power:

- A 900 MWe series consisting of 34 units with an average age of 36 years.
- A 1300 MWe series consisting of 20 units with an average age of 29 years.
- A 1500 MWe series consisting of 4 units with an average age of 17 years.

The regulator is the ASN (*Autorité de Sûreté Nucléaire*).

I-10.2. Safety improvements at EDF's NPP – Overview

EDF is implementing a continuous improvement approach to the safety of French nuclear reactors in operation since they were commissioned. This approach is based on taking into account:

- Lessons learned from French and foreign experience feedback (e.g.: internal controls, inspections, Operational Experience, ...);
- The results of research and development as well as advances made possible by the improvement of knowledge and technologies;
- The adaptations and evolutions necessary to meet regulatory developments and increased safety and environmental protection objectives.

Safety reviews are an integral part of this approach, which has resulted in the deployment over time of modifications to power plants and installation operating instructions to increase the level of safety of the French nuclear fleet. This approach also takes into account the continuous improvement of organizations and the feedback from the men and women who work on nuclear power plants on a daily basis.

Nuclear safety is subject to numerous feedback coming from internal control and external inspections.

EDF has implemented internal control procedures. For example, every three to four years, EDF performs overall safety assessments for each nuclear power plant, which take place over a three-week period and involve approximately 30 inspectors. In addition, the General Inspector for nuclear safety and radiation protection, reporting to and appointed by EDF's Chairman and CEO, performs annual audits, issues an opinion on the overall safety of the nuclear fleet and suggests improvement actions to the Company's management. Efforts by EDF, notably to improve human performance, have resulted in a halving over ten years of the annual average number of automatic reactor trips. In 2017, they totalled 22 throughout the fleet.

The external control of the safety of nuclear facilities in France is carried out by the ASN, at the national level. There are two types of audits:

- Scheduled or unannounced inspections carried out by the ASN (473 inspections in 2017 over all EDF nuclear facilities);
- A periodic (ten-year) review process (Periodic Safety Review) designed to improve the compliance of nuclear plants with applicable rules and update assessments of the risks facilities pose to the environment and public health, taking into account the state of the facilities, the experience gained during their operation, new developments in nuclear science, and rules applying to similar facilities. The targets are established by the ASN, which monitors compliance; EDF proposes solutions to meet these targets and implements them after obtaining the approval of the ASN. The periodic safety review is an important step in continuing the operation of power plants.

At the international level, regular inspections are held making it possible to share the experience gained worldwide:

- The OSART (Operational Safety Review Team) of the IAEA performs reviews at the request of the French government with the objective of formulating recommendations and promoting best practices. In particular, EDF's first Corporate OSART was held in 2014 and concluded that EDF is fully compliant with the standards defined by the IAEA; the Follow Up Corporate OSART took place at the end of 2016;
- The international "peer review" inspections carried out by the WANO (World Association of Nuclear Operators) are organized at the request of EDF to assess safety performance compared to best international working practices. For example, a Corporate Peer Review took place in 2017 aimed at assessing the mode of governance and relations between corporate HQ and the facilities. At the end of the Corporate Peer Review WANO identified two best practices to do with applying the Nuclear Rapid Action Force (FARN) under the Post-Fukushima resilience programme and with using digital technology to train maintenance workers. WANO also issued four recommendations (two relating to Corporate Leadership and Governance, one dealing with Monitoring and Oversight of Contractors and one to do with Completeness of Independent Oversight), which will lead to an action plan.

I-10.3. Safety improvements at EDF's NPP – Periodic Safety Review

The French Environmental Code (Articles L593-18 and L593-19) requires a periodic review for each nuclear installation at a frequency that does not exceed 10 years (the decree authorizing the creation of the installation may require a more frequent frequency). [24] [25]

The terms of this review were defined by Decree 2007-1557 of 2 November 2007. Since 4th March 2019, the requirements of this decree have been codified in the Environment Code: Article R593-62 reproduces the requirements relating to this periodic safety review. [26]

The periodic safety review process consists of several steps. Some of them are the responsibility of the operator (EDF) and some others of ASN. This process can be represented as shown in Fig. I-10-2:

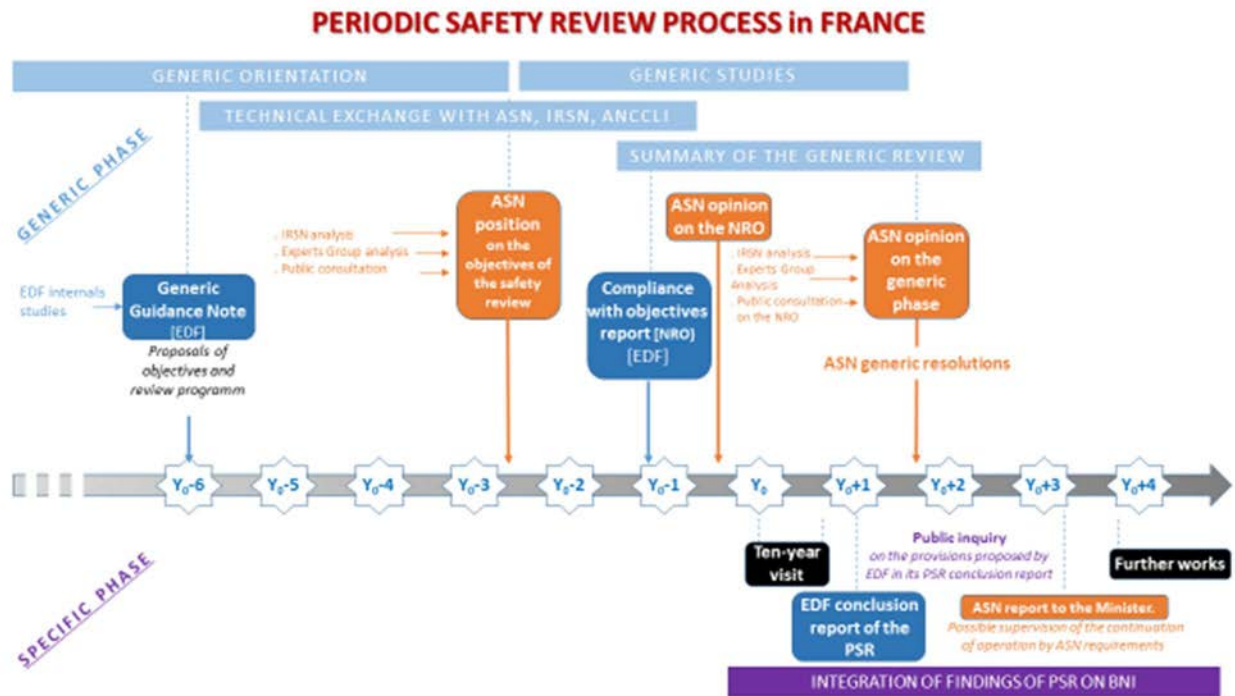


FIG. I-1-2. Periodic safety review process in France.

Compliance review

The compliance review consists in comparing the actual state of the installation with the applicable safety standards and regulations, including in particular its decree authorizing the creation of the installation and the requirements of the ASN.

This 10-year compliance review does not relieve the operator from its obligation to ensure the ongoing compliance of its facilities. This is regularly monitored by the ASN through frequent inspections it carries out at each site.

Safety reassessment

Safety reassessment aims to assess the safety of the installation and improve it in terms of:

- French regulations, objectives and the most recent safety practices, in France and abroad;
- The feedback of experience in the operation of the installation;
- Feedback from other nuclear installations in France and abroad.

The ASN shall decide, after possible consultation with the Standing Group of Experts for the Safety of Nuclear Reactors (GPR), on the list of topics selected for safety reassessment studies and the associated objectives during the so-called guidance phase of the periodic review. At the end of the studies carried out by EDF on each of the selected topics, modifications allowing safety improvements are defined. They will be deployed during the ten-year visit of the reactor.

The implementation of the improvements resulting from the periodic review

Decennial visits, which are long term shutdown, are privileged moments to implement the provisions resulting from the periodic review. In determining the schedule for ten-yearly inspections, EDF have to take into account the deadlines for carrying out hydraulic tests set by the regulations on nuclear pressure equipment and the ten-year periodicity of periodic inspections. The ASN verifies that the modifications, which will be implemented during the 10-year reactor visit, meet the objectives of the review.

The submission by the operator of a review finding report

At the end of the 10-year inspection, the operator shall send the ASN a report of the conclusions of the periodic review. In this report, the operator takes a position on the regulatory compliance of its installation, as well as on the modifications made to correct any discrepancies identified or improve the safety of the installation. The ASN shall communicate its analysis of the report to the Minister responsible for nuclear safety and may lay down additional requirements for the operator. This report shall be made public.

Practically, the chronology for implementation of the process for the different PSRs of the French Nuclear fleet is shown in Table I–10-1.

TABLE I–10-1. CHRONOLOGY FOR IMPLEMENTATION OF THE PROCESS FOR THE DIFFERENT PSRS OF THE FRENCH NUCLEAR FLEET.

| | PSR 1 (10 years) | PSR 2 (20 years) | PSR 3 (30 years) | PSR 4 (40 years) |
|------------------------|---------------------|---------------------|---------------------------------|---------------------|
| 900 MW (34 units) | | done | 2009 to 2020 (25 units done) | 2019 to 2030 |
| 1300 MW (20 units) | | done | 2015 to 2024 | 2025 to 2034 |
| 1450 MW – N4 (4 units) | done | 2019 to 2022 | 2029 to 2032 | 2039 à 2042 |

Safety improvements at EDF's NPP – Example of the 4th PSR of the 900MWe reactors

EDF's industrial goal for the preparation for the future of the nuclear fleet rests primarily on the following strategic areas:

- The implementation of technical conditions allowing the extension of the operational life of nuclear power plants beyond 40 years;
- Continued safety improvements, primarily by integrating lessons learned from the Fukushima Daiichi NPP accident in Japan;
- Implementation of a preventive policy with respect to ageing or obsolete equipment.

The first point can be illustrated by the 4th PSR of the 900 MWe NPPs. For those NPPs, EDF proposes to work towards the safety objectives of the most advanced technology ("generation 3") EPR-Flamanville 3 nuclear power plants, in accordance with the request of the Nuclear Safety Authority (ASN).

This proposal translates into safety objectives to strengthen the robustness of nuclear installations and aims to:

- Avoid the use of population protection provisions (preventive taking of iodine tablets, sheltering, evacuation) in the event of an accident without melting the reactor core;
- Obtain a level of risk of residual core meltdown of the same order as the target targeted by the Generation 3 reactors;
- Reduce the risk of early or significant radiological releases in the event of a core meltdown accident in order to eliminate the risk of lasting effects in the environment;
- Make residual the risk of uncovering assemblies stored under water in the spent fuel pool
- Avoid, in situations of extreme hazard, major releases and lasting radiological consequences in space and time. The answer to this objective is provided by the implementation of a set of design and resilience resources (in particular the Hard Core and the Nuclear Rapid Action Force - "FARN") robust to extreme aggressions with the lowest possible level of plausibility and meeting the risk reduction objectives set out above.

The human and financial investment associated with these ambitions is significant and is carried out by EDF in a manner proportionate to the safety and environmental protection issues and under economically acceptable conditions.

The answer to this general objectives requires that installations comply with the applicable rules, which are based on:

- The implementation of targeted controls through the examination of the compliance of the NPPs;
- The conduct of a programme of additional investigations to identify and remedy potential weaknesses in the maintenance program;
- The implementation of a review programme covering core cooling and backup systems, as well as support functions;
- The exhaustive analysis of the treatment of compliance deviations.

This general objective is broken down into specific objectives for reassessing the level of nuclear safety, divided into four main safety themes:

- Accidents without core meltdown
 - Comply with the safety criteria for accident studies in the safety report.
 - Tend towards levels of radiological consequences that do not require the implementation of provisions to protect the population.
- Hazards
 - Ensure the robustness of the installations at levels of hazards reassessed during the review and in accordance with international recommendations (WENRA).
 - Aim for a core meltdown risk of a few hundred thousandths (1/100,000) per year of reactor operation for all initiators.
- Spent fuel pool
 - Make it extremely unlikely that fuel assemblies will be uncovered during accidental draining and cooling loss.
- Core meltdown accidents
 - Make the risk of early and significant releases extremely unlikely.
 - Avoid long-lasting effects in the environment.

In addition to the risk component (incidents and accidents) covered by nuclear safety, EDF meets the objectives of compliance with the rules of the "detrimental" component (normal operation) of the periodic reviews, through multi-year assessments, water abstraction and consumption, discharges, pollution and waste management. The reassessment of the control of detrimental is the subject of improvement actions as well as an update of the assessment of the inconvenience that the plant presents to people and the environment.

The 4th periodic review of 900 MWe reactors also includes a section on "continued operation" that covers the control of ageing, obsolescence and the maintenance of equipment qualification under accident conditions.

It is based on a major programme to verify the ability of equipment to perform its functions and with the replacement of some of these equipment.

The 4th periodic review of the 900 MWe nuclear power plants marks the 40th anniversary of their operation; it is accompanied by a significant improvement in the nuclear safety of each of the reactors concerned. Studies and related work represent around ten million hours of engineering work and around 7 billion euros of work.

I-11. GERMANY

I-11.1. Regulatory framework

In Germany, safety improvements in Nuclear Power Plants have been implemented on a continual basis. Due to the German legislation the state-of-the-art in science and technology is mandatory for all decisions made by the regulator during licensing and oversight. The fundamental safety objective 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). 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 of the Atomic Energy Act (Atomgesetz – AtG) in Germany. In addition, § 7d of the Atomic Energy Act (Atomgesetz – AtG) requires that after a license has been issued the licensee is obliged to implement the necessary safety improvements. In the “Safety Requirements for Nuclear Power Plants”, priority to safety is further specified as follows:

- The licensee is responsible to assure plant safety. He shall give preference to meeting the safety objective over other plant operational objectives.
- The prime objectives of the integrated management system (IMS) are specified as:
 - the guarantee of safety;
 - the continuous improvement of safety;
 - the promotion of safety culture.

The national nuclear regulations in Germany have been constantly developed and adapted to the progressing state of the art in science and technology. Bundesministerium für Umwelt (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). 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 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. As part of the current state of the art in science and technology, IAEA safety standards are considered in the revision of the national nuclear regulations. The BMU also requests its advisory commissions RSK (the German Reactor Safety Commission), ESK (the German Commission on Radiological Protection) and SSK (the German Nuclear Waste Management Commission) to comment on selected developments and events in the area of nuclear safety and to make recommendations. 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. 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 - Weiterleitungsnachrichts) provided by the expert organization GRS (Gesellschaft für Anlagen- und Reaktorsicherheit gGmbH).

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. 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 [7].
- 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 commission and review committees (CSS, EPRESC, 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 [21].

In accordance with § 19a of the Atomic Energy Act (Atomgesetz – AtG) the licensees are required to conduct and to evaluate a periodic safety review (PSR) every 10 years of the installation in their responsibility, and to improve on this basis the nuclear safety of the installation continuously. They consist of a deterministic safety status analysis, a probabilistic safety analysis and a deterministic analysis of the physical protection of the installation. During the process the nuclear power plant under consideration is benchmarked against the latest state-of-the-art in nuclear safety. By the processes described above 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. Such safety improvements have been carried out both after major accidents in NPPs (such as TMI NPP, Chernobyl NPP, or Fukushima Daiichi NPP accident), 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 level of safety commensurate with the most recent state-of-the-art in science and technology.

I-11.2. Identification of safety improvements

That text summarizes the activities in Germany regarding the development and optimization of the severe accident management concept of German nuclear power plants (NPPs) as an example of a driver for the enhancement process among others (operational experience, PSR).

This Process has been performed for both PWR and BWR respectively between the 1980-ties and up to now.

I-11.3. Drivers for the enhancements process

The main drivers for both the development and optimization of the severe accident management concept for German NPPs have been the severe accidents in Three Miles Island (TMI) NPP, Chernobyl NPP, and Fukushima Daiichi NPP.

The discussions for the development of the severe accident management for German NPPs in order to be able to handle beyond design basis accidents started after the Three Miles Island severe accident. Later, the development of SAM was pushed especially by the severe accident in the Chernobyl NPP. The procedure of the development of the SAM concept is described in the country report of Germany, published in the frame of the European stress test (ENSREG) [27] and will be described in the section below. Finally, the result was a SAM concept for the German NPPs using both measures for prevention and mitigation. The focus of that concept lay on the preventive part of SAM. The realization of that concept inside the plant has been done in the 1980s and the 1990s.

The last driver for an optimization of the existing SAM concept of German NPPs was the Fukushima Daiichi NPP severe accident. After Fukushima Daiichi NPP accident the robustness of the German NPPs against Fukushima like conditions (external hazards, Station Blackout, loss of service water cooling chain) has been re-assessed 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 (RSK).

I-11.4. Selection process of safety improvements

As a response to the severe accidents at Three Miles Island NPP and especially after the Chernobyl NPP accident in 1986, the basis SAM concept for German NPPs has been discussed inside 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 county report of Germany (BMU) [28]. Selected parts of the German country report are given below:

First requirements for a Severe Accident Management (SAM) programme regarding beyond-design-basis events starting from power operation only were published in autumn 1988 after intensive discussions within the RSK [29]. The concept was called “Anlageninterner Notfallschutz,” and 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 that was recommended and installed very early, in the late 1980s [30], [31]. 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 [29]
- Filtered venting of PWR containment, 218th meeting, 17-12-1986 [29]
- Filtered venting of BWR containment, 222nd meeting, 24-06-1987 [29]

- N2 inertization of BWR containment, 218th meeting 17-12-1986 [29]
- Start of detailed discussions about accident management 1987/88;
- a. 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 [32]
- Hydrogen recombination, RSK Position Paper, 314th meeting, 17-12-1997 [33](Discussions since around 1987 regarding igniters or passive autocatalytic recombiners or dual concept)

Additional information was compiled by KTA in 1996 [34].

The final RSK recommendation regarding a Severe Accident Management Programme was published in 1992 [32] 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. The basic principles of the SAM-concept are described below:

...

Later on in 1997, another RSK recommendation was published [33], dealing with hydrogen countermeasures, especially the installation of PARs in large dry German PWR containments. Important aspects are described below:

...

Filtered venting of PWR containments was decided already at the 218th RSK meeting, 17-12-1986 [29]. Important aspects are described below:

...

For BWRs, N2 inertization of the containment was implemented where possible [29]. Important aspects are described below for BWR type 69:

...

Filtered venting of BWR containments was decided at the 222nd meeting, 24-06-1987 [29]. Important aspects are described below for BWR type 69:

...

The containments of the BWR type 72 differ considerably from those of the BWR type 69 (see for more details chapter 1). The licensee of BWR type 72 developed an inertization/recombination concept and a pressure suppression concept that took into account the differences of the plant design and considered the RSK recommendations. The concept was separately discussed and approved by the RSK [30] and thereafter realized. Details of installed Accident Management measures can be found in chapter 1 along with the general PWR and BWR plant description, in the individual Licensees reports and as well in the following chapters.

In addition to these recommendations of the RSK the following documents are provided for defining alert criteria to be used in case of an emergency and for the organization of external provision:

– RSK/SSK Recommendation: “Criteria for alerting civil protection authorities through operators of nuclear facilities” (“Kriterien für die Alarmierung der Katastrophenschutzbehörde durch die Betreiber kerntechnischer Einrichtungen”), published July 2004 [35]

– Federal government - Länder committee for nuclear technology: “General Recommendations for the Disaster Control in the Vicinity of Nuclear Facilities” (“Rahmenempfehlungen für den Katastrophenschutz in der Umgebung kerntechnischer Anlagen”) issued 01.01.1989, updated 27.10.2008, [36]

...

The efforts undertaken by the Licensees in the beyond-design-basis and severe-accident area related to the implementation of SAM Programs since the late 1980s has been on a voluntary basis first. The licensees agreed to implement the respective RSK recommendations. In the context of the now legally required Periodic Safety Reviews (PSR) every ten years the defence in depth 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.

Extensive documentation of all the measures implemented and especially of the hardware modifications performed in German NPPs both in the preventive and mitigative domain can be found in the reports of the German government to the Convention of Nuclear Safety, e. g. the report of 2005 [37].

The BfS on behalf of the BMU has compiled an overall status report of the implementation of AM-measures as recommended by the RSK and requested by the BMU. [38]

After Fukushima Daiichi NPP accident the robustness of the German NPPs against Fukushima Daiichi NPP accident like conditions (external hazards, Station Blackout, loss of service water cooling chain) has been re-assessed 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 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),
- 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;
- Need of a SAMG Concept.

I-11.5. Outcomes identified of safety improvements

I-11.5.1. Up to the Fukushima Daiichi NPP accident:

In the 1980s and 1990s most of the German NPPs has been equipped with severe accident management measures based on the requirements issued by the RSK. These measures have been both measures for prevention and mitigation of severe accident sequences. For most of the German PWRs in the 1990s following main measures have been realized:

- Secondary side bleed and feed by feedwater system and/or mobile pump;
- Passive feeding of steam generators from feedwater lines and feedwater tank;
- Primary side bleed and feed by installed ECCS systems;
- Restoration of AC power supply (e. g. third grid connection by cable);
- Increased capacity of the batteries;
- Secured containment isolation;
- Passive autocatalytic recombiners (PARs) (mitigative measure);
- Filtered containment venting (mitigative measure);
- Sampling system containment (mitigative measure).

The measures listed above are documented in the “Emergency Operating Manual” of the plants.

For BWR-72 the status-quo of the severe accident management at the beginning of 2000 was:

- Diverse depressurization of RPV;
- Diverse injection into RPV with pumps from different systems (high-pressure, medium-pressure, low-pressure);
- Mobile pump for feeding RPV;
- Restoration of AC power supply (e. g. third grid connection by cable);
- Increased capacity of the batteries;
- Secured containment isolation;
- Passive injection from feedwater system;
- N₂ inertization of wetwell;
- Filtered venting of wetwell (mitigative measure);
- Passive autocatalytic recombiners (PARs) in both wetwell and drywell (mitigative measure);
- Sampling system containment (mitigative measure).

The status-quo in 2010 of the severe accident management of German NPPs is shown in Table I-7 (PWR) and Table I-8 (BWR). The tables have been modified from the original tables documented in in chapter 3.1 and 3.2 of the KTA report KTA-GS-66 /KTA 97/[39].

TABLE I-7. SEVERE ACCIDENT MANAGEMENT MEASURES AND THEIR REALIZATION OF GERMAN PWR PLANTS (STATUS-QUO AT 2010; MODIFIED FROM KTA-GS-66) [39]

| 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 | Ensuring 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 | R/1995 | design | design |
| 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 | design | design |
| 3 | Activity retention and ensuring 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 | Limitation of hydrogen formation | 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 isolation | R/1991 | R/1988 | R/1991 | R/1990 | R/1991 | R/1991 | R/1991 | design | R/1990 | R | R | design | design |
| 3.4 | Control room supply air filtering | R/1990 | R/1992 | R/1989 | R/1991 | R/1989 | R/1989 | R/1992 | R/1990 | R/1990 | R/1998 | R/1989 | design | 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 | 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 | design | R/1991-92 | R/1989-93 | R/1991 | design | R/1995 | design | design ⁴⁾ | R | R/1989 | R/1988-90 | R/1988 |

| No. | Measure | KWO ⁹⁾ | KKS ¹⁾ | KWB A | GKN 1 | KWB B | KKU | KKG | KWG | KKP 2 | KBR | KKI 2 | KKE | GKN 2 |
|-----|--|-------------------|-------------------|--------|--------|--------|--------|--------|--------|--------|--------|----------------------|--------|--------|
| 4.3 | Restoration of grid supply | design | R/1990 | R/1990 | R/1989 | R/1990 | R/1989 | R/1990 | R/1990 | R/1989 | R/1995 | R | R/1996 | design |
| 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, 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 recombiners that had originally planned for the 2001 overall maintenance and refuelling outage was not carried out after all due to the fact that 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 is only planned after conclusion of the corresponding licencing procedure for KWB B.
- 7) Not planned.
- 8) Utilisation of a 20-kV connection to the Isar hydroelectric power plant chain (as for KKP 1, in that case a 6-kV connection).
- 9) Final shutdown KWO on 1/05/2005, the first licence for decommissioning and dismantling was granted on 28/08/2008.

TABLE I-8. SEVERE ACCIDENT MANAGEMENT MEASURES AND THEIR REALIZATION OF GERMAN BWR PLANTS (STATUS-QUO AT 2010; MODIFIED FROM KTA-GS-66) [39]

| 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 | Ensuring 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 and ensuring containment integrity | | | | | | |
| 3.1 | Filtered containment venting | R/1988 | R | R/1989 | R/1988 | R/1990 | R/1990 |
| 3.2 | Containment inertization | 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 | design | design |
| 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 | design | R/1987-88 | R/1990 | design | design |
| 4.3 | Restoration of grid supply | R | R | design | 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, 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 (as for KKP 2, in that case a 20-kV connection).
- 4) So far, no application has been submitted yet to the competent licensing and supervisory authority.
- 5) So far, no application has been submitted yet to the competent licensing and supervisory authority. The utility is in the phase of working out a concept, a realisation is planned for the year 2010/2011.
- 6) In connection with hydrogen recombiners (R/1999-2000) due to only partial inertization of the containment in the pressure suppression pool.
- 7) Not planned.

I-11.6. After the Fukushima Daiichi NPP accident:

Based on the recommendations of RSK derived from the national and European stress test mentioned above, the following additional SAM measures have been developed and realized for the German NPPs (PWR and BWR):

PWR:

- Two additional mobile Emergency Diesel Generators (EDGs);
- Two separated bunkered connecting pins for the mobile EDGs;
- A 'shortened cold chain' has been realized for PWRs without a second ultimate heat sink (mobile pump connected to the nuclear service water system, suction can be done from both the pool of the intake structure or the pool of cooling tower and discharge of the heated coolant into environment);
- External feeding of the spent fuel pool by a mobile pump from outside the reactor building and SFP cooling by evaporation of SFP water;
- Re-assessment of the robustness of the plants, internal flooding of the reactor building annulus;
- A new SAMG concept.

BWR:

- Two additional mobile Emergency Diesel Generators (EDGs);
- Two separated bunkered connecting pins for the mobile EDGs;
- Early opening of the three motor-driven pressure relief valves (discharge into suppression pool);
- Mobile "Hydrosub" pump for feeding the RPV via the nuclear service water system and heat removal can be done via wetwell and FCV system;
- External feeding spent fuel pool by a mobile pump from outside the reactor building and cooling by evaporation of SFP water;
- Additional PARs above spent fuel pool outside containment but inside reactor building;
- Re-assessment of the robustness of the plant, filtered containment venting, external flooding;
- A new SAMG concept.

I-12. HUNGARY

Development of the SAM strategy and post Fukushima safety improvements at PAKS NPP

I-12.1. Hungarian regulatory framework

The national competent authority is the Hungarian Atomic Energy Authority (HAEA). Regulatory oversight of nuclear facilities, peaceful use of atomic energy, security of nuclear and other radioactive materials and equipment that generate ionizing radiation, safety of packaging of nuclear and other radioactive materials is the main task of the HAEA.

Hungarian legal background is illustrated in Fig. I-12-1. On the basis of this legislation, acts and guides Authority can judge petitions from grantee (NPP).

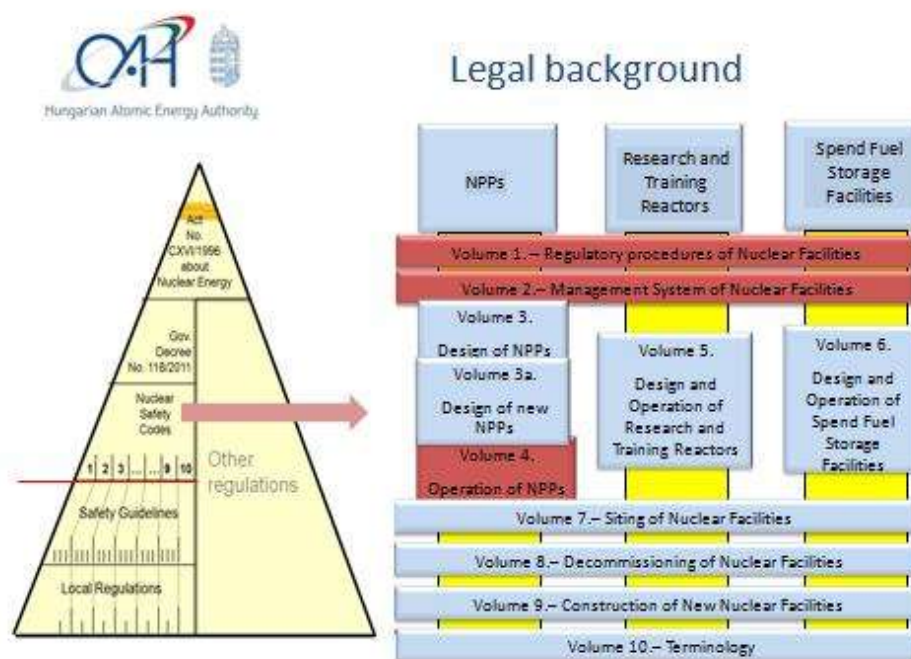


FIG. I-12-1. Legal background.

The fundamental legislative framework based on the Act CXVI of 1996 on atomic energy and the Govt. decree 118/2011. (VII. 11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities.

The Hungarian Parliament approved the Act on Atomic Energy in December 1996 (hereinafter referred to as the Act on Atomic Energy), which entered into force on July 1, 1997. The Act on Atomic Energy as amended several times considers all regulatory-related and operational experience gained during the construction and operation of Paks NPP, it considers the technological development, all international obligations, and obviously integrates the IAEA requirements. The main criterion and key point of this is: "In the use of nuclear energy, safety has priority over all other aspects". Those who drafted the Act on Atomic Energy utilized the recommendations of the European Union, the IAEA and the OECD NEA.

The Act on Atomic Energy requires the regular revision and update of the nuclear safety requirements for the application of atomic energy taking into account scientific results and international experience. The relevant governmental decree stipulates the periodicity as 5 years. The main characteristics of the Act on Atomic Energy are as follows:

- 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 HAEA;
- Declaration of the organizational and financial independence of the HAEA;
- 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 HAEA the right to impose fines if rules are violated.

The Nuclear Safety Code was further developed in line with the EU nuclear safety directive, Council directive 2009/71/EURATOM of 25 June 2009 [40] establishing a Community framework for the nuclear safety of nuclear installations, the Convention on the Physical Protection of Nuclear Materials, the safety requirements and recommendations published by the IAEA in the last five years as well as the WENRA reference levels. The nuclear safety requirements of the use of atomic energy in reactor facilities are regulated by the NSC issued as annexes of Govt. decree 118/2011. (VII. 11.) In addition to the continuous development of IAEA requirements and recommendations, the modification of the national regulations were performed more often than the required five year frequency.

Govt. Decree 118/2011 (VII. 11.) on the nuclear safety requirements of nuclear facilities and on related regulatory activities regulates the authorization of safety improvement modifications. The SSCs are classified base on their impact on safety. Different actions and different requirements belong to the categories. In order to enforce the differentiated approach, the modifications shall be categorized in accordance with their safety significance:

- Those modifications shall belong to Category 1, in which the modification has a significant effect on the radiation risk of the population and the persons present at the site of the nuclear power plant, or the modification changes those principles and conclusions that were used for the design and licensing basis of the nuclear power plant, or the modification changes the scope of events leading to TA3- 4 operating conditions and the way they take place, or the modification may lead to the change of the fundamental operational regulations of the nuclear facility.
- Those modifications shall belong to Category 3, in which the modification shall not have a safety consequence.
- Category 2 means any remaining modifications with a moderate impact on safety.

The fundamental rule of the modification is that the modification shall not decrease nuclear safety.

In accordance with the Atomic Act, HAEA requires for Paks NPP to have FSAR as an updated description of the current situation and also to do PSR in every 10 years. PSR assessment is covering all changes in the last 10 years and planned modifications in near future in fields as follows:

- Siting and design (taking into account the construction of Paks Unit 5 and 6);
- Condition of SSCs;
- Environmental Qualification (taking into account the global warming);
- Aging management (equipment, management);
- PSA and DSA (appearance of new external hazards);
- Maintenance of the technical conditions;
- Performances, operational experience, safety indicators;
- Science and technology development;
- Organizational and Administration (Methods and Procedures);
- Human factors;
- Emergency (taking into account the construction of Paks Unit 5 and 6);
- Radiation and Physical protection (taking into account the construction of Paks Unit 5 and 6).

The last PSR was fulfilled in 2017 for what a new Regulatory Guide was published, containing the WENRA references and the new IAEA considerations (SSG – 25 [4]). Some new topics were appeared as extended with experience of Fukushima Daiichi NPP accident, general and unit specific issues. The most important modifications of the coming years are related to the PSR.

I-12.2. Introduction of the MVM Paks Nuclear Power Plant

MVM Paks Nuclear Power Plant is located almost in the centre of Hungary, along the Danube; see Fig. I-12-2.

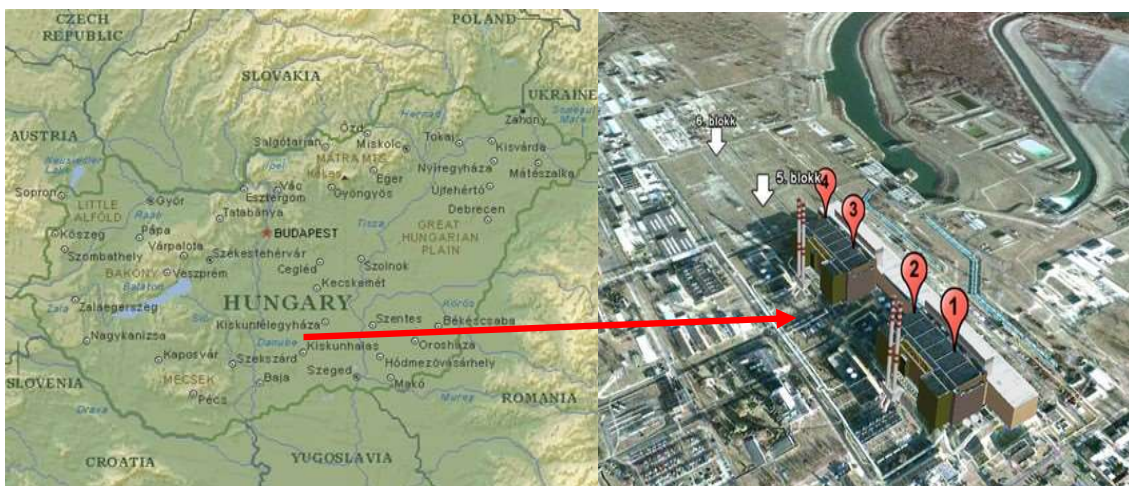


FIG. I-22-2. Paks NPP

MVM Paks Nuclear Power Plant Ltd operates 4 pressurized-water nuclear units of the type VVER-440/V-213; 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 units were commissioned in 1982, 1984, 1986 and 1987 and designed for 30 years. The original nominal thermal power of each unit 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 increased to 1485 MW and the electric power to 500 MW. Owing to the Operation Time Extension project, all 4 Units can operate 50 years long.

Paks NPP always made great efforts to increase safety of units, therefore a lot of safety improvements were implemented based on PSAR and Level 1 PSA results, the evolution of which is illustrated in Fig. I-12-3. Hard modifications were installed after seismic calculations in 2005 why the results are getting better again. Now decrease of risk from external events is a biggest challenge for nuclear experts.

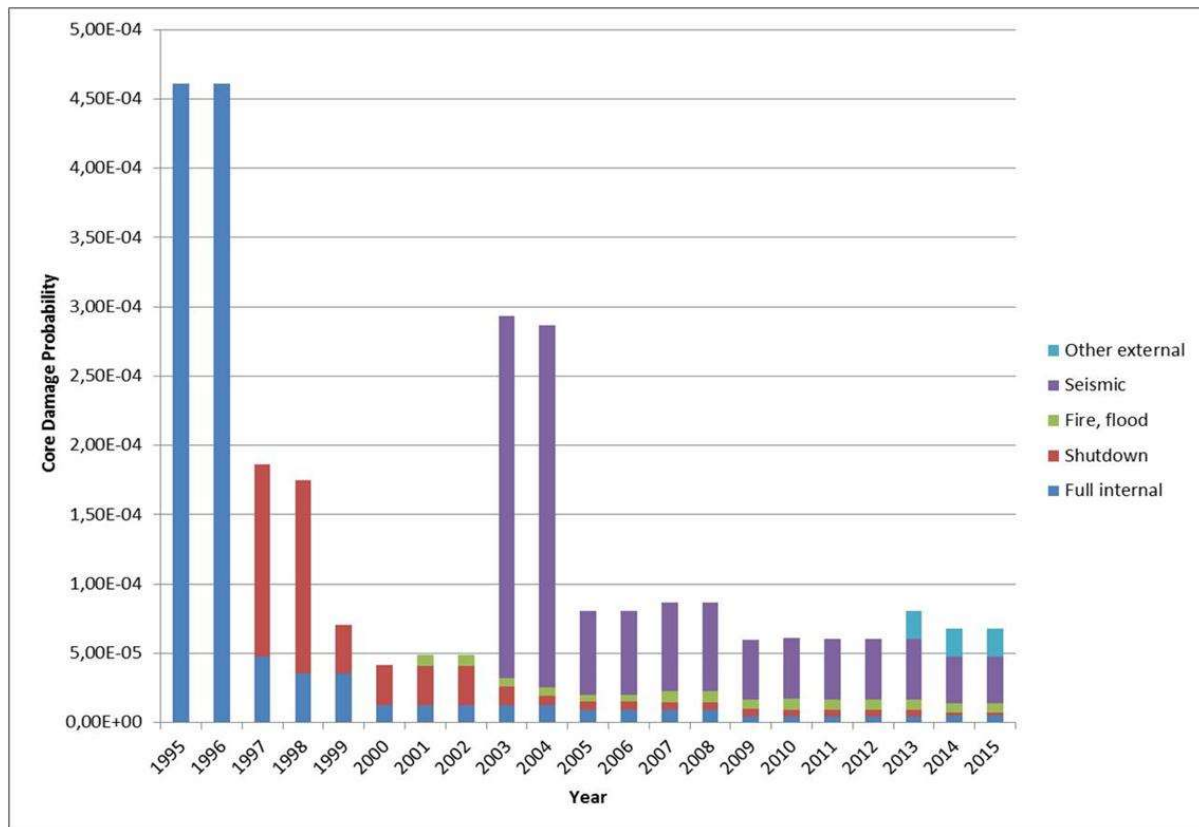


FIG. I-12-3. Decrease of Level 1 PSA results from the beginning.

Level 2 PSA was performed for the Paks NPP in period 2000-2003. The main objective of the study was to quantify the frequency of large radioactivity releases due to severe accidents and to develop risk-informed recommendations for accident management strategies and measures. The study identified the important plant damage states that were used for severe accident and containment event tree analyses in the Level 2 PSA of the plant. Analyses covered all sources of potential radioactivity release, not only the reactor core. All plant operational states (full power, low power and shutdown) and all types of important initiating events (internal and external) were within the scope of the study. Results of the Level 2 PSA served as a basis for development of mitigative accident management strategies. Level 2 PSA and a following uncertainty study provided the basis to establish the main strategy for SAMG. After stress test post Fukushima safety improvements were developed and mostly already implemented.

I-12.3. Specific design features with SAM implication¹⁸

Paks NPP has 4 units of VVER-440/213 type with bubbler condenser. These reactors have extremely large water reserves both in the primary and the secondary side. Expressed in terms of time, the reserves provide a longer time span, than it is usually available for reactors of other design. The relatively small reactor core, with a very low power density is situated in a long reactor vessel containing a large amount of iron structures below the core. The six-loop arrangement leads to a high volume of the primary circuit and this ensures that the core or the pressure vessel remains intact for a longer period of time even if the core cooling has been lost. The spent fuel pool is located in the reactor hall, which is outside of hermetic compartments.

Specific design features with SAM implication (+ or -) are characteristic to the fuel assemblies and the core, RPV, primary coolant loops with the horizontal steam generators, reactor cavity, and the containment, listed below.

- Core

There are absorber assemblies equipped with fuel follower assemblies in the VVER cores. In case of a tripped reactor about the 10 % of the assemblies are below the bottom of the core. This arrangement is influencing the core melt progression and relocation process. The relatively small core contains large amount of zirconium, thus the hydrogen generation and the threat of hydrogen combustion is a major vulnerability of the VVER units.

+ Vessel

The reactor pressure vessel has an elliptical bottom. In case of a core melt accident the highest elevation of the relocated debris bed will be in the cylindrical part of the vessel. That affects the outside heat transfer processes in case of external cooling, and the debris bed dynamics in the bottom of the reactor vessel would also be influenced. The RPV has relatively high surface area compared to the low decay power, which makes the external cooling more effective.

- Cavity

The reactor cavity has a relatively narrow design with a door at the bottom level leading to the non-hermetic compartments. The flow-paths between reactor cavity and the rest of the containment are small. Due to this arrangement the pressure transients in the cavity can lead to peak pressures that threaten the cavity wall or the access door integrity. Since these are containment boundaries, such loss of integrity would represent a containment failure. As designed, the cavity remains dry during any type of accident and presently there are no flow-paths available to flood the cavity.

¹⁸The sections I-12.3 – I-12.7 were presented as “Development of the SAM strategy for Paks NPP on the basis of Level 2 PSA” at OECD Workshop and published in “OECD (2010), Implementation of Severe Accident Management Measures, ISAMM 2009, Workshop Proceedings, Vol. I”

+ Loops

The horizontal SG design, in conjunction with the loop seals in the hot legs would very likely preclude any counter-current gas flow in the hot legs. This feature strongly influences the potential of induced primary system failures.

Containment

+ The containment is a system of interconnected compartments. There is a pressure suppression system consisting of a vertically mounted series of bubbler condenser trays and air traps to maximize condensation efficiency. This bubble condenser tower is connected to the main confinement by a large flow area opening and it contains about 1200 m³ of water.

- The compartment walls are approximately 1.5 m thick and have an internal steel lining. The design pressure of the containment is 2.5 bar. The mean overall containment capacity calculated by the aggregation of the capacities of all the structural elements was computed to be 0.35 MPa, and the “high confidence of low probability of the failure” capacity as 0.235 MPa. The containment integrity can be threatened by combustion of hydrogen inventory of about 45-50 % Zr oxidation fraction (350-400 kg of H₂). This value is within the possible range of in-vessel hydrogen production.

Earlier the containment was not equipped with any hydrogen control measures designed for severe accident conditions. Random ignition sources sufficient to initiate deflagration will probably be present, but a random nature of the ignition process is expected due to the high uncertainty over the time and availability of ignition sources.

+ The containments of VVER-440 units have relatively high (14.7 %vol/day) design leakage rates. Recently a lot of efforts have been made to improve the leaktightness of the containment. Now, leakage rates around 5-10 %/day are quoted for these units. Despite the improvements, management of the environmental releases due to the pre-existing leakage has a higher priority in the accident management, while containment overpressurization remains a relatively low level concern.

If the RPV failure occurs at high primary system pressure, then different energetic events might lead to the loss of containment integrity. Therefore, primary system pressure reduction is a measure of major importance.

I-12.4. Summary of Level 2 PSA results – Decision Making Process

Level 2 PSA analyses can be used for the development of a severe accident management strategy with the identification of the accident sequences that result in core damage, containment failure (or containment by-pass) and the release of fission products into the environment. It is assumed as ideal tool to assess the risk impact and the risk reduction efficiency of selected SAM actions and to identify reasonable design basis for mitigative systems. Decision making process described below could be used as a good example for other plants to develop SAM strategy.

The performed severe accident analyses for Level 2 PSA [41] were the basis for the development of SAM. Taking into account the frequency of different release categories, Level 2 PSA and uncertainty studies [42] could show the importance of the elements of the accident management procedures.

At first, containment failure modes caused by different physical phenomena were determined (see Table I-12-1). All possible reasons of the containment failure modes were assessed, and their probabilities were calculated. The total sum of the probability of different containment failure modes exceeded the value 10^{-6} 1/year. Early containment failure is caused mostly by hydrogen burn. Early release from the containment can also occur in case of steam generator tube or collector break (by-pass). These two elements are the main contributors to the early large release that need to be avoided.

Environmental consequences of the late enhanced containment leakage are smaller, than that of the early failures, but the frequency of late failures is higher. Late containment leakage is mainly caused by the corium melt attack on cavity door. Possible accident management measures for prevention of containment failure are shown in Table I-9.

TABLE I-12-1. CONTAINMENT FAILURE MODES AND THEIR REASONS

| Containment failure modes | Main reason of the containment failure (physical phenomena) |
|---|--|
| High pressure RPV rupture | Failure of primary depressurization (human error, valve failure) |
| By-pass | Steam generator tube/collector rupture |
| Early containment rupture | Hydrogen burn |
| Early enhanced containment leakage | |
| Late containment rupture | Containment slow overpressurization |
| Late enhanced containment leakage | Cavity door seal failure due to high temperature |
| Early containment rupture with spray | Hydrogen burn |
| Early enhanced containment leakage with spray | |
| Late containment rupture with spray | |
| Late enhanced containment leakage with spray | Cavity door seal failure |
| Intact containment | |
| Intact containment with spray | |

TABLE I-12-2. POSSIBLE ACCIDENT AM MEASURES

| Main reason of the cont. failure (physical phenomena) | Possible accident management measures |
|--|---|
| Failure of primary depressurization | SAMG |
| Steam generator tube/collector rupture | Bleed from ruptured SG to the containment |
| Hydrogen burn | Hydrogen recombiner, igniter or inerting |
| Cavity door seal failure | Isolation of room A004 or prevention of RPV failure |
| Containment late overpressurization | Filtered venting and/or spray |

In accordance with the identified main challenges, two SAM strategies were elaborated [43] (see Table I-12-3):

- The 1st strategy included hydrogen treatment using recombiners, filtered venting and prevention of the reactor cavity door damage as accident management measures;
- The 2nd strategy included in addition reactor cavity flooding for in-vessel retention and also for protecting from the basemat melt-through.

TABLE I-12-3. POSSIBLE AM STRATEGIES

| | Base case | Strategy I | Strategy II |
|--|----------------|------------------------|--|
| Prevention of RPV failure | ECCS recovery | ECCS recovery | ECCS recovery + reactor cavity flooding |
| Hydrogen treatment | - | 30 recombiners | 30 recombiners |
| Limitation of radioactive releases | Spray recovery | Spray recovery | Spray recovery |
| Prevention of containment overpressurization | - | Filtered venting | Filtered venting |
| Safe integrity of the reactor cavity | - | Isolation of door A004 | Solved by cavity flooding |
| (Ex-vessel cooling of the melt) | - | - | (Not challenged) |

Both strategies were investigated, and the corresponding probabilities were quantified. There were only minor differences between the strategies in terms of the probabilities of the early containment failure. However, if other considerations, such as the cost of the modifications and mitigation of the consequences are taken into account, then the two options differ substantially.

As a summary evaluation of the results the following conclusions were drawn:

- Due to AM measures, the distribution of release category frequencies is significantly modified compared to the present state.
- The frequency of most severe high-pressure sequences is not influenced by the AM measures; however, their frequency is not significant ($<10^{-7}$ /unit/year).
- The frequency of early containment failures can be reduced to 1/3 of by the AM measures.
- The frequency of early containment failure with spray can be reduced to 1/4 by hydrogen treatment.

The two AM strategies do not differ significantly in terms of atmospheric release; however, a difference can be found in the basemat failure. The reactor cavity flooding protects the basemat from melt through in many cases, decreasing its frequency on two order of magnitude. The cavity door protection does not affect the basemat failure probability. As a result of the studies a unified strategy has been developed. The main points of the proposed strategy were:

- Hydrogen mitigation with recombiners;
- In-vessel melt retention by flooding the cavity;
- Using filtered venting or containment cooling system to prevent late overpressurization.

After the quantification of event trees and the binning of end states into release categories, frequencies of the various release categories [44] were calculated (see Table I-12-4.). It was found that the released amount of ^{137}Cs isotopes and the doses at 1 km distance from the plant in the early and later phase of the accident are more or less proportional with each other in the case of release categories 1-13. In the other categories due to the extended decay periods the iodine content of the fuel is much less, and the released caesium activity is no more proportional with the consequences. For these cases the releases have milder health effects.

It is doubtless that releases belonging to release categories 1, 2, 3, 14, 15, 16 and 17 have to be considered to large radioactive releases while the release categories 8, 9, 10, 12 and 13 are really harmless.

Level 2 PSA analyses were performed also for shutdown mode operation and for the SFP. Shutdown mode analyses were arranged in two groups by the state of the reactor: open and closed reactor.

- Closed reactor: In these cases the PDS and Accident Progression Event Tree APET for nominal power were used. For mitigation the SAMG identified for nominal power operation can be used.
- Open reactor: Different initial events (heavy load drops, loss of decay heat removal from the core, before and after refuelling) were analysed. Releases in these cases are passing into the reactor hall, which is outside of the containment, so the releases are going directly to the environment. Therefore, a new type of SAMG is necessary.

TABLE I-12-4. RELEASE CATEGORIES.

| <i>Initially closed containment, full power and shut-down states</i> | |
|--|---|
| 1 | High pressure RPV rupture |
| 2 | By-pass |
| 3 | Early containment rupture |
| 4 | Early enhanced containment leakage |
| 5 | Late containment rupture |
| 6 | Late enhanced containment leakage |
| 7 | Early containment rupture with spray |
| 8 | Early enhanced containment leakage with spray |
| 9 | Late containment rupture with spray |
| 10 | Late enhanced containment leakage with spray |
| 11 | Intact containment |
| 12 | Intact containment with spray |
| 13 | Partial core damage |
| <i>Open containment, shut-down states</i> | |
| 14 | Loss-of fuel cooling (high decay heat) |
| 15 | Loss-of fuel cooling (low decay heat) |
| <i>Open containment, spent fuel pool accidents</i> | |
| 16 | Loss of cooling |
| 17 | Loss of coolant |

For the spent fuel storage pool the following initial events were considered [45]:

- Loss of heat removal (pump failure);
- Loss of coolant (pipe break).

Releases in these cases are also leading from the reactor hall to the environment. Therefore, mitigative actions are practically impossible, and AM measures need to focus on preventive measures. For prevention these accidents the development of extended shutdown EOP and some hardware modifications (additional automatic closing valves) are needed.

I-12.5. Accident management strategies and their components

The target of the accident management is the overall capability of the plant to respond to and recover from a severe accident situation. This capability could be increased by hardware modifications and with a guide to use the available resources in an optimal way. One of the key elements of this accident management programme is the SAMG that are already developed for Paks NPP and linked with the earlier implemented Westinghouse type EOPs.

The approach is structured around the four safety objectives of prevention of core damage, prevention of reactor vessel failure, prevention of containment failure and limitation of fission product release. Because of the variety of processes there is a need for different interventions and the selection of the right intervention needs a strategic decision. The main demand is consistency: there needs to be a comprehensive connection between each of the four elements and it needs to be ensured that they do not disturb or impair the effect on each other.

The key elements of AM strategies are the following: (1) prevention of CD (2) prevention of RPV failure by IVMR, or, in the case it appears unfeasible (3) ex-vessel debris cooling, and (4) release and containment management.

(1) The ***prevention of the core damage*** would be accomplished by recovery from inadequate core cooling or from loss of heat sink situations. There are certain strategies to cope with the station black-out and assure long term water sources from the bubbler tower for loss of emergency recirculation events as well. All those actions need to be included in the EOP.

(2) An indispensable part of the ***in-vessel melt retention*** is the strategy for *depressurization* of the primary circuit. Since the depressurization has not only mitigative but also preventive aspects it needs to take place early enough and in a very reliable way, in accordance with the relevant EOP. In a few special cases the in vessel flooding can prevent the reactor pressure vessel failure but creditable in-vessel corium retention can only be accomplished by the active cavity flooding. It is already clear that such IVMR is potentially feasible but certain measures and modifications would be required.

(3) ***Ex-vessel debris cooling*** has no more challenges, as it is solved by successful early cavity flooding.

(4) ***The containment strategies*** have three interconnected aspects. The first priority is given to the management of *pre-existing containment leakages*, which will predetermine the amount of the early release and will inherently affect *the late containment overpressurization*. The influence of the possible mitigation techniques (i.e. containment spray) need to be taken into account when developing the *hydrogen mitigation*.

Accordingly, these aspects of the accident management programme for Paks NPP has been developed and approved by the regulatory body in 2005. This programme prescribed a set of ***plant modifications*** as follows.

Primary circuit depressurization is a necessary measure to prevent high-pressure core meltdown sequences and to reduce the risk for induced steam generator tube rupture through the circulation of hot gases. In case the in-vessel retention of corium strategy is selected, a low primary system pressure is also a definite requirement. The recently implemented depressurization capability would reduce the pressure sufficiently as it was designed for potential releases of steam, two-phase mixture and water. In order to ensure sufficient opening reliability an independent so-called SAM power supply of the valves has to be installed.

The PSA results indicate that the risk of large releases dominated by *containment by-pass* sequences that caused leaks from primary to secondary side (PRISE) of the steam generators. It is an effective precaution against containment bypass to implement blow-down lines on the bottom of the steam generators that are directed to the containment.

Severe accident hydrogen is confirmed to be as a major threat to containment integrity. The rapid onset of flammable conditions in an unmitigated severe accident necessitates a means of control. With the help of level 2 PSA it was showed that implementation of about 30 large PARs would ensure that the containment would not experience high pressures loads in all those sequences that dominate the overall risk.

Both, *in-vessel melt retention* or ex-vessel debris cooling can only be accomplished by the active cavity flooding. It is already clear that IVMR is potentially feasible, but the potential for coolability of corium or core debris on the concrete basemat is still under investigation. There are double hermetic steel doors with rubber sealing in the sidewall of reactor cavity, which is a part of hermetic boundary. In case of ex-vessel cooling, the thermal protection of those doors against temperature loads is also to be solved. In order to avoid a number of specific loading mechanisms caused by the eventual melt ejection into the reactor cavity the necessary plant modifications needed for corium localization and stabilization inside RPV gets higher priority in the SAM program.

An IVMR concept with simple ECVR loop based only on minor but hard modifications of existing plant technology was proposed for Paks NPP [46][47]. Solution is shown in Fig. I-12-3. The analyses supported the assumption that the proposed solution is effective in preserving RPV integrity in the case of a severe accident. Efficiency of the ERVC loop in Paks-specific geometry was proven experimentally by AEKI on CERES test facility, in Budapest (see Fig. I-12-3). Scaling ratio for the reactor vessel surface is 1:40 (1/40 slice of the RPV) and for the elevations 1:1.

Filtered venting or long term containment cooling is used to prevent late containment failure. Both of them are effective precaution against late containment overpressure.

SAMG development needs to be based on the implemented plant modifications and measures. Important standpoint also, that SAMG need to be linked with the already implemented Westinghouse type EOPs. Therefore, SAMG development was done also in cooperation with Westinghouse Co.

Preventive measures for open reactor and spent fuel storage pool have also high priority because of large directly releases to the environment. They have two important aspects: one of them is the extension of Westinghouse type EOPs for shutdown mode. Another one is coming from Level 1 PSA for storage pool: the reinforcement of the SFP cooling system (installation of fast closing valves).

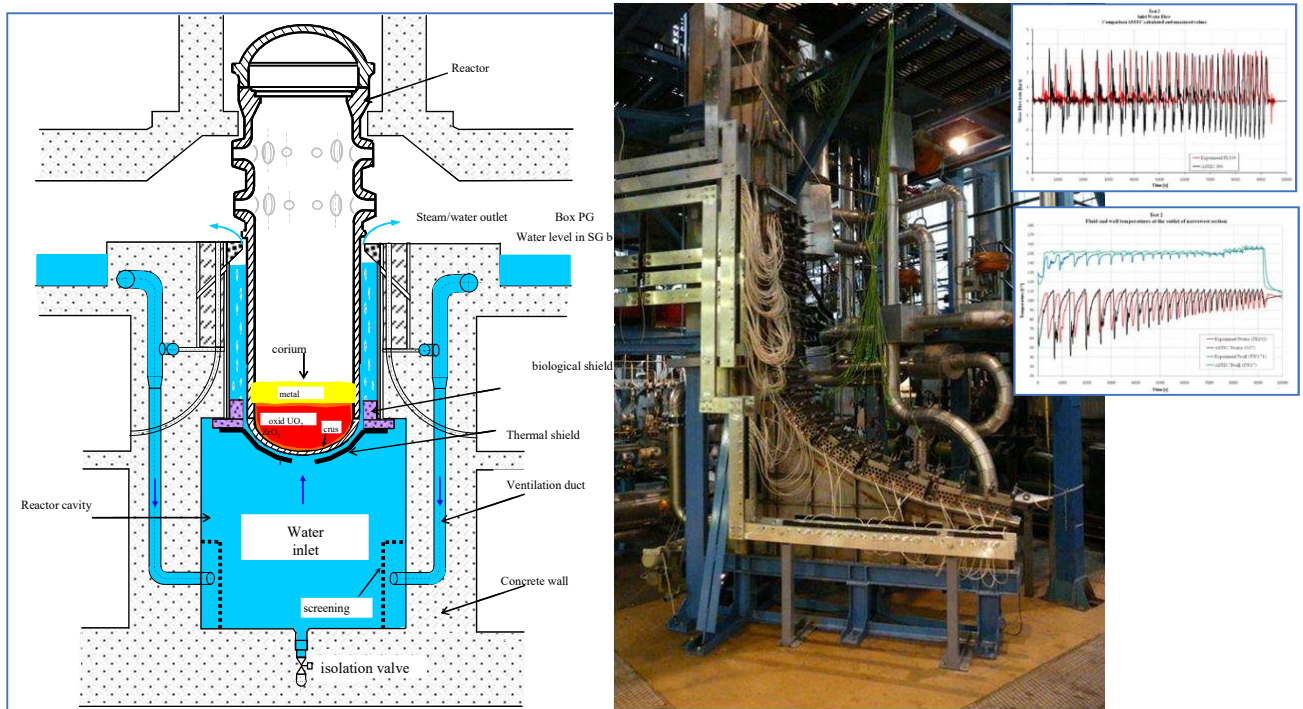


FIG. I-12-3. IVMR strategy and CERES test facility in Budapest

I-12.6. Plant modifications in 2-phase schedule

SAM measures and plant modifications in Paks NPP have been selected by two priorities:

- 1st priority measures: They are taken anyway, independently of the lifetime extension of the units. They are essential plant modifications, procedure development and organizational arrangements. These measures were scheduled up to 2012, which was the date for receiving the lifetime extension license for Unit 1.
- 2nd priority measures: They would be taken only in case of life time extension has been permitted by authority. These measures were scheduled after 2012.

These SAM measures can be selected also by different safety objectives/goals are listed below.

1st priority measures selected by safety goal, implemented on units in 2011-2014 (see Figs I-12-5 and I-12-6):

- Prevention of the core damage:
 - Extension of EOPs for shutdown mode and storage pool accidents.
 - Implementation of SA mobile diesels (4x100 kW) for autonomous electrical supply for SAM equipment: e.g. PRZ valves for successful primary depressurization, drainage valve of bubble condenser trays, new inlet valves for cavity flooding, SAM instrumentation.
 - Implementation of new PRISE strategy: bleed from ruptured SG to the containment before it filled up to prevent the SG safety valve stuck open.
 - Reinforcement of storage pool cooling system: installation of new automatic by water level fast closing valves to prevent flooding of both cooling pumps in the room A242.
 - Arrangement of duties for the other, non-damaged units.

- Prevention of RPV failure and early containment failure:
 - Development of SAMGs, all strategies validated by MAAP4 calculations.
 - Establishment of Technical Support Centre for using SAMG.
 - Installation of high capacity PARs to solve hydrogen issue: required capacity and distribution calculated by MAAP4 and GASFLOW 3D codes (NUBIKI).
 - Design and installation of cavity flooding flow path.
 - Installation of new independent severe accident measurement system: measurement of primary pressure and 12 core exit temperature up to 1200 °C, measurement of containment pressure, temperature, 8 O₂ and 8 H₂ concentration, measurement of water level in the containment, reactor cavity and SFP, high capacity dosimeters inside and outside of the containment.

2nd priority measures selected by safety goal (under design now):

- Prevention of late containment failure:
 - Increase reliability and protection of the spray system from common cause failures.
 - Prevention of late over-pressurization by filtered venting or long term containment cooling.

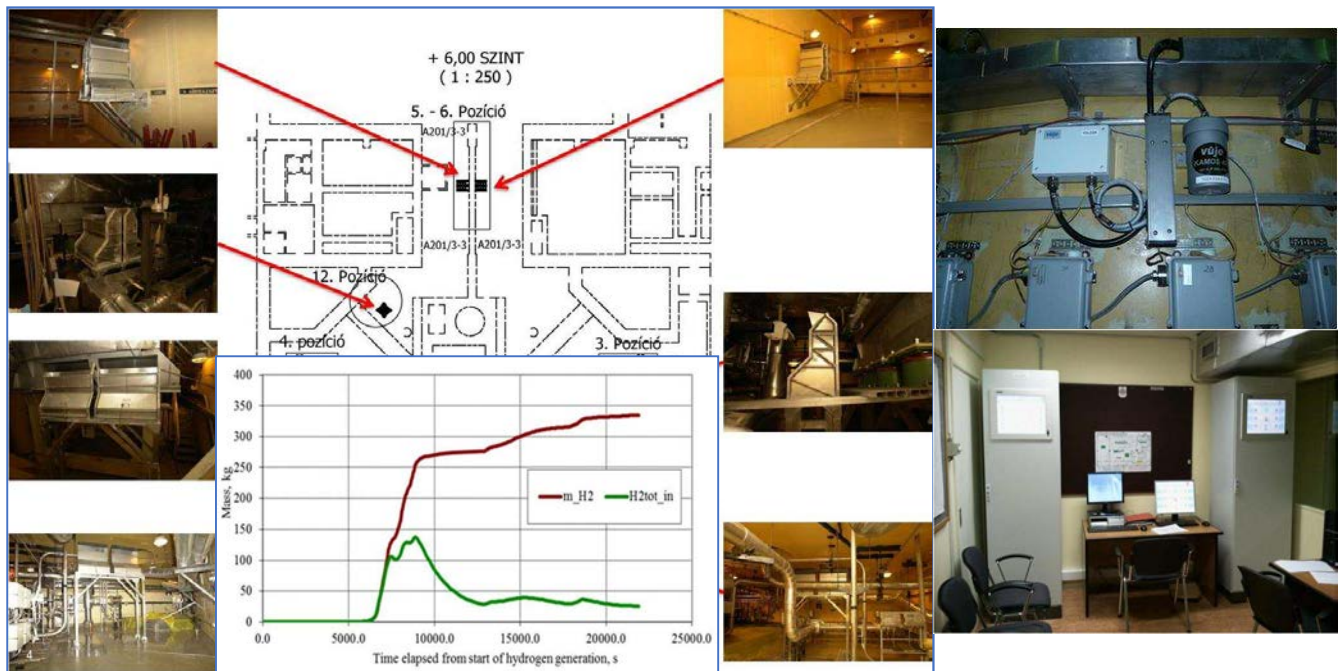


FIG. I-12-4. Hydrogen recombiners and H₂O₂ measurements in SG box, TSC room with SA measurement display.

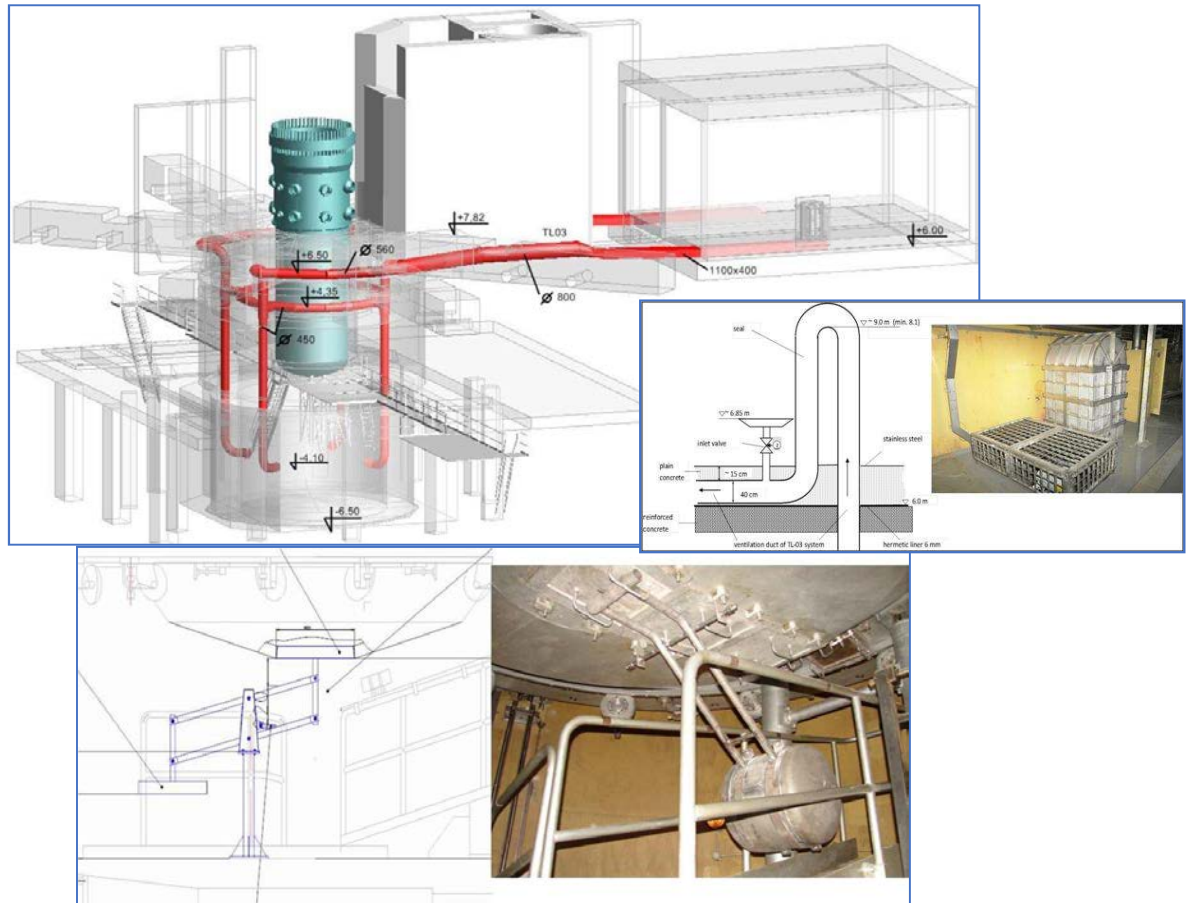


FIG. I-12-5. IVMR modifications - layout of ventilation ducts with drain valve and siphon.

Impact of implemented hardware and software components for SAM were analysed in the updated Level 2 PSA studies. Total frequency of large radioactive releases (LRF) in case of technological origin, internal fire or flooding events at nominal power now is $2,58 \cdot 10^{-7}$ 1/y, which is about 20% of previous value (reduction mainly in HPVF, early containment failure and by-pass cases).

Large early release frequency (LERF) now is $2,46 \cdot 10^{-7}$ 1/y, where “large” means that Cs release higher than 10^3 TBq and early means 24 h after TSC has been called (EOP, E-0, step 5).

I-12.7. Post Fukushima National Action Plan

Technical modifications in frame of Post Fukushima National Action Plan were decided after stress test initiated by Fukushima Daiichi NPP accident to prevent multiply unit severe accidents and to prevent late containment failure as follows:

- Different possibilities to connect service water line with fire water pumps and other water sources.
- Mobil diesel power supply of the coastal water well station.
- Installation of demineralized water tanks.
- Development of protection against condenser cooling water pipe rupture.
- Cross connection (longitudinal) between 6 kV reserve bus bars: possible connection of 6 kV safety supply between units.

- Autonomous water and boron acid supply to the containment and SFP.
- New SA diesels enable to use ECCS/SA pumps (2x2 MW).
- Long term containment cooling (SA spray + outside air cooling system: see Fig. I-25).

Most of these modifications have been already implemented except last two. Latest date for installation is the end of 2022.

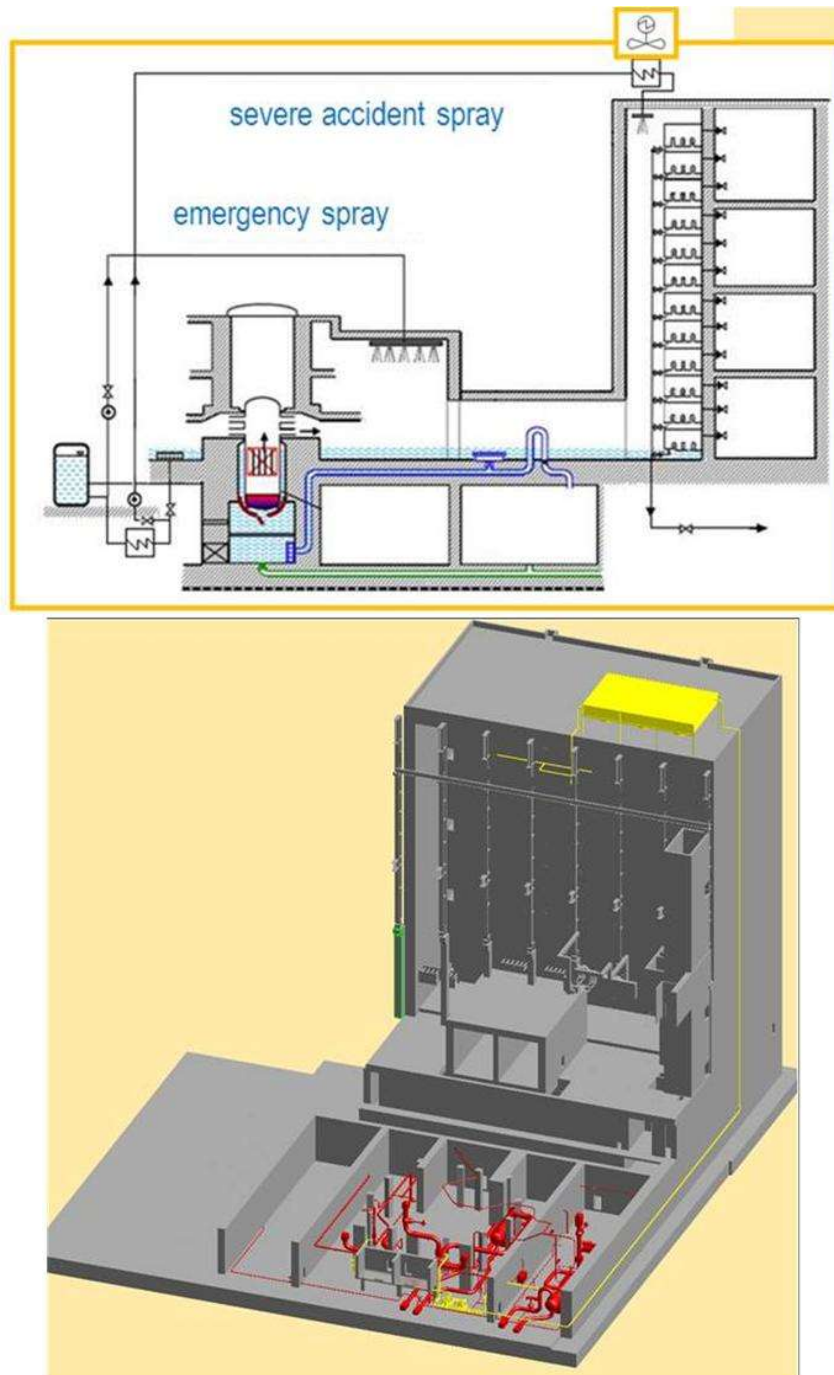


FIG. I-12-6. Planned long term containment cooling system

Abbreviations

| | |
|-------|---|
| APET | Accident Progression Event Tree |
| AM | Accident Management |
| CD | Core Damage |
| CERES | Cooling Effectiveness on Reactor External Surface |
| ECCS | Emergency Core Cooling System |
| EOP | Emergency Operating Procedure |
| FSAR | Final Safety Analysis Report |
| IVMR | In-Vessel Melt Retention |
| MCP | Main Coolant Pump |
| MLIV | Main Loop Isolation Valve |
| MCCI | Molten Corium Concrete Interaction |
| NPP | Nuclear Power Plant |
| PAR | Passive Autocatalytic Recombiner |
| PDS | Plant Damage State |
| PRISE | Primary to Secondary Leakage |
| PSA | Probabilistic Safety Assessment |
| PSR | Periodic Safety Review |
| RPV | Reactor Pressure Vessel |
| SAM | Severe Accident Management |
| SAMG | Severe Accident Management Guidance |
| SFP | Spent Fuel Pool |
| SG | Steam Generator |

I-13. JAPAN

I-13.1. Regulatory framework

I-13.1.1. Overview

In Japan there are a total of 58 reactors (24 PWRs (Pressurized Water Reactors), 33 BWRs (Boiling Water Reactors) and one FBR (Fast Breeder Reactor).

The following the accident of the TEPCO's Fukushima-Daiichi NPPs, nuclear regulation regime has been renewed, the Reactor Regulation Act and related legislation were revised and the Nuclear Regulation Authority (NRA) was established on September 2012. The new regulatory requirements for NPPs came into force on July, 2013. Licensees are required to obtain authorization of the NRA through the Conformity Review that assesses whether the reactor meets the regulatory requirements to resume operation. Figure I-13-1 shows the regulatory system.

The NRA has accepted applications of Conformity Review for 27 units of NPPs in 16 sites by the beginning of August, 2019. Commercial operation of Kyushu Electric Power Company's Sendai 1 and 2 have been resumed after the Conformity Review has been completed.

In the amendment of the Reactor Regulation Act on June, 2012, the operational period of NPPs is limited up to 40 years, in principle. The NRA has accepted applications for extension of the operational period for three units of two Nuclear Power Stations by the end of March, 2016. Among two units (Kansai Electric Power Company Takahama 1 and unit 2) are approved by the NRA in 20 June, 2016.

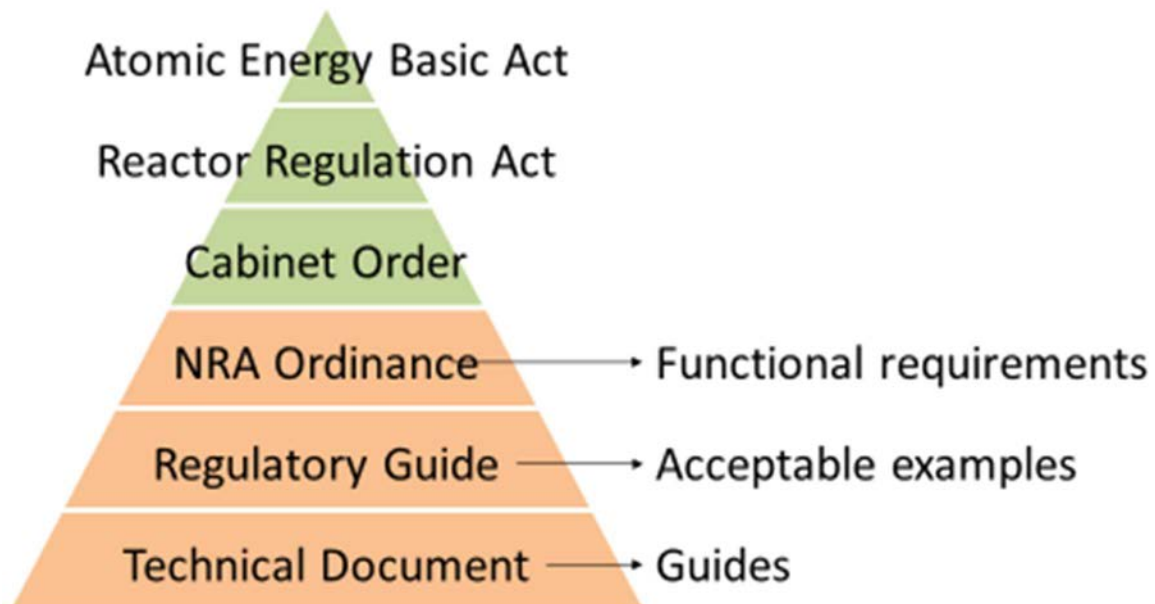


FIG. I-13-1. Regulatory system. (Courtesy of NSR)

I-13.2. Review of compliance with new regulatory requirements (See Table I-13-1)

To install and operate a new reactor, it is necessary to obtain Installation Permit and make a specific design; obtain approval on a construction plan and carry out construction work; and finally obtain approval on Operational Safety Programmes prior to reactor start-up. In addition, for NPPs on which authorization have been already granted, based on the back-fitting system introduced with the amendment of the Reactor Regulation Act in 2012, a review regarding conformity to the new regulatory requirements (Conformity Review) is to be conducted; approval of amendment of a Reactor Installation Permit already issued is to be granted; and approval on a Construction Plan and Operational Safety Programmes based on the approval of amendment is also to be obtained.

In the Conformity Review, a review for Amendments on a Reactor Installation Permit, a review on a Construction Plan, and a review on Operational Safety Programmes are conducted in parallel so as to review efficiently on both hardware and software in a unified manner.

The NRA implements a Conformity Review by holding an Examination Meeting where Commissioners participate. The Examination Meeting is made open basis to the public by allowing their attendances and the Internet broadcasting, along with material for the examination basically disclosed, thus maintain transparency of the review. In a process of the review, there are chances to hear the opinions of manufacturers and external experts depending on judgment by the Commissioners.

In addition to the Examination Meeting, hearings from a licensee as appropriate are occasionally held for purposes such as confirmation of facts related to matters included in an application. A summary of hearing proceeding is made open basis along with related material basically disclosed. A licensee who has opinions to a summary of hearing proceeding made open by the Secretariat of the NRA may present its opinions in a specified period of time.

TABLE I-13-1. STRUCTURE OF THE NEW REGULATORY REQUIREMENTS

| | | | |
|--|---|---|-------------|
| Suppression of radioactive materials dispersal | Beyond DEC | * Regulatory requirements were defined only in the DB before 1F Accident. | |
| Specialized Safety Facility | | | |
| Prevention of CV failure | DECs | | |
| Prevention of core damage | | | |
| Natural phenomena | The 2 nd and 3 rd layers of DiD | | Reinforced. |
| Fire, Internal flooding | | | |
| Reliability | | | |
| Reliability of power supply | | | |
| Ultimate heat sink | | | |
| Function of other SCCs | | | |
| Seismic/Tsunami resistance | | | |

I-13.3. Review of extension of operation period of NPPs

In accordance with the provisions of the Reactor Regulation Act, the period of allowable operation of NPPs is 40 years from the date they passed Pre-service Inspections, which means commercial operation. During this period, it is allowable to extend it one time, for a period of no more than 20 years, if approval is obtained from the NRA. The NRA approved for operation period extension approval from Kansai Electric Power Company for Takahama 1 and 2 on June 20, 2016 and Mihama 3 on November 26, 2015, Japan Atomic Power Company for Tokai-Daini on November 7, 2018.

I-13.4. Adaptation for IAEA safety standards and the Vienna Declaration on Nuclear Safety

In Japan, conventionally it has been required to take prevention measures on disaster caused by NPPs as regulatory requirements, and as a result of the amendment of the Reactor Regulation Act in 2012, measures for severe accidents were stipulated as regulatory requirements, resulting in strengthening of regulations.

Furthermore, this revision made it newly compulsory to conduct evaluation for safety improvement, report its results, and make them open to the public. Accordingly, periodical implementation of comprehensive and systematic safety evaluation and timely implementation of necessary improvement measures have come to be ensured along with applications of Periodic Facility Inspections, Periodic Safety Management Reviews, and Operational Safety Inspections.

In the 2012 the amendment of the Reactor Regulation Act, back-fitted system was introduced. In case of regulatory requirements are revised, licensees have obligation to meet their existing NPPs to new regulatory requirements. The NRA Ordinances where back fitting system is applied are the NRA Ordinance on the Reactor Installation Permit and the NRA Ordinance on the Technical Standards.

The back fitting system corresponds to measures taken to prevent operation of NPPs where safety is not ensured.

In the formulation process of regulatory requirements laid down by the NRA, the IAEA Safety Standards (e.g. GSR Part 4¹⁹, SSR-2/1²⁰, SSR-2/2²¹) and other international standards have been taken into account. As stated the above, Japan has already taken measures corresponding to elements of the Vienna Declaration on Nuclear Safety [1], [48].

¹⁹ INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4, IAEA, Vienna (2009)

²⁰ INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1, IAEA, Vienna (2012)

²¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Commissioning and Operation, IAEA Safety Standards Series No. SSR-2/2, IAEA, Vienna (2012)

The followings are overview of adaptation for each principle in the Vienna Declaration on Nuclear Safety [1], [48]:

- Principle 1 on “early or large releases be practically eliminated”:
 - It is stipulated in Article 1 of the amended Reactor Regulation Act to ensure public safety by preventing severe accidents that cause releases of abnormal levels of radioactive materials.
 - The new Regulatory Requirements require to take measures against DECs (Prevention of core damage and Prevention of CV failure) and to minimize the total amount of radioactive releases.
 - Also, it is required to evaluate effectiveness of the measured taken by using a combination of Probabilistic risk assessment (PRAs) and deterministic analyses.
 - It is stated in its review guide that “release amount of Cs-137 be less than 100 TBq.”

- Principle 2 on “comprehensive and systematic safety assessments”:
 - It is stipulated in the amended Reactor Regulation Act that licensees are responsible for improving safety taking into account state-of-the-art knowledge.
 - Licensees are also required to conduct the “Periodic Assessment of Safety Improvement” and notify the NRA of the results, and make them public.
 - The Operation Guide for the Assessment refers to GSR Part 4 (Rev.1) [49] and “14 Safety Factors” in SSG-25 [4] (PSR).

- Principle 2 on “timely implementation of safety improvements”:
 - “Backfitting” is newly introduced and New Regulatory Requirements are being backfitted.
 - Based on the lessons learned from the 1F accidents, the NRA has strengthened its own process to feedback the operating experience, state-of-the-art knowledgethrough an in-house committee “Technical Information Committee”.
 - The New Regulatory Requirements have been amended on the following additional events/phenomena as back fitted;
 - Single-phase open circuit
 - Toxic gas releases
 - HEAF: High Energy Arcing Faults

- Principle 3 on “... take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the RMs of the CNS”:
 - It is stipulated in the amended Atomic Energy Basic Act taking into account established international standards such as IAEA safety standards.....
 - Amendments of Reactor Regulation Act is in progress in response to the recommendations from the IAEA IRRS mission.
 - The NRA has participated in the activities of CSS and its five Committees, and is actively contributing to development and use of the IAEA Safety Standards [50].
 - The NRA has also contributed to the CNS Review Meetings.
 - It becomes more and more important to participate in and learn from the activities mentioned the above.

I-13.5. Compliance with the new regulatory requirement

In response to entering into force of the regulatory requirements in July 2013, the licensees have taken measures based on lessons learned from the Fukushima-Daiichi NPP accident in order to conform to the requirements. For example, they have taken necessary measures to improve fragility of protection against tsunamis, including installation of coast levees, installing watertight doors to important areas, enhancement of resistance to pressure and the waterproof property of outside walls of buildings. As measures for the case of water injection means at the time of station black out, they have arranged alternative electric power sources, such as air-cooled gas turbine generator vehicles, to high ground, increased the number of batteries, and constructed water reservoirs.

In addition, as measures to mitigate consequences of core damage, they have taken measures such as installation of top-vent facilities on the reactor building, top-head flange cooling lines to fill water into the top part of containment vessel (CV), filtered venting system. As for measures on the software side, emergency response organizations have been reorganized so that they can respond to accidents if severe accidents occur simultaneously in two or more units, so that a necessary number of the personnel for immediate response are ensured to enable first response on the emergency.

The new regulatory requirements state that preparation of necessary functions (of facilities or procedures) needs to be completed at the enforcement stage in July 2013 based on the lessons learned from the accidents. This condition is a further demand that preparation of backup facilities (a Specialized Safety Facility (see the link in the footnote²²) and a permanent DC power supply facility as the third power system) to further enhance reliability needs to be completed within five years from the date of approval of a construction plan related to measures to deal with severe accidents that are needed at the enforcement stage of the new regulatory requirements. The Specialized Safety Facility is a facility for measures against acts of terrorism attack such as intentional large-airplane crashes to the reactor building. It is required to have a facility necessary to prevent CV failure at a location about 100 meters or more from the reactor building, maybe at mountain side.

“Equipment to prevent CV failure” shall be equipped in Specialized Safety Facility. Practical requirements are;

- Depressurization function for Reactor Coolant Pressure Boundaries (e.g., equipment for reactor depressurization operation from emergency control room);
- Cooling function of molten core in the reactor (e.g., equipment for injecting low pressure water inside the reactor);
- function for cooling molten core that has fallen to the bottom of the CV (e.g., equipment for cooling water injection into the bottom of the CV);
- CV cooling/depressurization/radioactive material reduction function (e.g., equipment for injecting water into CV sprays);
- CV heat removal/depressurization function (e.g., venting system of CV excluding exhaust stacks);

²² The overview of “Specialized Safety Facility” is shown in “<https://www.nsr.go.jp/data/000067212.pdf> P.19”.

- Function of prevention of CV failure by hydrogen explosion (e.g., hydrogen concentration control equipment);
- Support function (e.g., equipment for power source, instrumentation, and communication);
- And installing the emergency control room to control above mentioned functions in emergency is also required.

I-13.6. Basic policy on defence in depth (see Table I-13-2)

In the past, before the NRA’s regulatory requirements were developed, Defence in Depth concept was stated in the Reactor Regulation Act and their Regulatory Guides issued by Nuclear Safety Commission, and requested as follows; for the 1st layer, ensure high reliability sufficient to meet importance of SSCs to prevent occurrence of abnormality; for the 2nd layer, necessary measures for early finding of abnormality and shut-down the reactor to prevent progression of abnormality; for the 3rd layer, core does not severely damaged and core is sufficiently cooled in case of Design Basis Accidents (DBAs) occurred to mitigate DBAs.

In the new regulatory requirements issued by the NRA, measures to eliminate common cause failures are significantly strengthened, based on the lessons learned from the Fukushima Daiichi NPP accident. In addition to the requirements mentioned the above, measures for preventing severe core damage in loss of function of equipment for addressing DBAs, and measures for preventing the CV failure in severe core damage, are required. Furthermore, the new regulatory requirements state measures for CV failure. The new regulatory requirement also states measures for preventing loss of large area of NPPs due to extreme natural disasters, intentional airplane crash or other terrorism.

It is required in the regulatory requirement that each layer of Defence in Depth independently performs its function effectively.

TABLE 1-13-2 STRATEGY OD DEFENCE IN DEPTH

| | | | | |
|---|------------------------------------|---|--|--|
| On-Site | Design Basis | 1. Prevention of deviations from NO | *. Regulatory requirements were defined only in the DB before 1F Accident. | |
| | | 2. Prevention of escalation to accidents | | |
| | | 3. Mitigaiton of DBAs | | |
| | Design Extension Conditions | 4-1. Prevention of severe core damage (DECs without significant fuel degradation) | | |
| | | 4-2. Prevention of CV failure (DECs with core melting) | | |
| | Beyond DEC | 5. Addressing loss of large areas | | |
| 5-1. Measures with mobiles | | 5-2. Specialized Safety Facility | | |
| 6. Suppression of radioactive materials dispersal | | | | |
| Off-site Emergency Preparedness and Response | | | | |

I-13.7. Requirements in each layer of defence in depth

I-13.7.1. Prevention of abnormality

For the purpose of prevention of abnormality, it is required that ensuring high reliability sufficient to meet importance of SSCs to prevent occurrence of abnormality, design with sufficient safety margin, as well as reactor has its core stability characteristics, preventing mis-handle, fail-safe design, and interlock function.

In the regulatory requirements, measures for seismic safety, tsunami safety, reliability of power sources, and fire protection are strengthened and introducing measures for internal flooding, volcano, tornado, forest fire are newly required.

I-13.7.2. Prevention of progression from normality

In order to detect deviation from normal operation (NO) and make it under control, measures to prevent anticipated operational occurrences (AOOs) in NPP, from progression of accidents such as preparing specific system and mechanism in the design, and establishing operation procedure to regain safety state of NPPs, are required.

I-13.7.3. Mitigation of DBAs

In case of progression of AOOs or postulated initiating events and cannot control them in the previous layer and allow to progress to DBAs, it is required that core is not severely damaged and be able to maintain sufficient cooling by the Engineered Safety System and core integrity.

I-13.7.4. Prevention of severe core damage in Design Extension Conditions without significant fuel degradation

Licensees are required to demonstrate effectiveness of measures to prevent severe core damage in Design Extension Conditions (DECs) without significant fuel degradation.

DECs without significant fuel degradations is identified as a “postulated accident sequence groups.” The NRA Ordinance, taking research results into account, stipulates accident sequence groups that cover the most of accident sequences with core melting as “designated accident sequence groups”, as shown in Table I–13-2.

TABLE I–13-3. DESIGNATED ACCIDENT SEQUENCE GROUPS

| BWR | PWR |
|---|---|
| Loss of high-pressure and low pressure water injection function | Loss of heat removal function of secondary cooling system |
| Loss of high-pressure water injection and depressurization function | Loss of AC power |
| Loss of all AC power | Loss of auxiliary component cooling function |
| Loss of decay heat removal function | Loss of CV heat removal function |
| Loss of reactor shutdown function | Loss of Reactor shutdown function |
| Loss of water injection during LOCA | Loss of ECCS water injection function |
| CV bypass (Interface system LOCA) | Loss of ECCS recirculation function |
| | CV bypass (Interface system LOCA, steam generator tube rapture) |

Considering the difference of each plant, internal events are evaluated by applying PRA and external events are evaluated by PRA or other applicable methods. As a result, in case that the accident sequence group that has significant frequency or impact is identified although it is not included in the “Designated Accident Sequence Group,” it is required to add into “Postulated Accident Sequence Group”.

In the next step, important accident sequences are identified in each of the Postulated Accident Sequence Groups in terms of the amount of equipment that loses its function simultaneously, spare time, level of equipment capacity necessary to prevent core damage, and whether represent the characteristic of the accident sequence group in question. Evaluation of effectiveness are performed to demonstrate that equipment for severe accident meets the evaluation requirements (e.g., maximum temperature of fuel cladding tube is below 1,200 degree Celsius) by simulation, and sufficiency of plan regarding necessary human-power and fuel from the view point of whether equipment required as severe accident measures can prevent core melting in the important accident sequence.

Equipment required to address DEC's have to meet following regulatory requirements; the equipment do not lose its function simultaneously with safety function of equipment to address DBAs caused by common cause failures; the equipment have anti-seismic function. In addition to these requirements, high reliability is required to permanently installed equipment. For non-permanent equipment, meeting general industrial standards and multiple deployment of equipment (coolant injection, power source) are required.

I-13.8. Prevention of CV failure in DEC's with core melting

Licenses are required to demonstrate effectiveness of measures to prevent CV failure in the case of the Design Extension Condition with core melting.

DEC's with core melting is identified as "CV failure mode". The NRA Ordinance, taking research results into account, stipulates "Designated CV failure mode" as the typical CV failure mode.

Practical items stipulated as CV failure mode make certain of assuming are:

- Static loads by internal pressure/temperature (damage by CV over-pressurization/over-heating);
- High pressure melt ejection/direct heating of CV atmosphere; Ex-vessel fuel-coolant interaction;
- Hydrogen explosion;
- Direct contact with CV (shell attack);
- Melted core and concrete interactions (MCCI).

Considering the difference of each plant, internal events are evaluated by PRA and external events are evaluated by PRA or other applicable means to identify CV failure mode based on the characteristics of each plant. As a result, in the CV failure mode that has significant frequency of occurrence or impact is identified although it is not included in the "Designated CV failure mode", it is required to add into "Postulated CV failure mode".

In the first step, for every Postulated CV failure mode, severe accident sequence from the point of load against CV is identified as evaluated accident sequence from among CV failure sequences based on the results of PRA. Subsequently, evaluation of effectiveness is conducted to demonstrate that equipment against severe accident meets the criteria such as maximum operating pressure or limiting pressure, provided by simulation code analysis, and sufficiency of plan regarding necessary man-power and fuel from the view point of whether equipment required as severe accident measures can prevent CV damage.

Equipment required to address DEC with core melting have to meet following regulatory requirements; the equipment perform its function in accident conditions; redundancy, diversity, and dispersed in the different locations have to be ensured in case that equipment to address DBA have no similar function, e.g., water injection to CV bottom, hydrogen explosion; equipment have anti-seismic function. In addition to these requirements, high reliability is required to permanently installed equipment. For non-permanent equipment, meeting general industrial standards and multiple deployment of equipment (coolant injection, power source) are required.

I-13.9. Measures to suppress dispersion of radioactive material

The NRA Ordinance requires measures to prevent severe core damage and CV failure, as measures to address DEC. The NRA Ordinance requires equipment to suppress dispersion of radioactive material to outside of site based on appropriate analysis of dispersion mode from the point of preventing abnormal level of release of radioactive material into the environment, even if assuming severe core damage and CV failure occur beyond DEC. For example, water cannon is required to suppress dispersion of radioactive material in aerosol form leaking from the reactor building.

I-13.10. Measures to address loss of large area

Loss of large area is the large-scale destruction of NPPs caused by extreme natural disaster, intentional airplane crash or other terrorism. Extreme natural disaster means the natural disaster beyond design basis in the NRA Ordinance on Standards for the Location.

In the NRA Ordinance, measures with non-permanent equipment and Specialized Safety Facility, as installed facility, are required.

I-13.11. Measures with non-permanent equipment

Loss of large area by airplane crash leads to severe destruction of certain area of NPPs. In this case, it is important to take measures not by based on assumption of certain accident sequence but to avoid losing all measures for decreasing release of radioactive material, based on the destruction occurred.

In case of natural disaster extremely beyond design basis or large airplane crash, it is required non-permanent equipment is not become unavailable simultaneously by taking measures of dispersed deployment.

In practical, access routes have to be repaired by heavy machinery stored in dispersed locations when access routes such as road, are destroyed by natural disaster beyond design basis,; ensuring to prepare connection points in the opposite side of damaged side to be able to connect non-permanent equipment such as feed water pumps or power sources in case of connection points are lost by airplane crash into one side of reactor building, are required.

I-13.12. Measures with Specialized Safety Facility

“Specialized Safety Facility” shall be equipped with adequate measures for preventing the loss of necessary function due to the intentional crashing of a large airplane into the reactor building.

Practical requirements are:

- Ensure enough distance (e.g., 100 m or more) between Specialized Safety Facility and reactor building to prevent simultaneous failure of both facilities; or
- Specialized Safety Facilities have to be in a robust structure that can withstand an intentional airplane crash, or else be inside facilities that have an equivalent or more effective level of protection.

Licensees shall demonstrate that evaluated equipment has to keep its necessary function by performing structural evaluation of building and functional evaluation of equipment at the event of airplane crash, with specifying characterization of airplane and identifying exact point of crash.

“Equipment to prevent CV failure” shall be equipped in Specialized Safety Facility. Practical requirements are:

- Depressurization function for Reactor Coolant Pressure Boundaries (e.g., equipment for reactor depressurization operation from emergency control room);
- Cooling function of molten core in the reactor (e.g., equipment for injecting low pressure water inside the reactor);
- function for cooling molten core that has fallen to the bottom of the CV (e.g., equipment for cooling water injection into the bottom of the CV);
- CV cooling/depressurization/radioactive material reduction function (e.g., equipment for injecting water into CV sprays);
- CV heat removal/depressurization function (e.g., venting system of CV excluding exhaust stacks);
- Function of prevention of CV failure by hydrogen explosion (e.g., hydrogen concentration control equipment);
- Support function (e.g., equipment for power source, instrumentation, and communication)
- Installing the emergency control room to control above mentioned functions in emergency is also required.

I-13.13. Safety evaluation of reactor installation

I-13.13.1. Safety evaluation in the reactor installation permit phase

In the application for a reactor installation permit, the applicant is required to present the conditions used to evaluate the necessary equipment to manage accidents due to abnormal transient events, DBAs and severe accidents and the extent and effect of the anticipated accidents, and explain that the safety of the NPPs is ensured, based on the results of the evaluation.

I-13.13.2. Evaluation of safety improvement

A new safety improvement evaluation programme was introduced into the 2012 revised Reactor Regulation Act. The programme requires the licensee to conduct an evaluation of the safety of the NPPs by themselves no later than six months after the day on which periodic inspection of the NPP ends (normally, it will be ended in one month after restart). After the evaluation, the licensee shall report the results of the evaluation to the NRA without delay and disclose the results.

The NRA states the guidelines for “Periodic Safety Assessment of Continuous Improvement of Commercial Power Reactor” in 2013, and revised the guidelines taking into account the 14 safety factors in Table 3 of IAEA Safety Standards Series NO. SSG-25 [4]. (See the table I–13-3.)

For the report of evaluation of safety improvement, the investigation, analysis or assessment conducted by the licensee is determined not to comply with the method set forth in the NRA Ordinance Concerning the installation and operation of Commercial NPPs, the NRA can order the licensee to change to improve the method of investigation, analysis or assessment.

In addition, evaluate the safety margin to determine to what extent a target NPP can withstand an beyond the design basis events without causing severe damage to fuel assemblies, loss of confinement function, or abnormal release of radioactive substances.

In order to take measures to prevent severe damage to fuel assemblies, containment function loss, or abnormal release of radioactive substances, present their effectiveness, identify cliff-edge effects (e.g., damage to or flooding of components or the like caused by an earthquake or a tsunami beyond the design basis leading to a series of function losses of items important to safety and subsequent fuel damage), and clarify the potential vulnerability of equipment from the viewpoints of “defence in depth.”

In this way, make a comprehensive assessment of the robustness of the NPP withstand beyond the design basis external events.

TABLE I–13-3. PERIODIC ASSESSMENT OF SAFETY IMPROVEMENT: CORRESPONDENCE TO SAFETY FACTORS IN SSG-25 [4]

| Factor# | Safety Factor | Frequency |
|----------------|---|--------------------|
| 5 | Deterministic safety analysis | Every 5 years |
| 6 | Probabilistic safety assessment | |
| 7 | Hazard analysis | |
| 1 | Plant design | Mid-and-long term: |
| 2 | Actual condition of SSCs important to safety | Every 10 years |
| 3 | Equipment qualification | |
| 4 | Ageing | |
| 8 | Safety performance | |
| 9 | Use of experience from other plants / research findings | |
| 10 | Organization, management system / safety culture | |
| 11 | Procedures | |
| 12 | Human factors | |
| 13 | Emergency planning | |
| 14 | Radiological impact on the environment | |

I-13.14. Contents of the report for safety improvement

1. Scope with confirmed legal conformity to safety regulations
 - (1) Overview of the NPP
 - (2) Site characteristics
 - (3) Structures, systems, and components
 - (4) Management system and managed matters for ensuring operational safety
 - (5) Safety evaluation results for checking legal conformity
2. Voluntary measures to enhance safety
 - (1) Policy for continuous efforts to enhance safety
 - (2) Investigation
 - (a) Situation of operational safety activities
 - (b) Latest scientific and technical knowledge in domestic and overseas
 - (c) Study to gain a detailed understanding of the current status of the NPP (plant walkdown)
 - (3) Safety improvement plan
 - (4) Contents of additional measures
 - (5) Results of external evaluation (if any)
3. Investigation and analysis of voluntary measures for enhancing safety
 - (1) Evaluation of activities involved in safety enhancement
 - (2) Probabilistic risk assessment (PRA) conducted with regard to internal events
 - (3) Probabilistic risk assessment (PRA) conducted with regard to external events
 - (4) Safety margin evaluation enhancement
4. Comprehensive evaluation

I-13.15. Confirmation process by the NRA

The NRA verifies that the investigations described in a written notification pursuant to the Reactor Regulation Act were conducted by employing methods as specified in the Commercial Reactors Ordinance.

I-13.16. Identification of safety improvements

Drivers for the enhancement process

After the Fukushima Daiichi NPP accident, utilities are improving measures for severe accident. Main drivers for improvements are as follows:

- The New Regulatory Requirements after the Fukushima Daiichi NPP accident:
 - Enhanced design basis against earthquakes and tsunami (revised);
 - Enhanced countermeasures to prevent severe accidents (countermeasures against fire, power supply reliability were revised and countermeasures against natural phenomena such as volcanoes, tornadoes and forest fires and internal flooding were added);
 - Countermeasures to mitigate severe accidents and terrorism.

- Operational experience at other utilities, and self-assessment were implemented, as part of Quality Assurance activities of each utility.
- “Evaluation of safety improvement” including safety review of various aspects such as safety activities in general, new scientific knowledge and technical knowledge, PRA, stress test which are expansion of conventional PSR and utilization of RIDM (Risk Informed Decision Making). Each utility is promoting introduction of it.

Selection process of safety improvements

Under the new regulation, specific severe accident countermeasures such as alternative water injection systems are required. When these systems are installed, utilities shall confirm that the systems function effectively against severe accident using probabilistic risk assessment and deterministic safety analysis, and report the result to NRA.

“Evaluation of safety improvement” is a framework for improving the safety and reliability of nuclear power plants voluntarily and continuously by utilities. In “Evaluation of safety improvement”, utilities periodically assess the nuclear facilities in as-is state, and from the insights, finds further safety improvement measures and prepares plans to implement those measures. In “Evaluation of safety improvement”, utilities conduct comprehensive evaluation with various aspects such as newer knowledge, PRA, stress test (safety margin assessment) etc. Through this activity, utilities will continually recognize risks of plants and reduce and eliminate those risks.

Outcomes identified of safety improvements

Based on the lessons learned from the Fukushima Daiichi NPP accident, utilities are fully committed to strictly adhering to the New Regulatory Requirements. Utilities are also promoting voluntary safety initiatives.

After the Fukushima Daiichi NPP accident, utilities have introduced countermeasures for severe accident (examples are shown in below) and been also working on maintenance and improvement of emergency response capabilities (include education/training program, administration, reviews of procedures, etc.), in order to prevent the occurrence, progress and expansion of accidents.

Additionally, as a party who experienced Fukushima Daiichi NPP accident and failed to prevent the accident, TEPCO is committed on clarifying the accident at Fukushima Daiichi NPP to contribute to enhancing nuclear power plant safety. TEPCO continues working to gain a thorough understanding of what happened through planned investigation of the site and simulation analyses. TEPCO would verify effectiveness of own safety measures for NPP whenever new findings are obtained. Findings through this effort are publicly available as progress report "Evaluation of the situation of core and containment vessels of Fukushima Daiichi Nuclear Power Plant Unit-1 to 3 and examination into unsolved issues in the accident progression".

Detailed design of safety improvements

New regulatory requirements were developed taking into account the lessons-learned from the Fukushima Daiichi NPP accident considering the harsh natural conditions unique to Japan. Under the new regulation, it is legally required to implement back-fitting and improvements required before restarting a reactor.

A new safety improvement evaluation programme was introduced into the 2012 revised Reactor Regulation Act. The programme is to require a licensee to conduct an evaluation of the safety of the power reactor facility by themselves no later than six months after the day on which periodic inspection of the facility ends. After the evaluation, the licensee has to report the results of the evaluation to the NRA without delay and disclose the results.

Here are examples of the measures implemented for plant restarted or plants aiming for restarting in such a scheme. Further measures are introduced on The Kansai Electric Power Company, Inc. website and Tokyo Electric Power Company Holdings, Inc. website.

- The Kansai Electric Power Company, Incorporated²³,
- Tokyo Electric Power Company Holdings, Incorporated²⁴

I-13.17. Countermeasures against tsunami and flooding (The Kansai Electric Power Co., Inc.)

I-13.17.1. Tide embankment at Mihama Power Station

A 11.5m high (above sea level) and approx. 100m long tide embankment was installed outside Wakasa Bay, which could be directly reached by a tsunami. In addition, in order to protect against flooding due to tsunami flowing into the inland sea, the Kansai Electric Power are installing a 6m high (above sea level; approx. 2.5m actual height) and 1.4km long tide embankment in Nyu Bay: see Fig. I-13-6.



*FIG. I-13-6-. Tide embankment at Mihama Power Station
(Courtesy of © The Kansai Electric Power Company, Incorporated).*

²³ For more information please visit < https://www.kepc.co.jp/english/energy/nuclear_power/

²⁴ For more information please visit < <https://www7.tepco.co.jp/ourbusiness/nuclear/kashiwazaki-kariwa/emergency-e.html>>

Watertight doors that protect important safety components

The doors of any building that contains important safety components, such as emergency diesel generators were replaced with watertight doors to prevent the building from being inundated in the event of a tsunami. The watertight door is at least 10cm thick (example of The Kansai Electric Power) and has a robust, waterproof structure: see Fig. I-13-7.



*FIG. I-13-7. Example of Watertight doors
(Courtesy of © The Kansai Electric Power Company, Incorporated).*

I-13.18. Countermeasures against natural hazards, such as volcanic activity, tornados and forest fires (The Kansai Electric Power)

In order to prevent the simultaneous loss of functionality of important safety components due to natural phenomena, protective measures have been taken by assuming significantly more severe natural disasters than conventional assumptions.

Protection against external fire

A 18m wide firebreak zone was created by cutting down trees in the periphery of the site to prevent the equipment from being damaged by a forest fire near the power station: see Fig. I-13-8.



*FIG. I-13-8. Image of firebreak zone
(Courtesy of © The Kansai Electric Power Company, Incorporated).*

Protection against flying objects from tornados

Tornado protection has been installed for the seawater pumps in case steel objects fly toward the pumps in a tornado with wind speeds of 100m/sec. (see Fig. I-13-9)

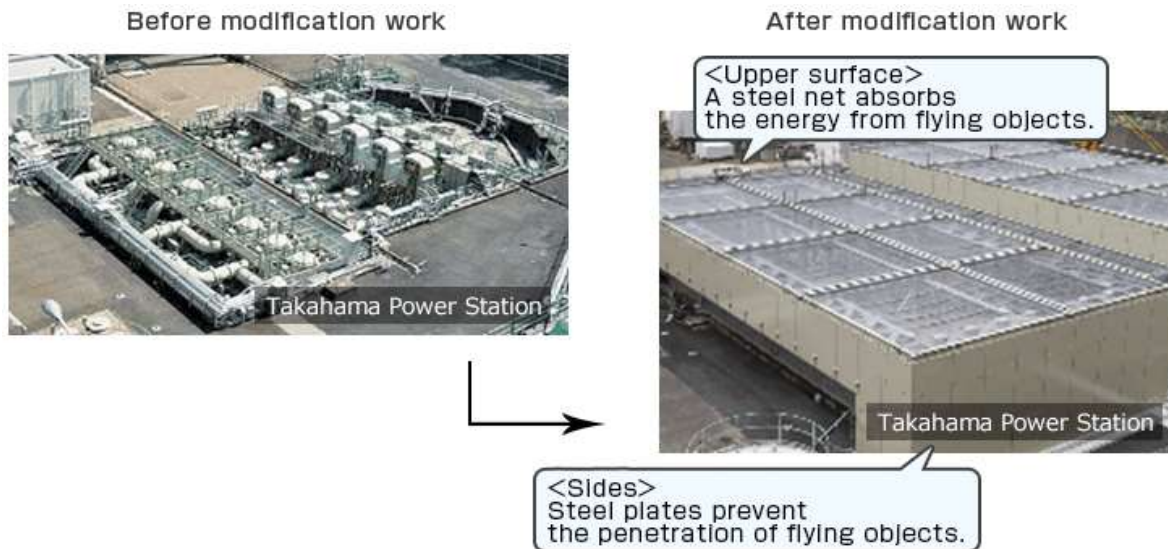


FIG. I-13-9. Example of tornado protection for the seawater pumps
(Courtesy of © The Kansai Electric Power Company, Incorporated).

I-13.19. Power supply reinforcement (Tokyo Electric Power Company Holdings, Inc. Kashiwazaki-Kariwa Nuclear Power Station (TEPCO KK NPS))

In order to promptly secure power to operate the critical equipment in the case of station blackout, large capacity air-cooled gas turbine generator cars are deployed on the high ground in addition to power supply cars. High-voltage switchboards, permanent cables and underground gas oil tanks (for the gas turbine generator cars) have been installed to allow for a swift power supply connection. See Fig. I-13-10.

- Air-cooled gas turbine generator car: Deployed
- Power supply cars: 14 deployed
- Other equipment/materials (such as connecting cable): Deployed

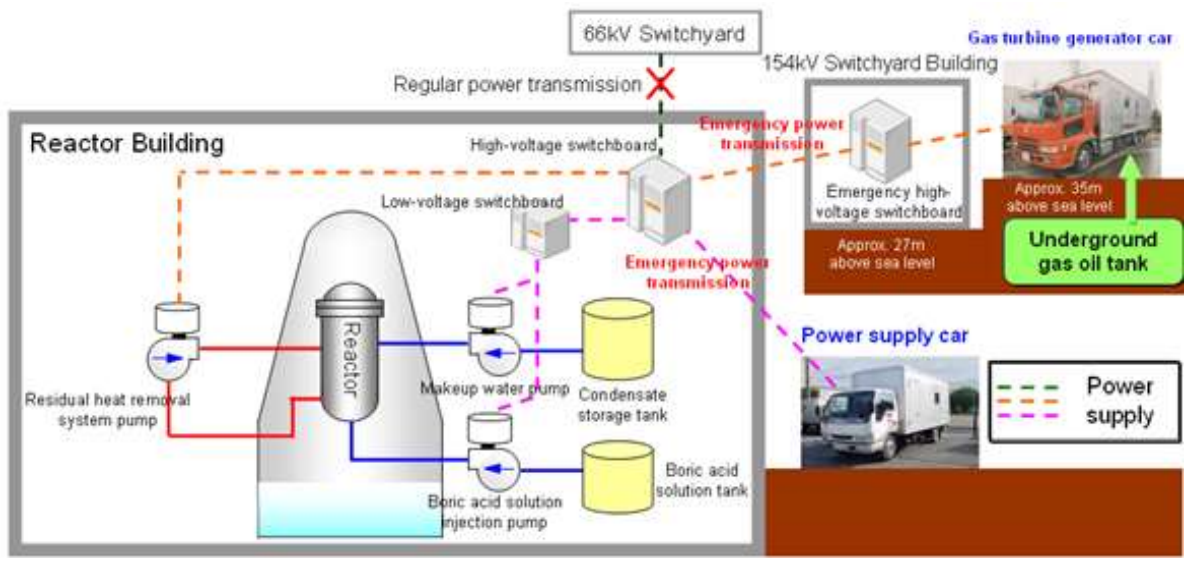


FIG. I-13-10. Power supply reinforcement for TEPCO KK NPS
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

I-13.20. Enhanced water injection system (TEPCO KK NPS)

Even in the case that the existing water injection system is disabled, sufficient amount of water can be secured for cooling the reactor with multiple alternative water injection functions. A freshwater reservoir (Capacity: approx. 20,000 tons) on the high ground at 45 m above sea level and a well to supply water to the reservoir was built to secure freshwater storage used for cooling: see Fig. I-13-11.

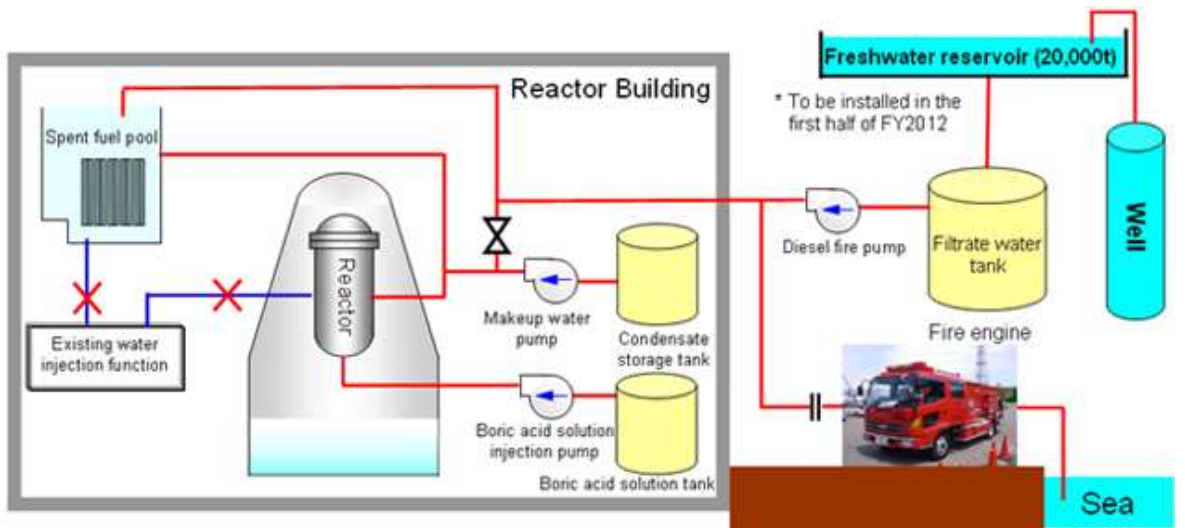


FIG. I-13-11. Alternative water injection for TEPCO KK NPS
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

I-13.21. Enhanced residual heat removal system (TEPCO KK NPS)

Considering the case that the equipment installed in the Seawater Heat Exchanger Building are flooded and disabled by tsunami, mobile alternative seawater heat exchanger facilities are deployed on the high ground to ensure the capability of cooling the reactor and the spent fuel pool: see Fig. I-13-12.

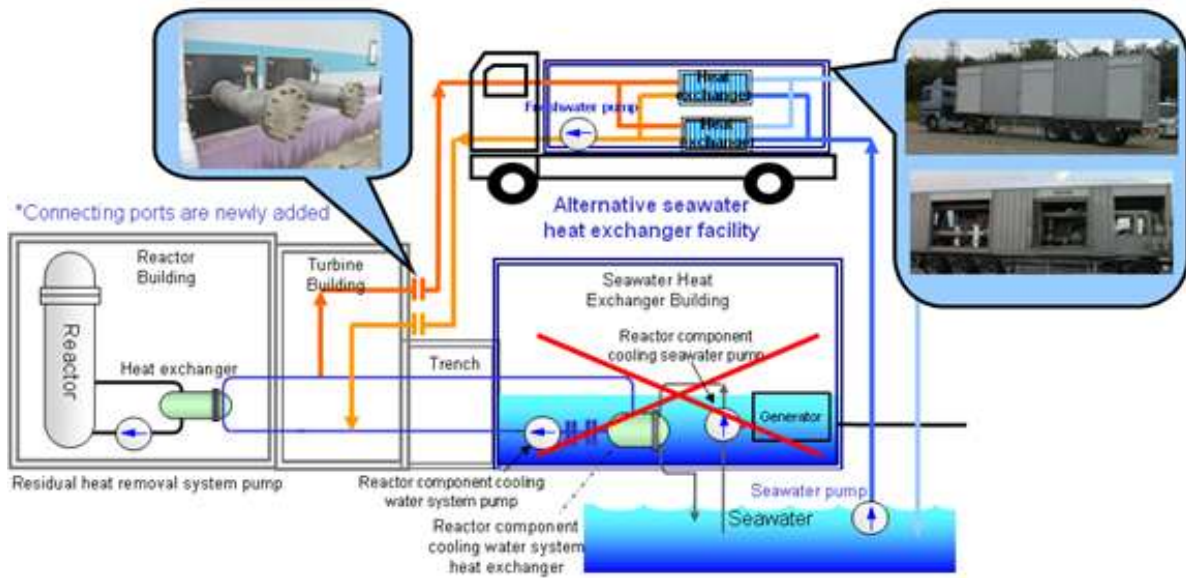


FIG. I-13-12 Mobile alternative residual heat removal system for TEPCO KK NPS (Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

I-13.22. Filtered Containment vent system

Filtered vent system is installed to reduce the release of radioactive substances when the core is damaged. (It will reduce the release of radioactive substances to ca. 1/1000.) PCV vent valves can be operated under SBO condition (batteries, gas cylinders, manual remote operate), and manually operated from the outside of the secondary containment (the measure against radiological protection): see Fig. I-13-13.

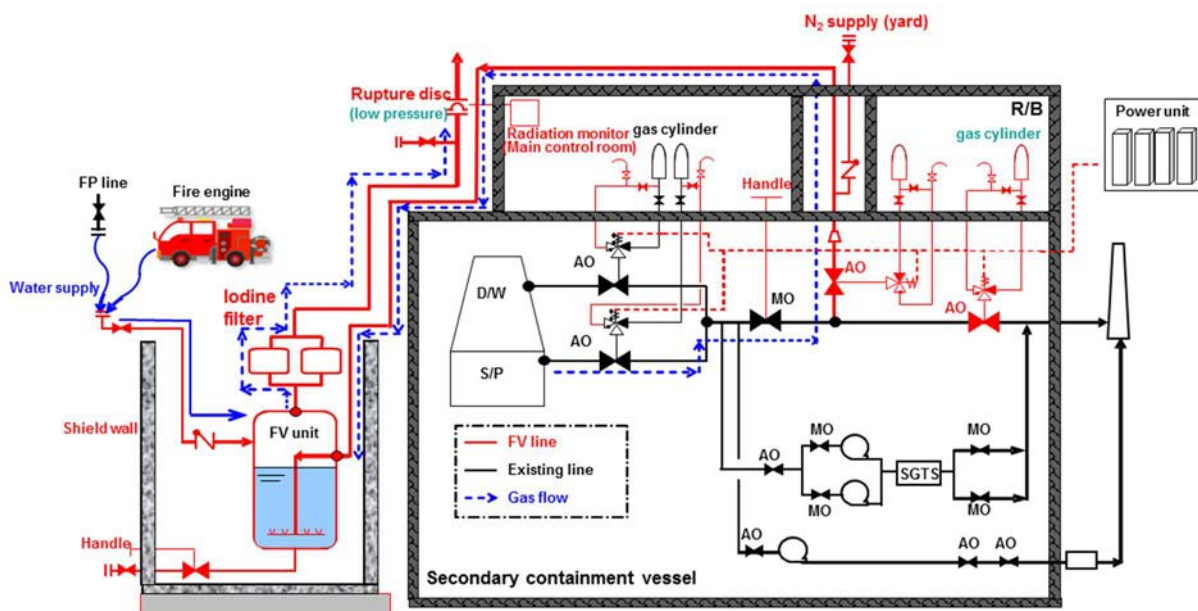


FIG. I-13-13. Filtered Containment vent system for TEPCO KK NPS (Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

Enhancement of the Emergency Response Capabilities and the other (TEPCO KK NPS)

A number of emergency drills are also performed in KK NPS. By doing so, emergency personnel strive to improve the reliability of response to accidents and are working on improving response capabilities in complex events such as accidents of multiple units. In addition to these activities, TEPCO has been executing “the Nuclear Safety Reform Plan” to improve its “safety awareness,” “technical capabilities” and “ability to dialogue” to raise the level of nuclear safety as an organization: see Fig. I-13-14.



FIG. I-13-14. Example of emergency drills
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

Utilities evaluate the effectiveness of the measures described above using PRA. An example of TEPCO KK NPS’s PRA results is shown in Fig. I-13-15.

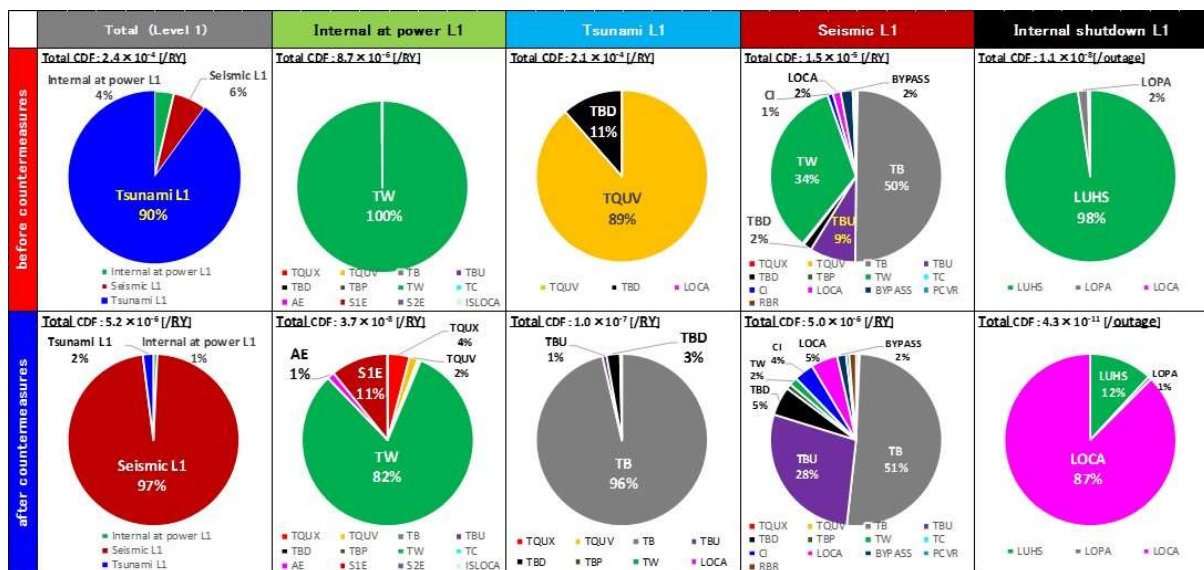


FIG. I-13-15. Example of PRA results of TEPCO KK NPS
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

These PRA results were evaluated in 2015. The PRA models used included simplified models due to the fact that some facilities designed for DEC were conceptual design state at the time. In addition, these PRA models had room for improvement in terms of "Scope and level of Detail", "Realism", "Conservative assumption in Seismic-PRA", and "HRA when using mobile equipment in DEC". Currently, these PRA models are being improved to develop more sophisticated PRA models with as-built modeling of safety enhancement measures. The results of sophisticated PRA will be reported in “evaluation of safety improvement”.

I-14. JAPAN – APPROACHES OF ACCIDENT ANALYSIS ON DESIGN EXTENSION CONDITIONS BASED ON THE CASE OF KASHIWAZAKI-KARIWA NPP

I-14.1. Selection of representative sequences of DEC's

In order to evaluate that safety measures for DEC's are effective, it is necessary to identify the accident sequence. However, there are numerous accident sequences of severe accidents with multiple failures of design basis equipment (DBE), it is not realistic approach to evaluate for thus numerous sequences. Therefore, we referred to the ET method used in PRA for selection of representative sequences of DEC. Since it aims to evaluate the effectiveness of additional safety measures, this ET(Event Tree) is without additional safety measures. In addition, it is necessary to consider for natural phenomena not applying the PRA method other than internal events, earthquakes, and tsunami. It is important to confirm that extracted accident sequences are representative from a qualitative point of view. see Fig. I-14-1 and Fig. I-14-2.

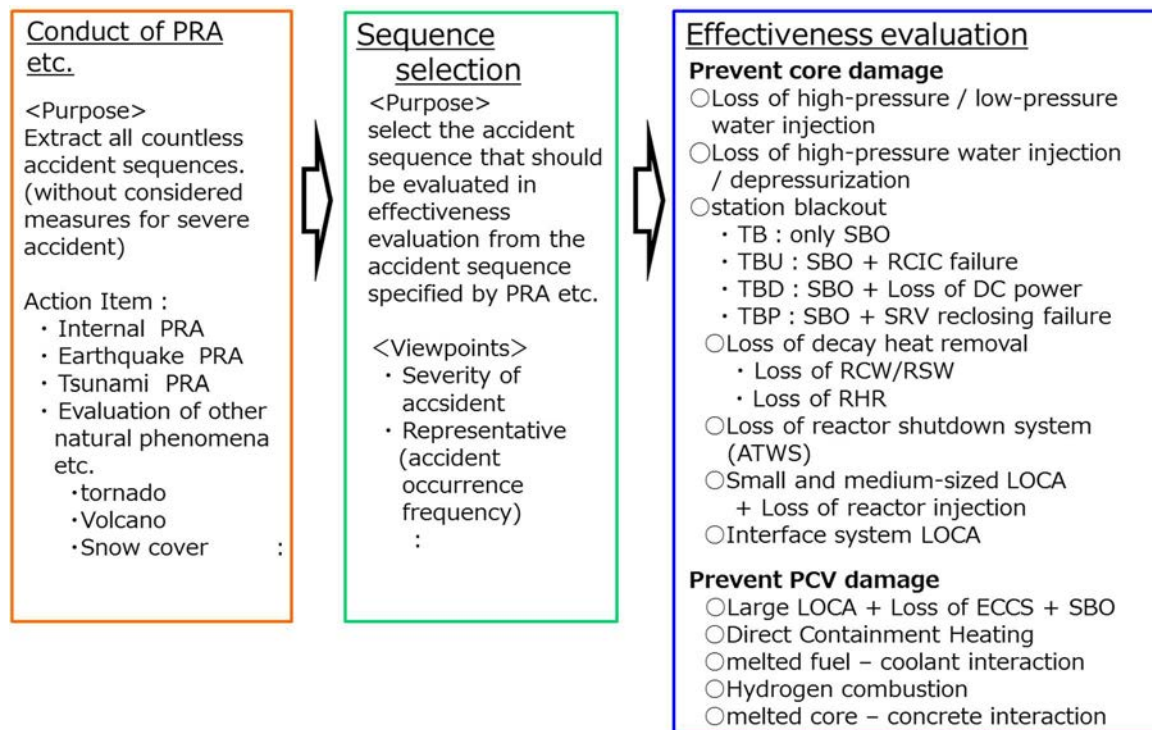


FIG. I-14-1. Example of selecting process for BDBA.
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

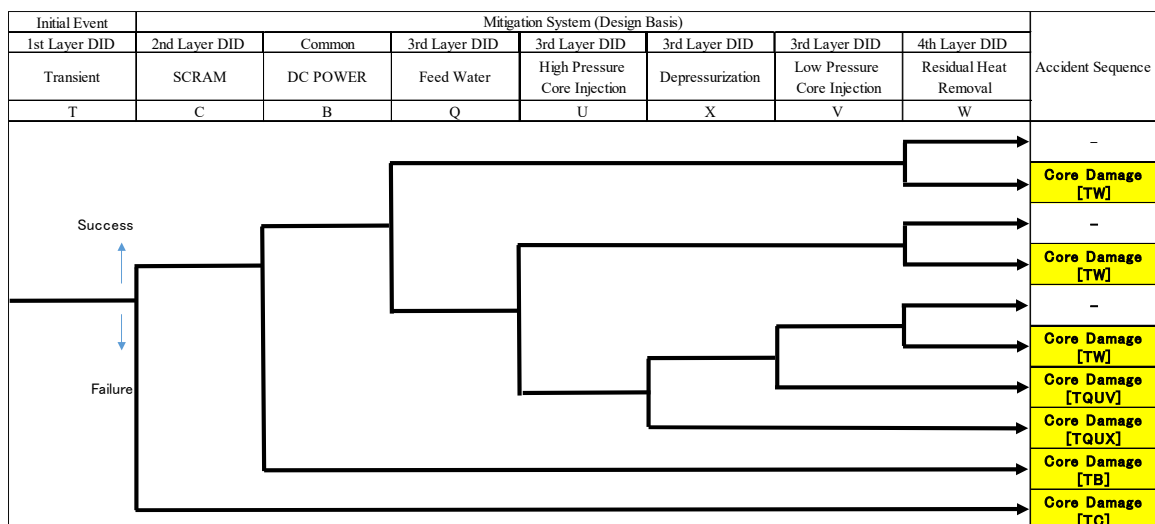


FIG. I-14-2. Selection of representative sequences using event tree.
(Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

I-14.2. Accident analysis for accident sequence

In order to confirm that additional safety measures are effectiveness for each selected accident sequences, safety evaluation is conducted. The safety evaluation was divided into several phases. First, the accident sequence that can prevent core damage by additional measures is evaluated. If the core damage cannot be prevented even if safety measures are used, the safety evaluation shifts to next phase evaluation for prevention of the primary containment vessel (PCV) damage. In addition, since core melt progression has large uncertainly, when evaluating after core damage, physical phenomena in PCV such as MCCI, FCI and DCH are evaluated under assuming conservative conditions that it cannot inject water into the reactor vessel.

I-14.3. Criteria for accident analysis for accident sequence

The accident evaluation for prevention of core damage scenarios is based on the criteria that Peak Clad Temperature (PCT) does not reach 1200 degree and Equivalent Clad Reacted (ECR) does not exceed 15% as threshold of fuel failure. It is necessary to promptly inject to reactor core with the alternative core injection system to prevent core damage. In addition, in order to maintain the stable core cooling, it is necessary to secure a residual heat removal function from PCV such as the venting system. The accident evaluation for prevention of PCV failure is based on the criteria that the atmosphere in PCV does not reach the limiting temperature (200 degree) and the limiting pressure (2PD) as threshold of containment boundary failure (See Fig. I-14-3.). On this accident scenarios, it is important to secure PCV heat removal systems (the filtered venting system or the alternative PCV recirculation cooling system). As explained above, effectiveness of additional safety measures for selected accident scenarios is confirmed using confirmed criteria.

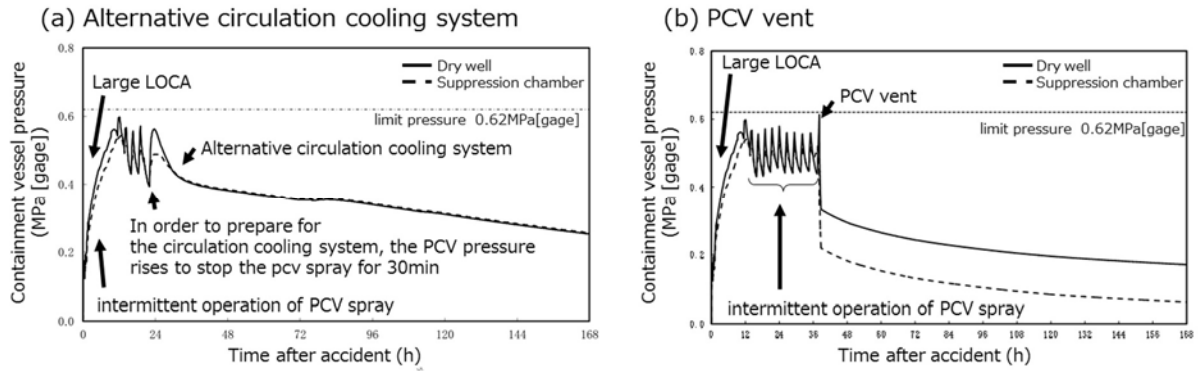


FIG. I-14-3. Example of DSA results (LBLOCA with core melting).
 (Courtesy of © Tokyo Electric Power Company Holdings, Incorporated).

I-14.4. Accident analysis codes and uncertainties

The PCT analysis code for prevention of core damage scenarios used SAFER (TQUV, SBO, LOCA, etc.), REDY/SCAT (ATWS), these codes are for Design Basis Accident using licensing report. Core damage analysis and PCV analysis are used MAAP code. It is necessary to confirm that codes are within the applicable range. For the purpose of realistically evaluation as much as possible, input Parameters are based on nominal conditions. Parameters with a large influence on the results need to conduct sensitivity analysis from the point of uncertainty and to verify whether the parameter affect the effectiveness of additional safety measures. In particular, the progression of core melting in RPV has large uncertainty. Severe accident countermeasures include both equipment and operational actions by emergency responder. When evaluating a severe accident, it is important that the uncertainty of the analysis confirm the impact to the timing of these.

I-15. LITHUANIA

Lithuania has only one nuclear facility, which is no longer operational. The Ignalina Nuclear Power Plant consisted of two Chernobyl NPP-type 1,500 megawatt RBMK reactors. Unit 1 at Ignalina came online in December 1983 and Unit 2 came online in August 1987. As a condition of Lithuania's European Union accession agreement, Unit 1 was shut down on 31 December 2004 and Unit 2 on 31 December 2009.

I-15.1. Regulatory framework

The fundamental safety objectives are set up in the provisions of the Law on Nuclear Safety and are intended to ensure the highest level of nuclear safety as possible to achieve and the lowest risk to people, their property and environment as regards the activities of both nuclear installations and any other activity related to the use of nuclear and (or) nuclear fuel cycle materials. The fundamental safety objectives are further implemented by, e.g., prescribing, that particular safety related decisions must be made in consideration of “risk to people, their property and the environment”. The fundamental safety objectives are in compliance with the fundamental safety principles established in IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles.

In addition, the Law on Nuclear Safety implements the provisions of Nuclear Safety Directive of European Union, concerning practical elimination accidents leading to large or early releases.

The Republic of Lithuania is a part of all the most important international treaties. The ratified international treaties as well as internationally recognised IAEA safety standards are taken into consideration while preparing the drafts of the Laws and other appropriate legal acts.

In accordance with the Law on Nuclear Safety State Nuclear Power Safety Inspectorate (VATESI) when drafting regulations shall follow principle of practical elimination of large or early releases following IAEA safety requirements as well as WENRA publications.

VATESI continuously monitor and incorporate into regulations international practices and lessons learned from local and foreign operational experience. Nuclear Safety Requirements BSR-1.1.1-2014 “Rules of Procedure for Drafting Nuclear Safety Requirements and Nuclear Safety Rules” require that compliance of draft regulations with IAEA, Western European Nuclear Regulatory Authorities Association (WENRA) and other international organizations documents would be evaluated and comparison with practices of other countries would be performed. [51]

In accordance with Nuclear Safety Requirements BSR-1.1.3-2016 “Inspections Conducted by the State Nuclear Power Safety Inspectorate”, inspector beside violations may indicate the non-conformity from good practice. Non-conformity from good practices means non-compliance with IAEA safety fundamentals, safety requirements, safety guides, principles of nuclear security, nuclear security recommendations, implementation of these recommendations and technical guides, WENRA safety levels, objectives and (or) other IAEA and (or) WENRA published information or other recognized good practices with regard to the safety. [52]

Due to development of nuclear energy sector that was planned and decommissioning of Ignalina NPP in Lithuania a legal and regulatory reform was performed in order to strengthen the regulatory body and to improve efficiency, transparency and streamline the regulatory process.

The amendments to the Law on Nuclear Energy of 2011 establish the basis for a stronger nuclear regulatory authority with functions clearly separated from the functions of other authorities, institutions or organizations engaged in development of the nuclear energy or use of nuclear energy, including production of electricity.

Regulatory body VATESI is now accountable to the President of Republic of Lithuania and the Government of Republic of Lithuania.

The Law on Nuclear Safety, adopted in 2011, among other provisions, establishes a detailed procedure for issuing licenses, permits and other types of authorization, including the documents required and conditions to be fulfilled in order for an activity to receive authorization. This law also establishes the main principles for safety assessment and provides for different types of enforcement measures, including economic sanctions (penalties) for the most severe cases of noncompliance with safety requirements. The right of VATESI to issue mandatory legal safety requirements was clarified as well.

In response to the events at Japan's Fukushima Daiichi Nuclear Power Plant, the stress tests were conducted in 2011 – 2012 at Ignalina NPP, in accordance with the ENSREG stress tests specification. On the basis of conducted stress tests several safety improvements measures associated with improvement of emergency preparedness and development of accident management at Ignalina NPP were included in the Ignalina NPP prepared plan of the stress tests safety improvement measures.

The Ignalina NPP prepared the plan of the stress tests safety improvement measures, which were approved by VATESI in May 2012. These safety improvement measures were included into the Ignalina NPP Safety Improvement Programme. The plan for upgrading of nuclear safety in Lithuania (National Action Plan of Lithuania) for improvement of nuclear safety encompassing INPP stress tests safety improvement measures as well as regulatory measures was prepared by VATESI and reviewed in ENSREG National Action Plans Workshop. This plan summarises the European stress test peer review results and other recommendations related with post-Fukushima lessons learned. The review of regulations in accordance of updated WENRA safety reference levels was included in the plan as well.

Moreover, the regulatory framework was updated in accordance with the amendment of Nuclear Safety Directive – specified provisions related to periodical safety review for nuclear installations, limitations of nuclear and radiological accidents consequences.

I-15.2. Identification of safety improvements

In accordance with the Law on Nuclear Safety, the operating organizations of nuclear installations and other holders of licences and/or permits has to analyse the level of nuclear safety on a regular basis and improve it.

Moreover, in accordance with the Law and secondary legislation, licensees have to constantly analyse the experience of its own and other persons operating in the nuclear energy field, to share and take the necessary preventive and/or corrective measures ensuring the proper performance of nuclear safety requirements.

Licensees have to make a periodic safety analysis and justification not less frequently than every 10 years and prepare a periodic safety evaluation report, which shall be submitted to VATESI for its review and evaluation. If deviations from mandatory requirements or design documentation are identified the corrective measures shall be timely applied. If there are identified areas for improvement of nuclear, radiation and physical safety and emergency preparedness, the licensee needs to identify safety improvement measures and implement it. These corrective and safety improvement measures and terms for their implementation have to be specified in the periodic safety evaluation report. These safety improvement measures need to be in line with proven engineering practice.

In accordance with requirements approved by the Head of VATESI, the nuclear, radiation and physical safety status of nuclear facilities needs to be continuously analysed and evaluated in the light of the latest research results, changes in international nuclear safety standards, own experience and experience of other persons acting in the nuclear energy field, and the information obtained would be used to provide safety improvement measures. The safety improvement programme needs to be reviewed and modified once a year, removing the safety measures that are implemented and adding new safety improvement measures, if necessary. The operating organization of the nuclear installations shall submit a safety improvement programme and its amendments to VATESI.

I-15.3. Drivers for the enhancement process

For identification of safety improvements at Ignalina NPP, there are the following main drivers:

- Requirements of National Laws and Regulatory Body;
- Analysis of own and external Operating Experience (Events related to safety assurance; Results of Tests and Inspections of Safety system components; Causes of Failures and Defects of Safety system components; Ageing management programme, Results of personnel training programme on emergency);
- Self-assessment and independent assessment processes of Licensee' (safety inspections, audits of management system, safety improvement programme implementation);
- External evaluations (Inspections of Regulatory Body, IAEA missions, WANO missions, different peer-reviews);
- Performance of periodic safety review.

All safety improvements implementing in accordance to Nuclear Safety Requirements BSR-1.8.2-2015 "Categories of Modifications of Nuclear Facility and Procedure for Performing the Modifications," issued by regulatory body.[53]

I-15.4. Selection process of safety improvements

The selection of safety improvements is performed by the licensee based on the following methods:

- Initially, the assessment of the safety impact of the improvement is based on engineering judgment or a simplified analysis.
- The further analysis investigate the level of the safety improvement and describe the work and time needed to make the improvement, together with an estimated efforts. Here PSA, DSA and other analyses may be used.
- Based on this above presented analysis and based on the budgetary situation the improvement that gives the highest improvement for the money (cost benefit analysis) is chosen. An identified improvement is then planned for installation and a project starts in accordance with this plan.
- For regulatory request the same process is followed but only to define the scope of work. There is also in this area a discussion regarding “Reasonable and Practicable.”

I-15.5. Outcomes identified of safety improvements

The safety improvement implemented after permanent shutdown of both reactor units in Ignalina NPP (according the Ignalina NPP plan of the stress tests safety improvement measures, approved by VATESI in May 2012):

- To evaluate (including radiological impact) the spent fuel cask tip over in case of earthquake during transportation;
- To assess (and to improve) the robustness of accident management centre of organization of emergency preparedness against an earthquake;
- To provide data transfer of the seismic alarm and monitoring system to the computer information system of organization of emergency preparedness;
- To improve procedures for the organization of emergency preparedness in the case of Beyond Design-Basis Emergency scenarios for the operating spent fuel interim storage facility and new spent fuel interim storage facility;
- DG and Mobile DG connection to the temperature and level indicators of the SFP;
- Use of domestic potable water pumping system with own backup DG as diverse heat sink cooling SFPs;
- Modifications to supply Unit 1 systems by Unit 2 DGs;
- Make the mobile DG connections for power supply backup of the I&C important to safety, radiation monitoring system, communication system, recharge points for the batteries of flashlights, temperature and level meters of SFPs.

Detailed design of safety improvements

The terms for implementation of corrective measures shall be set taking into account, that in accordance with the legal framework, the nuclear facility cannot be operated if legal requirements are violated.

In accordance with Nuclear Safety Requirements BSR-1.1.3-2016 “Inspections Conducted by the State Nuclear Power Safety Inspectorate”, in addition to violations, the inspector may indicate deviations from good practice, what might be deviations from the provisions of IAEA safety standards, in the report of the regulatory inspection. In that case the operating organization is asked to take decision on corresponding corrective measures and terms for implementation of them. The decision is upon the operating organization. [54]

In accordance with law on Nuclear safety, if during periodic safety analysis and justification are identified areas for improvement of nuclear, radiation and physical safety and emergency preparedness, the licensee has to identify safety improvement measures. These safety improvement measures and deadlines for their implementation have to be specified in the periodic safety evaluation report. The decision on terms on implementation of when mainly is upon the operating organization.

I-16. NETHERLANDS

This annex describes the regulatory approaches and practical implementation of reasonably practical safety improvements at the only NPP in the Netherlands: NPP Borssele (Siemens KWU, 2-loop PWR, 1975). The plant has undergone a long range of safety improvements since its start-up through back fits, increasingly more structured (periodic) safety reviews, complementary safety assessment (stress test) and lessons learned from internal and external events. The cumulative effect of all improvements on the Total Core Damage Frequency (per year) is shown in Fig. I-16-1 [55]:

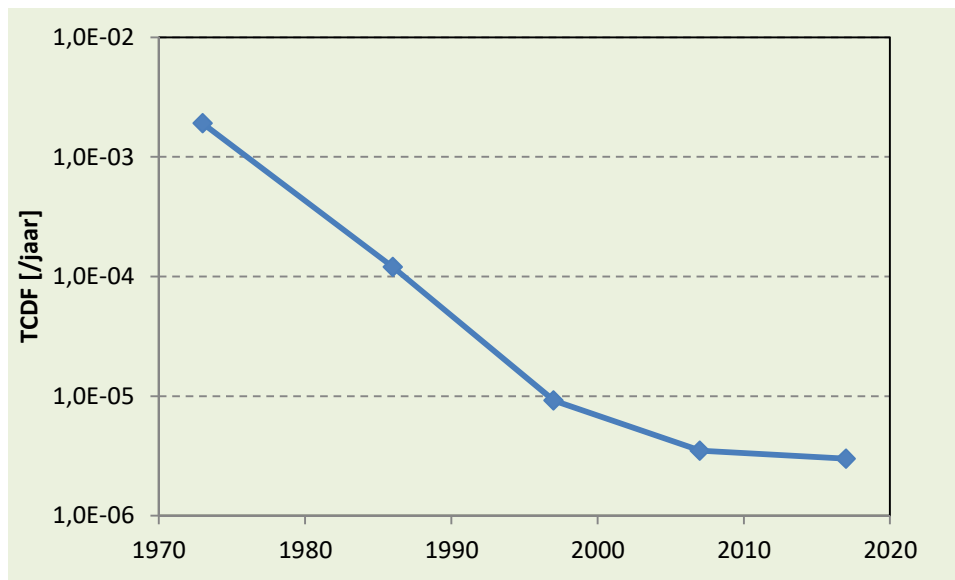


FIG. I-16-1. TCDF (/year) evolution in time. (Courtesy of EPZ)

In general a lot of information about the regulatory framework, the regulatory authority, the application of PSA and the implemented nuclear safety improvements in the history of the power plant can be found in the National Report on the Convention on Nuclear Safety.

I-16.1. Regulatory framework and practices

As in other countries, the Netherlands regulatory framework consists of a regulatory pyramid with layers of legally binding regulations, starting with the Nuclear Energy Act, supplemented by non-legally binding guidance. Details of this are available in the National Report to the Convention on Nuclear Safety (CNS). An important Fukushima Daiichi NPP accident related modification to the regulatory framework was the transposition of the adapted EU Nuclear Safety Directive from 2014 into a Ministerial Order in 2017. This for example strengthened the requirements on the independence of the Nuclear Safety Authority, the application of defence in depth, periodic safety review, and safety assessment to improve existing NPPs and safety culture.

Both the upcoming changes in the EU-directive as well as the IAEA IRRS-mission of 2014 around 2012 triggered a review of the regulatory body, leading to a decision by the Government in 2014 to create a new single authority, and locate it under the Ministry of Infrastructure and Water Management. Since 2015, the newly formed Nuclear Safety Authority in the Netherlands ANVS (Authority for Nuclear Safety and Radiation Protection; www.anvs.nl) started its activities, integrating a number of regulatory tasks that were located in different

ministries. Per the 1st of May 2015, the responsibility for the Nuclear Energy Act moved from the Ministry of Economy to the Ministry housing the ANVS. Per the 1st of August 2017, the ANVS got legally the status of an independent administrative organization. The ANVS will employ about 140 staff, supported by some TSO's, e.g. GRS (German safety advisor to the central government), and is responsible for proposing policies and legally binding law and regulations, licensing, supervision and enforcement as well as public communication, advising on emergency response and international cooperation.

The design of the reactor originated from German specifications and standards. As a small country, the Netherlands decided not to develop their own detailed safety requirements, but to make use of IAEA Standards, German and US regulations. They were used as reference (current at that time) during what now can be called a first comprehensive safety review after the Chernobyl NPP accident. Then it was decided to adopt IAEA safety standards as NVR (Dutch Nuclear Safety Rules) and attach them to the license as a condition. The number of NVRs was growing gradually this then. An updated set of NVRs was created by 2010 and included in the license by 2011 (a license modification was implemented to include MOX fuel in the core). It includes the safety requirements and guides on Siting, Design, Safety Assessment, Operation and Emergency Preparedness. It is interesting to note that the IAEA Standard NS-R-1 [56], applied as NVR-NS-R-1, in 2010 was modified to include: "An essential objective is, that the need for external intervention measures is limited or even eliminated in technical terms, although such measures may still be required" [57] and "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} /reactor year". [57]

In Europe, the WENRA (association of European Nuclear Authorities) has published so-called Safety Reference Levels (SRLs) around 2007 to harmonize safety. They have been included in the NVRs in 2010. After Fukushima Daiichi NPP accident WENRA modified and expanded the SRLs in 2014. The next NVR-update will be around 2020. In the CNS-report, a table with the current set of NVRs is included.

In 2009, there were two initiatives for a new NPP and one initiative for a new Research Reactor. This triggered the creation of Dutch Safety Requirements for New Reactors, that includes Safety Objectives of New Reactors (WENRA) [21], multiple failures, and accidents with core melting. For the conception of this guidance, ANVS used modernized German design requirements, the latest IAEA design requirements, WENRA Safety Objectives for New Reactors [21], WENRA reference levels for existing reactors (version 2014, as modified by Fukushima lessons) and other Fukushima insights [7]. This document was published as a guidance in 2015. It is applicable to NPPs, but contains an annex describing how to apply it to RR. Further, it will be applicable as reference for the existing reactor during the next PSR (evaluation 2021-2023, implementation 2024-2029).

Based on a decision by the European Council and a WENRA Specification adopted by ENSREG in 2011 the European stress test was carried out, culminating in an action plan end of 2012. The stress test and the implementation of measures were imposed, using an article in the Nuclear Energy Act, giving the possibility to request additional safety assessments. Most of the actions were implemented by 2017, together with a modification programme from the latest PSR.

The license of the NPP is not limited in time, but reactor life will be ending 31 December 2033 by law (60 years old). It contains the requirement for Periodic Safety Review every 10 years and implementation of improvements within 5 years.

I-16.2. Regulatory limits to be met by the design in terms of radioactive releases or dose limits to the public

For existing and new reactors, the probability of occurrence that a person, located permanently and unprotected outside the facility, dies as a result of an accident, shall not exceed 10^{-6} per year. The probability of occurrence that a group of at least 10 persons would directly die as a result from an accident shall not exceed 10^{-5} per year, or for n times more fatalities a probability that is n^2 times smaller.

At levels 2 and 3 of DID (abnormal events and prevention of accidents; accidents without core melt:

- There shall be no off-site radiological impact or only minor radiological impact;
- The maximum radiation exposure of personnel in connection with the planning of activities for the control of events, the mitigation of their effects, or the elimination of their consequences, shall be kept ALARA and shall not exceed the relevant limits for normal circumstances as specified in the Radiation Protection Decree, taking into account all circumstances of individual cases;
- Any release shall only happen via the therefore intended release paths; the release shall be kept as low as reasonably achievable and shall not exceed the limits for the public shown in Table I-16-1, taking into account all circumstances of individual cases.

TABLE I-16-1. EVENT FREQUENCY PER YEAR AND MAXIMUM ALLOWABLE EFFECTIVE DOSE PER PERSON

| Event frequency per year | Maximum allowable effective dose per person (over 70 years) |
|--------------------------|---|
| $F > 10^{-2}$ | 0,1 mSv |
| $10^{-2} > F > 10^{-3}$ | 1 mSv |
| $F < 10^{-3}$ | 10 mSv |

For new reactors at level 4 of defence in depth, accidents with core melt:

- The total probability of occurrence of accidents with core melts shall be as low as reasonably achievable, but shall not exceed 10^{-6} per year.
- Accidents with core melt that would lead to early or large releases shall be practically eliminated.
- Or accidents with core melt that cannot be practically eliminated only limited protective measures in area and time shall be needed for the public (no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no permanent relocation, no long term restrictions in food consumption) and sufficient time is available to implement these measures.
- The maximum radiation exposure of personnel in connection with the planning of activities for the control of postulated core melt accidents, the mitigation of their effects, or the elimination of their consequences, shall be kept ALARA and do not exceed the limits for intervention as is specified by EURATOM when implementing ICRP103, taking into account all circumstances of individual cases.

- Any release of radioactive materials from the plant shall be monitored and documented and specified in accordance with the type and activity of release.

I-16.3. Identification of safety improvements

I-16.3.1. Drivers for the enhancement process

Main drivers for improvements are the PSRs and lessons learned from TMI NPP, Chernobyl NPP, and Fukushima Daiichi NPP accident. At the time before the PSR was a legal requirement it was the regulatory authority, that requested evaluations and improvements. The large modification programme of the nineties was based on a GRS comparison of the plant with modern German designs. This analysis was an activity requested by the regulatory authority. Nowadays the PSR is the structured approach, where the licensee proposes the scope and review basis, including additional requests by the regulatory authority. The authority finally agrees. This document is created in an interaction with the regulator. An important basis for the PSRs is IAEA Safety Standard Series No. SSG-25 [4] on Periodic Safety Review of NPPs. The improvements are identified e.g. by comparing the existing plant with more modern standards and designs, carrying out weakness analyses with PSA, operating experience and regulatory list of potential improvement.

Safety improvements have been done in the 1980-ties, introducing already bunkered backup cooling systems. PSRs have been introduced about 25 years ago and from the beginning did the comparison of design with modern plants and regulations. Results of comparison with the most modern German plant (Convoi) led to 16 large modifications in 1993-1998, including filtered venting, PARs, introduction of more bunkered systems, increased defence in depth and application of design principles like redundancy, separation, single failure and Westinghouse EOPs and SAMGs. The introduction of Westinghouse EOPs and SAMGs itself introduced the need to modify. More improvements have been done in further PSRs (2001-2003, 2011-2013). The latest major safety improvements were based on PSR and post Fukushima stress test. The main period of analysis was 2011-2012, and implementation was done in 2013-2017, with remaining elements for 2018-2019. In 2017, In Vessel Retention was implemented. Refer to the National Action Plan stress test²⁵ and the lists of safety improvements based on three PSRs as part of the report to CNS7.

The plant designer (AREVA-Germany) and independently the designer of Belgian plants (Tractebel) play an important role in support of the licensee. The authority has GRS as her TSO for reviewing many PSR reports or modification plans. GRS also supports ANVS in the future to search for potential safety improvements.

The time duration is about three years for the evaluation and 5 years for implementation.

For the stress test, the identification was based on a specification to carry out analysis what improvements and additional means can be applied if more and more existing provisions fail. Leading to adding all kinds of mobile equipment for backing up electricity supply or water supply, but also creation of alternative way to cool and refill the fuel pool, increase robustness of emergency response centre, robustness improvements related to earthquake (e.g. firefighting and venting system) and improvements of emergency operating procedures and severe accident

²⁵ Information on the stress test is available at < <http://www.ensreg.eu/EU-Stress-Tests> >

management guides. Several mobile equipment have been added to the protection concept. How to deal with them has been described in the Flexible Support Guidelines, which will be used from the EOPs and or SAMGs.

License renewal is not applicable as such, but we did an LTO-programme in accordance with IAEA Safety Report Series No. 57 on Safe Long Term Operation of Nuclear Power Plants, including some additional Safety Factors from the PSR. The programme was accompanied by a series of SALTO missions (2009, 2012 and 2014). It started 2008 and led to a license modification in 2012. LTO-actions are still going on till 2020. Next PSR evaluation will be 2021-2023.

Covenant to belong to the best 25% of water-cooled reactors in the western world. Independent benchmark commission produces a report every 5 years. Next report in 2018.

I-16.4. Selection process of safety improvements

The general steps to select reasonably practicable safety improvements are based on a structured approach, as included in Fig. I-16-2 [55]. It starts with the analysis phase, resulting in a report containing findings within each of the 14 Safety Factors (SF) described in IAEA Safety Standards Series No. SSG-25 [4]. Then a global assessment takes place, clustering findings, determining potential safety importance and estimating the costs. The global assessment finishes with a set of potential improvement measures. In the subsequent Integrated Safety Assessment the decision is made about measures to be implemented. Then these measures are included in the Conceptual Improvement Plan. The potential safety importance is determined using both a deterministic and probabilistic safety matrix with 5 levels, ranging from very small to very large: see Table I-16-2.

TABLE I-16-2. POTENTIAL SAFETY IMPORTANCE

| | | Potential Safety Importance | | | | |
|--------|-----------|-----------------------------|-------------------------------------|---|--------------------------|-----------------------------------|
| | | very small | small | average | large | very large |
| Action | No action | | Risk-evaluation needed, weigh ALARP | Attention required, weigh safety improvement vs. cost | Measures can be planned. | Measures are implemented a.s.a.p. |

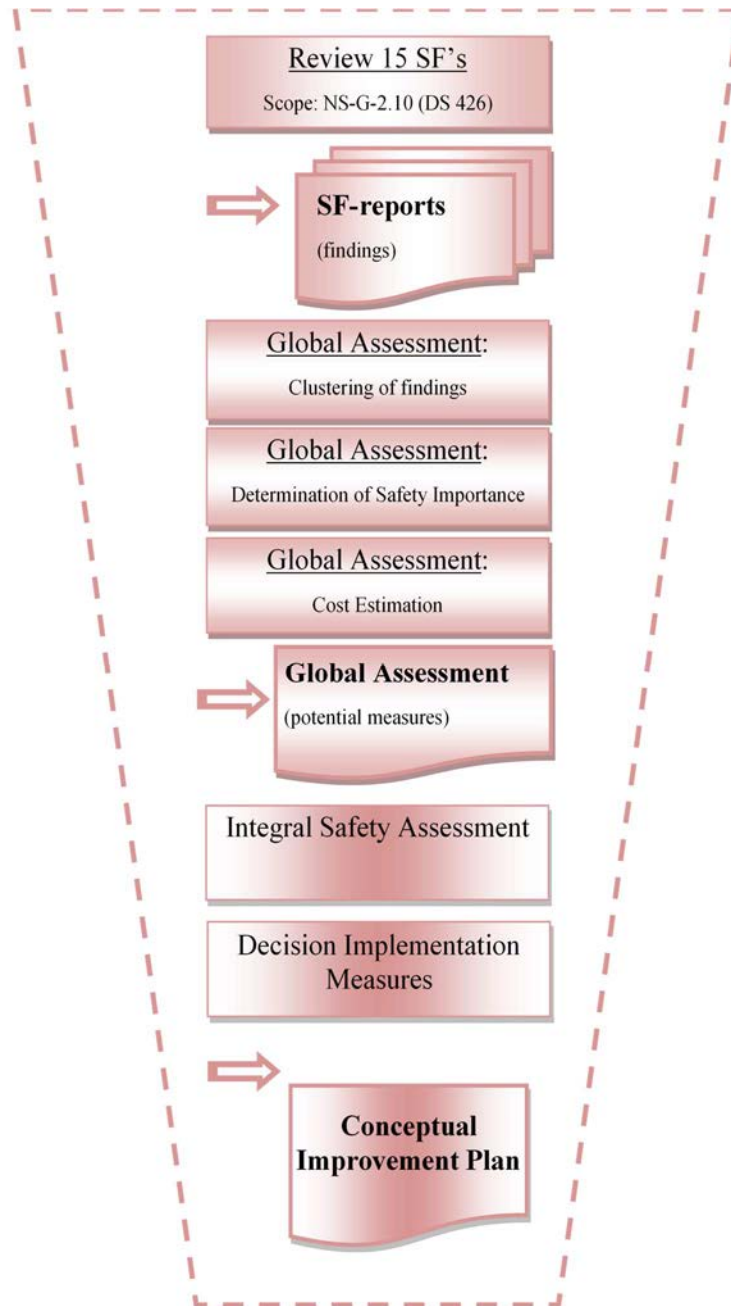


FIG. I-16-2. Global steps for the identification of safety improvements. (Courtesy of EPZ)

The safety impact versus cost determines the decision, as shown in Table I-16-3.

TABLE I-16-3. SAFETY IMPACT VERSUS COST DETERMINES THE DECISION

| | | | | | |
|---------------------------------|------------|-------------|---------------|-----------|------------|
| Deterministic importance | very small | small | average | large | very large |
| Probabilistic importance | very small | small | average | large | very large |
| Cost | < 20 k€ | 20 - 200 k€ | 200 k€ - 2 M€ | 2 - 10 M€ | > 10 M€ |

I-16.5. Introduction of the identified safety improvements

After the measures to be implemented have been chosen, there will be created a planning, taking into account e.g. safety importance in the prioritization. In accordance with the license, measures have to be implemented ultimately within 5 years, unless this is not reasonable.

Each measure will be designed in detail, in accordance with the normal modification procedures as agreed with the regulatory authority.

I-17. ROMANIA

The only nuclear power plant in Romania is Cernavoda NPP, PHWR type, CANDU-6 project (700 MWe), (CANDU – CANAdian Deuterium Uranium), with 2 units in operation and other 2 under preservation, as follows:

- Cernavoda NPP Unit 1 in operation since 1996;
- Cernavoda NPP Unit 2 in operation since 2007;
- Cernavoda NPP Units 3, 4, currently under preservation (there are plans to restart their construction, stopped in the past in different stages).

I-17.1. Regulatory framework

Romanian framework in the nuclear field consist basically from a specific act, with high level statements regarding the regulation and control of the domain, Law no. 111/1996 on the Safe Deployment, Regulation, Licensing and Control of Nuclear Activities in Romania, republished, and regulations (norms with requirements and guides) issued by the Romanian regulatory authority, CNCAN (National Commission for Nuclear Activities Control), empowered by Law 111/1996.

Fundamental safety objectives and expectations in the nuclear field are expressed in the Romanian regulatory framework mainly in the Law no. 111/1996, modified and republished, and in the NSN-21 regulation “Fundamental norms on nuclear safety for nuclear installations”, issued in May 2017. Fundamental safety objectives in Romanian regulatory framework in nuclear domain are in line with IAEA safety fundamentals, WENRA safety objectives, and EU safety directives.

As an IAEA Member State, Romania through the Romanian regulatory body takes into consideration, in the process of development or revising the regulatory requirements and recommendations, the IAEA safety standards and, as much as possible, participates directly with qualified staff members in the process of development of IAEA standards. In the same time, Romanian regulatory body is part of WENRA (Western European Nuclear Regulators Association) since 2005 and it observes the WENRA Reference Levels in the development of its regulations. A set of new regulations have been developed by the Romanian authority, mainly during 2006-2010, after benchmarking the Romanian national regulatory framework against WENRA reference levels, that resulted in a large plan of actions to develop new regulations or to revise the existing ones. As a member of European Union, Romania takes into consideration, in the development of its regulatory framework, the European directives regarding nuclear and radiologic safety, as well as other recommendations resulted from international treaties or cooperatively developed approaches (e.g. EURATOM, CTBTO, BSS), including at the highest level (Law 111 amended to align to EU directives).

Development of new regulations or revision of the existing ones is a continuous core process to the Romanian regulatory body, but significant events in the field have determined important changes in the regulatory framework. Following the nuclear accidents at the Fukushima Daiichi NPP in 2011, Romanian participated together to the other European countries with NPPs in operation to the safety reassessment organized by the European Commission under the name of “stress tests”. New analyses, regulatory inspections and reviews as well as peer-reviews performed by other regulatory bodies members conducted to actions comprise in the “Romanian National Action Plan”, including actions for the Romanian regulatory body too. These actions were implemented, and a number of regulations were issued, with increased

requirements to cover situations as in Fukushima Daiichi NPP accident case, especially considering the need for an adequate response to extreme external conditions and for severe accident management. Reaction of the licensees direct to such event, as well as to recommendations resulted from stress tests and to the new regulatory requirements determined important plant safety improvements to the NPP operating units. The capability of Cernavoda NPP units to respond to external events and for accident management, including for severe accident conditions, increased.

Considering the lesson learned from Fukushima Daiichi NPP accident, Romanian regulatory body increased also requirements for safety analysis (for severe accidents) and for safety margins demonstrations, as for seismic margins (0.4g PGA seismic margin, for a Design Basis Earthquake of 0.2 g PGA), and the inspection process improved.

I-17.2. Identification of safety improvements

I-17.2.1. Drivers for the process of safety enhancement

The following drivers are mainly conducting to safety improvements to operating NPP' units in Romania:

- Increasing in regulatory requirements, as a response to Nuclear Safety Convention decisions, IAEA standards, WENRA reference levels, EU directives;
- Regulatory requests, as a reaction to regulatory inspections, oversight, audits or to operating experience or R&D results with important safety significance;
- Periodic Safety Review process;
- NPP internal self-assessment and audits, licensee' oversight process;
- Operating Experience (OPEX) including operational experience in COG (CANDU Owners Group) countries; analysis of operating events;
- External evaluations (IAEA missions, WANO missions, different peer-reviews);
- Licensee's programme for safety improvement;
- Ageing management programme, including obsolescence and the need for modernization ;
- PSA results, live PSA and PSA reviewing by the licensee;
- Regulatory review of nuclear safety analyses (DSA, PSA, severe accidents analyses);
- Safety re-assessments performed following important events (as in the frame of Stress Tests, 2011-2012, after Fukushima Daiichi nuclear power plant accident) or important European or international projects (e.g. EU Topical Peer Review 2017 on "Ageing Management of Nuclear Power Plants" [58]).

I-17.2.2. Selection process of safety improvements and decision of implementation

The selection of safety improvements is performed by the licensee based on the following process:

- Assessment of the safety impact of the improvement proposed by the licensee starting from different reasons/issues (deterministic approach or engineering judgement). The results of this assessment of safety impact of each solution has also to consider the time available/required for the implementation.
- Implementation as soon as possible for high safety impact modifications that usually come from a regulatory requirement, request, regulatory inspection, a PSR finding with high level safety significance or from operating experience (to fix a problem discovered, with high safety significance).
- Implementation based on an action plan, for modifications that are coming from PSR findings with medium or low safety significance, for safety improvements to align NPP to current standards requirements, improvements coming from NPP internal self-assessment and audits, from - ageing management programme, to solve some generic or project safety issues (eventually discovered in a peer-review), improvements coming from operating experience (with medium or low safety significance).
- Risk informed decision at the licensee (as for example for improvements that are coming from PSA evaluation performed by the licensee), to reduce radiologic risk – as by decreasing the CDF (Core Damage Frequency) with efficient modifications and relatively not very expensive or complicated NPP modifications.
- A cost benefit analysis could be conducted internally by the licensee in case of improvements resulted from the PSA evaluation or from NPP safety improvements programme (selection of modifications with highest safety impact and reasonable costs, as the licensee is promoting these modifications); this analysis is not required or considered by the regulatory body in the analysis of modification proposal approval.

Romanian regulatory body requires in Art. 5 of the NSN-02 regulation “Nuclear safety norms on the design and construction of nuclear power plants“, [59] issued in 2010: “The design and construction of a new NPP shall include all reasonable and practicable measures in order to prevent the events that might lead to the exposure of workers and the public above the allowed limits by the legislation in force. All reasonable and practicable measures shall also be considered in order to limit the consequences generated by nuclear accidents, for situations where such events can occur”. This requirement for new NPPs is applied for the existing ones based on a Periodic Safety Review approach, in which the operating NPPs are improving their safety with the aim to go as close as reasonable possible with the current standards and fulfil requirements and safety objectives for new NPPs. Differences need to be identified and justified by the licensees.

The safety objectives for new NPPs presented in NSN-02 regulation are also applicable for operating NPPs based on a PSR approach, considering activities of safety improvement to bring the plant as much as possible toward the level (regarding dose limits) required for new NPPs. In accordance with NSN-02, the safety objectives are represented by dose limits of radioactivity for different classes of events and accident sequences, as shown in Table I-17-1.

TABLE I-17-1. QUANTITATIVE NUCLEAR SAFETY OBJECTIVES FOR THE DESIGN OF THE PROTECTIVE SAFETY SYSTEMS (DESIGN BASIS QUANTITATIVE OBJECTIVES)

| Event Class | Estimated annual frequency of occurrence for an event or sequence of events <i>(95% confidence value)</i> | Maximum value for the effective dose for the most exposed individual situated outside the exclusion zone <i>(calculated for 30 days since the start of the release, for all paths of exposure)</i> |
|-------------|--|---|
| Class 1 | $f > 1E-2$ | 0.5 mSv |
| Class 2 | $1E-2 > f > 1E-3$ | 1 mSv |
| Class 3 | $1E-3 > f > 1E-4$ | 10 mSv |
| Class 4 | $1E-4 > f > 1E-5$ | 50 mSv |
| Class 5 | $1E-5 > f > 1E-6$ | 100 mSv |
| Class 6 | $1E-6 > f > 1E-7$ | 250 mSv |

Currently, Romanian regulations do not use the concept of practically elimination, even if the NSN-02 regulation requires analysis and provides dose limits for the “credible” event sequences, till $10e-7$ events/year frequency. Romanian safety regulations require application of the defence in depth concept, the safety analyses performed for NPPs having to demonstrate the adequate implementation of this concept and to demonstrate that the safety objectives are fulfilled.

Specific safety improvements are proposed in Romania by the licensee, starting from different drivers and passing through the licensee’ internal assessment and approval process, based on specific procedures. Modifications with safety impact have to obtain the regulatory approval before their implementation. This approval decision is taken only after the analysis of the safety modification impact and review of the support analyses, assessments and safety documentation submitted by the licensee, as applicable.

I-17.3. Outcomes identified of safety improvements

The main outcomes considered for the safety improvements are:

- Fulfilment of the requirements of the new regulations and the requirements and recommendations of new applicable safety standards;
- Increasing in the robustness of the NPP – prevention of transients and accidents, improving the defence in depth;
- Decreasing of doses resulted from accidents as a result of development of mitigative measures and improving of emergency preparedness and response actions (demonstrated in safety analyses);
- General increasing in the nuclear safety for the existing NPPs, as close as reasonable possible to the new NPPs, including by improving safety culture;
- A lower value for CDF (Core Damage Frequency), toward the values required for new NPP’s, in accordance with current standards;
- Elimination of project safety issues, if some are identified;
- Use of operating experience, applicable or in similar NPPs;
- Alignment to the best international practice.

I-17.4. Detailed design of safety improvements

Detailed design of safety improvements needs to consider all regulatory requirements related to SSCE of NPPs. Special requirements for design exist in the Romanian regulation NSN-02 “Nuclear safety norms on the design and construction of nuclear power plants” for safety related systems and their support systems; furthermore, for CANDU type NPPs, Romanian regulatory body endorsed Canadian regulations related to special safety systems (2 reactor shutdown systems, containment systems and Emergency Core Cooling System).

Qualification of the new SSCE has to be established by the licensee and reviewed by regulatory body, based on the safety classification of the respective SSCE, considering safety functions that shall be fulfilled by the new or modified SSCE and specific conditions in which these functions are fulfilled (including for severe accidents, as the case). A stronger qualification might be required by the regulatory body in some cases; for example, Romanian regulatory body required a seismic margin of 0.4g PGA to be demonstrated for systems that are qualified to remain functional after a DBE (Design Basis Earthquake), for a better protection of the plant against this type of extreme external conditions. This regulatory requirement occurred after Fukushima Daiichi nuclear power plant accident and European NPPs and “stress tests”. Quality assurance requirements are also well established in specific regulations, for SSCE of different safety classes and different types of activities, and their application in safety improvements is mandatory.

The regulatory body approval is necessary for the implementation of the safety improvements proposed by the licensee, resulted from the licensee’ initiative (from different reasons) or as response to a regulatory requirement, to a recommendation coming from a peer review or a mission (IAEA). The licensee’ procedure “Operating principles and policies”, reviewed and approved by CNCAN, establishes what kind of modifications require CNCAN approval. Besides this, CNCAN approval is necessary for all modifications to the plant hardware, operation or organization that can affect the design or licensing bases (affecting for example safety analyses results, operating or process procedures that require CNCAN approval, modifications in the licensing documentation – as in the Final Safety Analysis Report).

For very expensive or complex plant modifications, a principle approval is first obtained usually by the licensee from the regulatory body, this approval being accompanied by different regulatory conditions regarding the design, qualifications, additional safety analyses or other types of analyses (reliability, mechanical, hazard) to be performed and submitted to the regulatory body before modification implementation. Documentation with detailed design and necessary analyses are usually submitted by the licensee to the regulatory body in a second stage (at regulatory demand) to demonstrate how the new or modified SSCE fulfil the design requirements, qualifications to environmental conditions, seismic qualification, including all additional conditions required by the regulatory body in the principle modification approval letter. Such important modifications are usually accompanied by regulatory inspections, to verify conditions for implementation.

The design of the new or modified SSCE, to improve safety, have to be realized only by the organizations that are licensed by the Romanian regulatory body for the Management System, having demonstrated qualifications in that area of activity. Audits and inspections are conducted by regulatory body to these organizations, in order to obtain a Management System licence. As a rule, until now, for modifications with important safety impact, the plant project designer is questioned, and traditional or proven manufacturers are selected.

Regulatory requirements related to the safety analysis increased too in the last decade, severe accident analyses and Level 1 and Level 2 PSA being required for relicensing of NPP. In the same time, there were developed a set of specific guides and procedures to be used by the regulatory body staff for the review of safety analyses and documentation submitted by the licensee to the regulatory body for different applications. Safety analyses or other types of analyses that support safety improvements have to demonstrate their efficiency and in the same type to avoid the negative impact on other systems/functions in NPP.

Currently, there are not regulatory requirements regarding the time frames for implementation of safety improvements. However, for each type of safety improvements, depending by the driver, there are action plans established by the licensee and usually approved by the regulatory body; for example, safety improvements resulted from the stress tests were included in the “National Action Plan”, safety improvements resulted from PSR are implemented based on the action plan established with regulatory body, safety improvements resulted from different missions or peer-reviews are implemented based on a specific action plan too.

Romanian regulatory body requires plant configuration control and modification of all affected documents after implementation of a plant modification. The licensee has the duty to maintain a strict control of the plant configuration, the process procedure being approved by regulatory body, and this is verified in the regulatory inspections.

I-17.5. Important safety improvements to operating NPP units in Romania

In the last decade, many and important safety improvements were implemented at the Romanian operating NPP units. Safety improvement is a permanent concern but one driver, that conducted to the implementation of many and important safety improvements was represented by the safety reassessment performed after the Fukushima Daiichi NPP accident.

This safety reassessment included the review of the effective implementation of the defence in depth concept, considering mainly external hazards such as seismic, flooding events and extreme weather conditions, their most unfavourable consequences as the total loss of power (SBO- Station Blackout) or loss of ultimate heat sink (UHS), the measures for prevention and mitigation of severe accidents and adequacy of existing Severe Accident Management Guidelines (SAMG) and emergency preparedness and response. Reviews included in the same time the resistance of the defence in depth barriers but also the availability and adequacy of the equipment necessary to support this, the diversity, redundancy and qualification of this equipment, required to mitigate external hazards.

The reviews performed by the licensee, verified by the regulatory body, as well as the peer-reviews performed in the frame of the “stress tests”, showed that Cernavoda NPP units have a good robustness against external hazards beyond those conditions explicitly considered in the Design Basis. This resistance is due to the inherent safety features of the CANDU-6 design, but also to the continuous improvements implemented during the time, determined by both the licensee and regulatory body attitude regarding the nuclear safety. The CANDU-6 design comes with independent, separated and diverse safety systems, Special reliable Safety Systems (including two fast reactor shutdown systems), multiple physical barriers, multiple and large inventory of water surrounding the reactor core and calandria vessel, a large amount of water for passive supplying of boilers for many hours, spent fuel pools with no need for cooling for a relatively long period of time due to the low power density of the fuel stored. Administrative measures, good procedures and adequate emergency preparedness contributed too to the good results obtained from this safety reassessment (stress tests).

Some important safety improvements implemented at Cernavoda NPP are, beside others:

- Provision of equipment and fast connections to ensure power supplying after any internal or external hazard that can conduct to the loss of those provided by project (two large mobile Diesel generators, smaller mobile Diesel generators, adequate fuel inventory for large autonomy and qualified storage tanks and locations).
- Provision of equipment to provide water for cooling in essential systems of the plant, as necessary (primary circuit, boilers, Emergency Core Cooling, Spent Fuel Pool) – mobile pumps, already installed connections and piping, fire tracks, deep wells.
- Provision of seismically qualified locations on site for the storage of the mobile equipment required for emergency conditions; procedures for testing the mobile equipment, sufficient and qualified personnel to use the new equipment.
- Improvement of the seismic robustness of the existing Class I and II batteries, solutions to ensure power to instrumentation and control after batteries depletion.
- Installation of systems and equipment to ensure water inventory to the boilers (from a passive water source – dousing tank), in the calandria vessel (surrounding the pressure tubes) and in the calandria vault (surrounding the calandria vessel) to provide cooling when the normal and protective cooling systems are lost, in order to avoid the loss of barriers and stop the evolution of the accident. These new provisions consider the loss of power conditions, rely mainly on manual actuation of valves, fire tracks or on mobile pumps and diesel generators that could be easily installed.
- Provide equipment and procedures for boilers depressurization during SBO to ensure conditions for addition of water in boilers.
- Protection of the containment over-pressurization, in order to avoid large radioactive releases. This is done by installation of emergency filtered containment venting systems (EFCVS) to both Cernavoda NPP units, to act during a severe accident.
- Hydrogen management and monitoring, for containment protection too. Passive autocatalytic recombiners (PARs) were installed to both units, to prevent detonation and deflagration. Hydrogen concentration monitoring system installed to verify conditions for implementation of accident management measures during a severe accident.
- Instrumentation for monitoring the plant conditions during severe accidents (qualified to these conditions).
- Improvement of seismic robustness of the existing site Emergency Control Centre; construction of a new, larger and qualified site Emergency Control Centre (still in progress).
- Revision of SAMGs (Severe Accident Management Guides) to be specific for each unit and to consider all plant operating states (including outages). Verification and validation of Emergency Operation Procedures and Severe Accident Management Guides.
- Improvements in the Emergency preparedness and response. Qualification of the technical and operational personnel, adequate number of qualified personnel, improvements in exercises and drills, protocols established for emergency situations with different organizations involved.

I-18. RUSSIAN FEDERATION

Regulatory safety of nuclear power plants, implementation of technical measures to eliminate safety deficiencies (by the example of NPPs with vver-1000 reactors of small series), and stress test methodology.

I-18.1. Introduction

In the Russian Federation, there is one nuclear power generation utility - the Russian Concern for the Production of Electric and Thermal Energy at Nuclear Power Plants (Rosenergoatom Concern JSC), which includes 10 operating nuclear power plants.

In accordance Russian Federation Government decree, adopted in 2002, the rules were established for allocating funds to target reserves, designed to ensure the safety of nuclear power plants at all stages of their life cycle.

These funds, as a share of the revenue from the sale of electricity (power) produced using atomic energy, may be used only for the following purposes:

- To ensure nuclear, radiation, technical and fire safety;
- To ensure the physical protection of nuclear power plants;
- To ensure the decommissioning of nuclear power plants;
- To ensure the disposal of radioactive waste.

The list of safety measures is determined on a systematic basis as part of the state safety regulation in the process of obtaining licenses for the construction and operation of NPPs, with periodic safety assessments of NPPs, while regulatory requirements are changed and based on the results of using NPP operating experience.

I-18.2. Regulatory framework for the use of atomic energy in the Russian Federation

The national policy of the Russian Federation to ensure the safety of nuclear power plants is based on the Constitution and federal laws, which establish the priority of state administration of nuclear energy.

The pyramid of Russian Federation documents regulating the rules for ensuring the safety of operating (and designed) NPPs is shown in Fig. I-18-1.

Currently in the Russian Federation there are more than 47 federal laws, 60 - Russian Federation Government resolutions, 100 federal norms and rules concerning use of atomic energy, 138 safety guides.

Continuous work is underway to improve the regulatory framework, its approval, harmonization with the publications of the IAEA, EUR, WENRA, and other countries. Only during last 5 years, from 2014 to the present, 139 safety guidelines have been developed and/or revised and enacted.

The state regulation of safety use of atomic energy is implemented by the Federal Service for Ecological, Technological and Nuclear Supervision (Rostekhnadzor). The regulation is under authority of Russian Federation Government and independent of government bodies (Rosatom, Rosenergoatom) responsible for use of atomic energy.

The regulatory body has approved and is currently implementing a plan for improving regulatory and safety regulation and standardization for atomic energy use, for the period from 2015 to 2023, aimed also at harmonization with the IAEA safety guides and procedures.

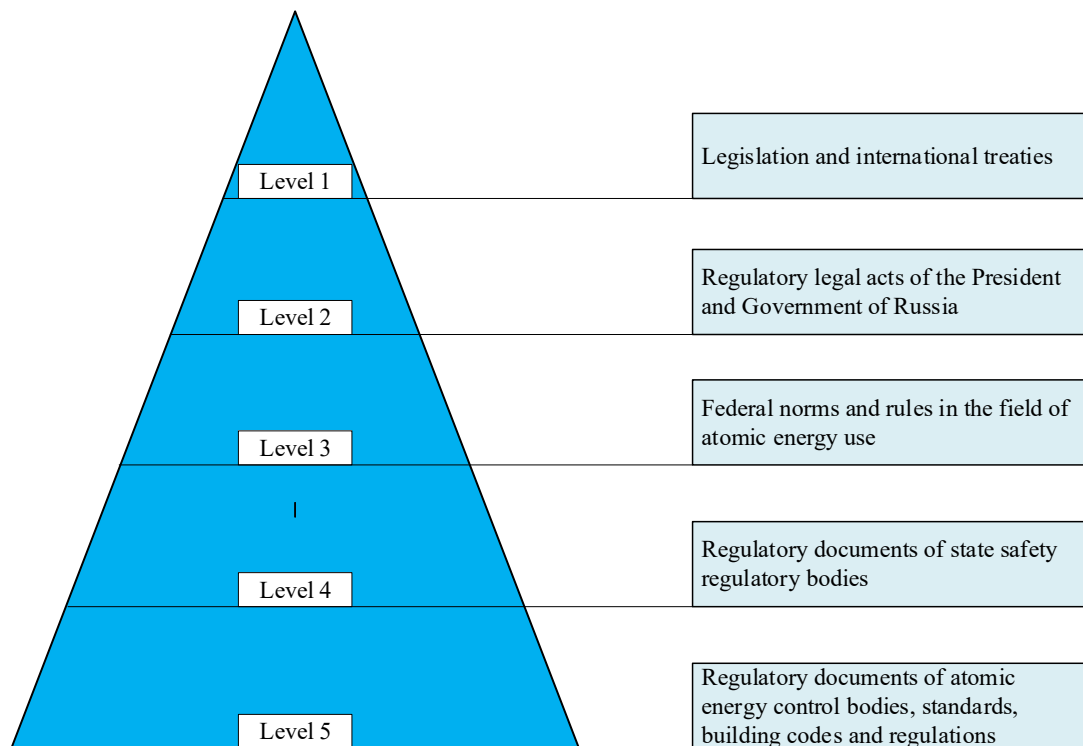


FIG. I-18-1. Pyramid of Russian Federation regulating documents establishing the rules to ensure the safety of operating (and designed) NPPs.

I-18.3. Harmonization of Russian regulating documents with IAEA safety standards

In 2015, a new edition of the federal rules and regulations “General Provisions for the Safety of Nuclear Power Plants” (NP-001-15) was approved. A comparative analysis of the current regulatory document requirements with the IAEA Safety Standards Nos SSR-2/1 (Rev. 1) [6] and SSR-2/2 (Rev. 1) [3] demonstrates that the Russian safety requirements for nuclear power plants generally comply with the requirements of the IAEA safety standards. At the same time, there were identified areas where Russian regulatory documents need to be updated to achieve more complete harmonization with the IAEA safety standards. [60]

In the new edition of the “General Provisions for the Safety of Nuclear Power Plants”:

- The concept of “safety of a nuclear power plant” has been brought into a line with the IAEA safety regulations of the SF-1 top level.
- The requirements to the scope and procedure of design and beyond design analysis basis accidents were formulated.

- The rules for classification of systems and components of NPPs were clarified, in particular taking into account the requirements of IAEA Safety Standards Series Nos SSR-2/1 (Rev. 1) [6] and SSG-30 [61].
- The formulation of the NPP probabilistic safety targets has been changed.
- In addition to the previously considered categories “systems and components of normal operation” and “systems and safety components”, the new category “special technical means to manage design basis accidents” was introduced. Also, there were introduced a number of requirements for the special technical means.
- When disclosing the notion of a safety culture, its interpretation in the line with INSAG-4 is currently used.
- The requirements for implementation of defence in depth at the nuclear power plant were clarified. In particular, the requirement has been introduced to take all reasonably achievable measures to ensure the mutually independence of defence in depth levels, as well as measures aimed at preventing depended damage of physical barriers due to damage of other one or several physical barriers by one impact.
- The concept of “decommissioning an NPP unit” was introduced.
- The concept of “safety management” was introduced (as provided for in SSR-2/1 (Rev. 1) [6]);
- The requirements for the analysis of operating experience were expanded - in particular, the concept of a “precursor of a severe accident” was introduced, and an additional procedure for considering events-precursors with a significant conditional probability of transition to a severe accident stage was indicated.

I-18.4. Examples of newly released or replaced with a new edition federal rules and regulations in the field of the use of atomic energy

Among the newly issued federal rules and regulations in the field of atomic energy are the following:

- “Requirements for managing the life of equipment and pipelines of nuclear power plants. The main provisions”(NP-096-15) [62];
- “Rules for the Construction and Safe Operation of Equipment and Pipelines of Nuclear Power Plants” (NP-089-15) [63];
- “Rules for control of the base metal, welded joints and weld surfaces during operation of equipment, pipelines and other components of nuclear power plants” (NP-084-15) [64];
- “Safety rules for handling radioactive waste from nuclear power plants” (NP-002-15) [65];
- “Basic requirements for substantiation of the strength and thermomechanical behaviour of fuel assemblies and fuel elements in the core of water-cooled power reactors” (NP-094-15) [66];
- “Basic requirements for probabilistic safety analysis” (NP-095-15) [67];
- “Regulations on the procedure for declaring an emergency, prompt information transfer and organization of emergency assistance to nuclear power plants in cases of radiation-hazardous situations” (NP-005-16) [68];
- “Requirements for the content of the safety justification report for a nuclear power plant with a VVER-type reactor” (NP-006-16) [69];
- “Basic requirements for extending the life of a nuclear power unit” (NP-017-2018) [70];
- “Rules for the Construction and Safe Operation of Cranes for Nuclear Facilities” (NP-043-18) [71];
- “Rules for the Construction and Safe Operation of Pressure Vessels for Nuclear Facilities” (NP-044-18) [72];

- “Rules for construction and safe operation of steam and hot water pipelines for nuclear facilities” (NP-045-18) [73];
- “Account of external, natural and human induced events at the nuclear facilities” (NP-064-17) [74].

I-18.5. Implementation of technical measures to eliminate safety deficiencies (the case of NPPs with VVER-1000 reactors of small series)

I-18.5.1. Grounds for the implementation of technical measures to address the safety deficiencies

A significant revision of regulatory requirements requires from the management of operating NPPs to organize procedures to perform analysis of non-compliance to new requirements, the development and implementation of safety measures corresponding to the current level of knowledge in nuclear energy science.

An example of the implementation of such measures are solutions to resolve the issues outlined in the IAEA document “Safety Issues and Categories of Power Plants with VVER-1000 Reactors of the ‘Small Series’” (IAE-EBR-WWER-15) in Novovoronezh and Kalinin NPPs.

The issues were resolved completely or partially in the framework of the works described in the following documents:

- “The programme of work to eliminate non-compliance with the requirements of regulatory documents on NPP safety” for Novovoronezh and Kalinin NPPs;
- “The measures to compensate non-compliance with the requirements of regulatory documents on NPP safety” for Novovoronezh and Kalinin NPPs;
- “The working programme to eliminate the commented issues and implement the proposals of the Expert opinions on the safety of operation of power units No. 5 of the Novovoronezh NPP and No. 1.2 of the Kalinin NPP”.

Some of the problems cited in the IAE-EBR-WWER-15 IAEA were resolved by adopting special decisions by Rosatom State Corporation and Rosenergoatom Concern JSC.

I-18.6. List of safety issues for small series VVER-1000 NPPs (IAEA-EBP-WWER-15)

TABLE I-18-1. LIST OF SAFETY ISSUES FOR NPPS WITH A SMALL SERIES VVER-1000

| Category | Name of problem |
|--------------------------------------|--|
| General | |
| II | Component classification |
| III | Equipment classification |
| II | Reliability analysis of 1 and 2 safety class systems |
| Reactor core | |
| II | Prevention of uncontrolled dilution of boron solution |
| II | The reliability of the CPS (Control and protection system) control rods insertion / deformation of the fuel assemblies |
| I | Subcriticality control at the shutdown reactor |
| Component integrity | |
| III | Metal embrittlement of the reactor vessel and its control |
| III | Non-destructing tests (NDT) |
| II | Restriction of pipe whip in the primary circuit |
| III | Integrity of steam generator collectors |
| II | Integrity of steam generator heat exchanger tubes |
| III | Integrity of steam line and feed water line |
| II | Structural integrity controls |
| Systems | |
| II | Primary circuit protection against cold overpressure |
| II | Limiting the effects of a vessel rupture |
| I | Main coolant pump (MCP) sealing water system |
| II | Certification of pressurizer safety and discharge valves to the water flow |
| III | ECCS Sump Blockage |
| I | Integrity of the ECCS Suction Lines |
| II | Integrity of ECCS Heat Exchanger |
| I | Valves with electric actuator on the ECCS (Emergency core cooling system) injection lines |
| II | Certification of steam generator safety and dump valves to the water flow |
| I | The operation of the steam generator safety valve at low pressure |
| I | Steam Generator Level Control Valves |
| II | HVAC system for control panels (Main control room, reserve control room) |
| II | Hydrogen removal system |
| III (NV NPP) | The performance of the boron injection system |
| I (Kalinin NPP) | |
| III | Disadvantages of an emergency feedwater system |
| III | Physical separation and functional isolation of ECCS |
| II | Limited reserve of boric acid for high pressure injection |
| Control and management system | |
| II | C&I equipment reliability |
| I | Concept layout of safety system automation |
| I | Automatic protection of the reactor in case of power disturbance and heat transfer crisis |
| II | Ergonomics of control panels |
| III | The reactor protection system redundancies |
| I | Monitoring of mechanical equipment |
| II | Primary Circuit Diagnostic Systems |
| II | Accident Monitoring Measurement Instruments |
| II | Technical Support Centre |
| I | Equipment for the control and monitoring of chemical water content (primary and secondary circuits) |
| II | Separation of selections to instrumentation and automation sensors in the measuring channels of the primary circuit |
| I | Overlapping ranges of the neutron flux monitoring system |

| | |
|-------------------------------|--|
| Power supply | |
| I | Reliability of diesel generators |
| I | Protection Signals for Diesel Generators |
| II | Internal power supply for incident and accident management |
| III | Emergency battery discharge time |
| II | Grounding faults in DC circuits |
| Containment | |
| II | Containment bypass |
| Internal hazards | |
| II | Systematic fire safety analysis |
| III | Fire Prevention |
| II | Fire detection and suppression |
| II | Flame exposure limitations |
| I | Systematic flooding analysis |
| II | Protection against flooding of emergency power distribution boards |
| II | Protection against dynamic effects caused by rupture of the main steam and feedwater pipelines |
| II | Locking of the polar crane |
| External hazards | |
| II | Seismic design resistance |
| I | Natural external hazards for the NPP site |
| II | External man induced events |
| Accident analysis | |
| II | The scope and methodology of accident analysis |
| I | Quality assurance of the plant data used in accident analysis |
| I | Certification of computer codes and NPP models |
| I | Use of accident analysis results to support NPP operation |
| I | Analysis of the accident with the main steam line rupture |
| II | Transient modes with chilling associated with thermal shock under pressure |
| II | The steam generator collector mechanical damage failure analysis |
| II | Low power and when the shutdown reactor accidents |
| I | Severe accidents |
| I | Probabilistic Safety Analysis (PSA) |
| I | Boron dilution crashes |
| II | Transients without triggering reactor emergency protection (ATWS) |
| II | Full blackout |
| II | Complete loss of heat sink |
| Operating Instructions | |
| | Normal operating instructions |
| | Emergency operating instructions |
| | Instructions for severe accident conditions |
| | Restrictions (limits) and conditions |
| Control | |
| | The need to improve safety culture |
| | Accounting for operating experience |
| | Quality assurance program |
| | Data management and workflow |
| NPP operation | |
| | Philosophy of application instructions |
| | Survey programme (operational supervision) |
| | Communication system |
| Radiation protection | |
| | Radiation Protection and Monitoring |
| Exercise | |
| | Studying programs |
| Emergency planning | |
| | Crisis centre |

I-18.7. Examples of the implementation of technical measures to address safety deficiencies at the Kalinin and Novovoronezh NPPs

A full description of how the safety deficiencies were eliminated at the power unit No. 5 of the Novovoronezh NPP and the power units No. 1, 2 of the Kalinin NPP occupies more than 115 pages of text, therefore the following are just some examples of implemented measures.

I-18.7.1. Issue: Lack of reliability analysis of systems belonging to class 1 and 2 on safety.

The reliability analyses of classes 1 and 2 systems are necessary to confirm the compliance of the reliability indicators to the indicators set in the design documentation. It is also important to collect data on the reliability of the components during operation and to confirm the results of the initial analysis. Reliability analyses for class 1 and 2 systems are to be performed taking into account common cause failures and personnel errors. The results of the analysis are used to perform the PSA.

To eliminate the safety issue at the both plants, an equipment reliability database has been created that has been used since 1990. A quantitative reliability assessment for class 1 and 2 systems using specific data on the reliability of the components and equipment of power units.

As the result the safety issues at all three units were eliminated.

In addition, a number of documents have been developed to prevent the occurrence of this type of safety issue. These include RB-100-15 "Recommendations on the order of performing an analysis of the reliability of systems and components of nuclear power plants important to safety and their functions", NP-006-16 "Requirements for the content of the safety analysis report for a nuclear power plant with WWER reactors". These documents describe in detail the requirements for conducting such analyses, including methodologies for calculating reliability indicators.

I-18.7.2. Issue: Control of subcriticality at the shutdown reactor

A reserve of subcriticality is sufficient for all states of shutdown reactor and relevant information about the reserve needs to be available to operators. In the "small series" VVER-1000, subcriticality is controlled by measuring the concentration of boric acid and neutron flux.

The neutron flux control is difficult at the stages of emergency shutdown and power output, because the neutron flux in the ionization chambers is very low ($10^4 - 10^3$ n/cm² s) and it cannot be reliably registered with the existing equipment (AKNP-3).

To eliminate the safety issue, AKNP-3 was replaced with AKNP-07-02, which has greater sensitivity, reliability and performance.

The original NAR-B boron meter was replaced with a new device having higher reliability indices and less time lag to be able to measure the content of B-10 with an alarm.

As the result the safety issues were resolved.

Issue: Integrity of the Steam generator (SG) collector

In the period from the end of 1986 to 1991, cracks were discovered between the sections of the pipes in the 24 SG manifolds on 6 VVER-1000 units. The operating time before the detection of defects ranged between 7000 and 60 000 hours. Cracks were mostly found in cold collectors, but indications were also noted in hot collectors. The material of the collectors is carbon steel clad on the side of the primary circuit. The damage was the result of a combination of high residual stresses associated with manufacturing technology, cracking under the influence of working environment parameters and due to violation of the requirements to the water-chemical regime of the secondary circuit caused by poor quality collector material (non-metallic inclusions).

In the future, to ensure the quality of the chemical composition of water in secondary circuit, it will be necessary to take into account the combination of materials in secondary circuit that leads to conflict of requirements and compromise solutions. This can also lead to accelerated wear of SG components. There was also a re-injection of chlorides into the secondary circuit through damaged condenser tubes. The existing control of the chemical composition of water, leaks in I and II circuits and the prevention of deterioration of the properties of components are unsatisfactory from the point of view of not exceeding the design limits of safe operation.

On all SGs of operating NPPs, compensatory temporary measures were applied, which proved to be successful, with the exception of one failure at Balakovo NPP in 1995. Further careful monitoring of SG integrity is necessary.

To eliminate safety deficiencies at unit 5 of the Novovoronezh NPP:

After the discovery of cracks in the same SG vessel, all 4 SGs were replaced in 1989 with PGV-1000M using some corrective measures. In 1996, an additional heat treatment of the collectors was carried out. After replacing the SG, a stricter control was established for the chemical composition of water in secondary circuit. Collectors are checked in accordance with the recommendations of the design organization. Annually tests of collectors are carried out by eddy currents. After replacing the SG cracks were found.

The safety deficiency was reduced.

To eliminate safety deficiencies at units 1 and 2 of the Kalinin NPP:

There were no cracks in the vessels and the plant is working with the original SGs. Several corrective measures have been taken (for example, additional heat treatment). Collectors are inspected in accordance with the recommendations of the design organization. Automatic chemical control of water was applied.

In order to early detect the possible defects of the SG collector link pipe runs at the Kalinin NPP, monitoring of the respective collector areas by non-destructive methods is carried out at intervals and in amounts determined by the General Designer in the steam generators manual.

The safety deficiency now is reduced.

I-18.7.3. Issue: ECCS sump grid clogging

Unlike the serial modification of the VVER-1000/320, the VVER-1000 of the modifications 187, 302 and 338 have three separate containment sumps. The basemat openings in the containment are covered with metal grids that are designed to prevent fragments from entering to the inlet of the ECCS and containment spray system.

The equipment and piping of the primary and secondary circuits inside the containment are covered with fibre thermo-insulation. The insulation inside the containment and the limited area of the grids over the sumps forms a combination that leads to a safety problem regarding to the ability to maintain ECCS circulation after medium or large LOCA accidents. Operational experience, based on similar events in Sweden and the USA, showed that even a relatively small amount of such fibre could significantly clog large areas of the grid. In addition, tests at the Zaporozhe NPP have shown that small amounts of fibrous material can clog ECCS heat exchangers.

The effect of clogging of the sump grids depends on many factors, such as the type of thermo-insulation, the size and placement of the sump grid and the mechanism of transfer of this material to the sump.

With the elimination of safety deficiencies in all three power sumps, the ECCS sumps in the containment were upgraded. A new filtering system was installed with a filtration area that repeatedly exceeds the area of the previous system. The characteristic of the system is confirmed by the tests.

I-18.7.4. Issue: Technical Support Centre

Current world practice is related to the design of NPPs with special facilities where up-to-date information on the state of the plant is collected at the video terminal, enabling technical experts to assist operators in the accident management process. This room is separated from the control room. VVER-1000 nuclear power plants did not have technical support centres.

To resolve the safety issue an emergency technical centre was created at NV NPP and in the city of Novovoronezh. The emergency centre is equipped with a computer system that is able to predict the development of the accident and the radiation level in various locations inside and outside the plant, as well as an air purification system and a telephone line connection. Safety parameters detection system (SPDS) was implemented.

There are special rooms at Kalinin NPP and in the town of Udomlya, where the current information on the plant status is collected on the video terminal, enabling technical experts to provide support to operators in the accident management process. These rooms are located in the plant and city emergency technical centres.

The safety issue was resolved.

I-18.8. Conclusions

By analysing deviations from the requirements of regulatory and technical documentation that was developed or updated after the development of design documentation and construction of unit No. 5 of the Novovoronezh NPP and units No. 1 and 2 of Kalinin NPP, 87 items of work (safety issues) were identified in 16 areas.

To resolve safety issues, appropriate work programs have been developed.

Then, for all NPPs with the VVER-1000 "small series", a large complex of works was carried out to eliminate safety issues, listed in Table I-21. At the same time, the work to improve the safety level has been not limited and is not limited to this list.

As a result of this work, safety issues either have been completely resolved or substantially reduced.

Work in this direction is underway at all Russian NPPs in operation.

I-18.9. Methodology of assessment of the current level of NPP unit safety by means of stress tests used in Russia

I-18.9.1. Organization of work to improve the safety of existing nuclear power plants based on the results of stress tests

In General, the following organization of works to improve the safety of existing nuclear power plants based on the results of stress tests is used:

- The implementation of stress tests;
- The definition of safety deficiencies;
- Development of an action plan to improve safety;
- Procurement of necessary equipment;
- Development of design and estimate documentation;
- Implementation of additional activities;
- Assessment of the achieved safety level on the basis of deterministic and probabilistic analyses.

I-18.9.2. Objectives of stress tests for Russian nuclear power plants

The objectives of stress tests are:

- Additional evaluation of the adequacy of the design of technical solutions, the efficiency of the systems safety, reliability of defence in depth levels, to ensure NPP safety in case of emergency external influences, provided that project;
- Assessment of efficiency and sufficiency of technical means and preventive nature organizational measures of provided at the NPP site in case of threat of appearance of extreme external events;
- Assessment of efficiency and sufficiency of technical means and organizational measures provided at the NPP site for management of severe beyond design basis accidents and mitigation of their consequences;
- Assessment of NPP safety for the extreme external event with intensity exceeding the NPP design margin values.

I-18.9.3. The scope of the stress tests

Stress tests are performed in the following scope:

- For all nuclear power plants in operation in Russia.
- For all potential extreme external events on nuclear power plants, characteristic for the area of its location:
 - natural events (earthquakes, floods, tornadoes));
 - man induced external events (forest fires, dam breaks and dams, the sharp fluctuations in the external electric grid);
 - for all various combinations of extreme external hazards on nuclear power plants.

During the "stress tests" the consequences are evaluated:

- The loss of power supply nuclear power plant, including a complete blackout;
- The loss of the ultimate heat sinks, ensuring the residual heat removal from the reactor, fuel pools and spent fuel storage;
- The loss of integrity (tightness) of the reactor containment;
- The event combination of the loss of power of the NPP, loss of ultimate heat sinks and loss of integrity of the reactor containment.

The main aspects covered in the reports on stress tests

The reports on the results of the “stress tests” reflect the following aspects:

- Design requirements for NPP safety with respect to external events;
- Basic design decisions to account external events;
- The list of organizational measures and preventive nature technical means provided at the NPP site in case of the potential threat from extreme external hazard;
- Compliance of the NPP current state to its design requirements;
- The results of effectiveness and adequacy evaluation of the site technical and organizational measures envisaged at the NPP to manage severe beyond design basis accidents and mitigation of their consequences in case of extreme external events, including the potential radiological consequences of accidents;
- The results of NPP safety assessment for the external hazards with intensity exceeding the design requirements, including the potential radiological consequences of the accidents;
- The list of additional technical means and organizational measures to ensure NPP safety and accident management in case of extreme external events exceeding design requirements to keep:
 - The basic safety functions (reactivity control, fuel cooling, radioactivity retention);
 - The emergency power supply of nuclear power plants;
 - the final heat sinks (for removal of residual heat from reactor core, fuel pool and spent fuel storage);
 - Integrity of the containment;
 - Schedule for the implementation of recommendations developed as a result of stress tests, including the installation of additional technical means and the implementation of organizational measures to ensure the safety of nuclear power plants and the management of accidents in the event of extreme external events.

The information in the stress tests reports

The reports on the implementation of stress tests:

- Contain information (with reference to the PSAR, FSAR, PSA) on the parameters of all natural and human induced events parameters for the NPPNPP site. For human-induced factors, specific sources are indicated. The information about the sources of human-induced hazards, located both outside and on -site the nuclear power plant is provided.
- Set out the conclusion about completeness of the design account of external events and on the validity of the adopted values of the external events.
- Substantiate of account (or not account) in the analysis of external events, characteristic or not characteristic for the site and the location of the NPP.
- Describe the approaches taken in the analysis of the natural event combinations.
- Describe scenarios of the analysed accidents (taking into account the fact that the external event affects several units of the multi-unit NPPs).
- Assess the need to ensure the cooling of the fuel in the fuel pool and spent fuel storage;
- Provide the analysis of external event combinations.
- Describe the sources of human induced hazards at NPP site (sources of flooding, fires, emissions of toxic substances).
- Provide the analysis results related to earthquakes impacts on:
 - Beyond design accident (BDA) management;
 - NPP components, whose failure can cause damage of safety related components and result to fire consequences;
 - The transportation routes needed for accident management.
- Discuss the qualification of equipment to operate under external events.
- Provide the results of the analysis of the sufficiency of the existing requirements of documentation on the personnel actions in the conditions of external events.
- Define the timing and stages of severe accident development, describe the strategy of actions at each stage before transition of accident to a severe stage and after such transition.
- Analyse the adequacy of:
 - Accident mitigation documents;
 - NPP staff (number and degree of qualification);
 - Information on the NPP status received by the staff;
 - Technical means to control the beyond design accident, their accessibility and efficiency in specific accident scenario.

Systems and components resistance analysis to external events

The resistance analysis of systems and components to the external events:

- Provides information (with reference to the available justification) on the resistance of systems and components (including buildings and structures) important for safety to external events. Special attention is paid to the equipment located outside the buildings.
- Defines systems and components for which resistance to external events is not confirmed.
- Provides information on the resistance of available technical means on management of beyond design basis accidents to the analysed external events (with reference to the available justifications).

- Analyses the impact of external hazards to the components (including buildings), in which damage induced by external hazard could:
 - Damage the safety components;
 - Cause fire consequences (explosion, release of toxic media).
- Presents estimates of the external impacts to transport communications at the NPP site NPP (used when performing accident management actions) and outside NPP site (to be used to provide emergency assistance to the NPP plant), impact to the personnel location rooms (first of all control panels).

Description of the consequences of the analysed external events

For each analysed external event, the following are described:

- Dependent effects (fire due to plane crash, fire due to earthquake);
- The impact of external events on safety - critical components (function or loss of function);
- The impact to the technical means used for the management of the beyond design accident, and to the transport routes used in the accident management;
- The impact of non-safety related equipment failure to the safety components;
- The impact to personnel (control boards, traffic routes).

Analysis of severe accident management processes

Analysis the processes of severe accident management:

- Identifies typical stages of development of the accident (from the start of fuel rods cool-ant heat up, damage of fuel and further subsequent stage of a severe accident) in the event of complete lack of the accident management measures for from the staff;
- The separate plant states are selected, where:
 - The state of physical barriers;
 - The values of the parameters of the reactor pressure vessel;
 - Spent fuel storage, environment parameters under the containment;
 - The state of safety systems, as well as systems used for accident management;

All require the implementation of the beyond-design basis accident management strategy (including the objectives of the management of the BDA, ways of their implementation, the list of available technical means and capabilities for the actions of the NPP personnel), different from the accident management strategy at other states.

- For each of the selected states are specified:
 - The state of physical barriers, safety systems, means of management of BDA;
 - The strategy of accident management, the availability of relevant emergency response documentation (refer to Ref. section), no conflict in the requirements of the documentation;
 - Sufficiency of information received by NPP personnel to act in accordance with emergency documentation;
 - Availability of technical tools for the management of pad, the adequacy of technical resources and personnel as (given the fact that pad affects all NPP units), access routes and transport communications, communications and alerts;
 - Assessment of the feasibility of effective accident management actions.

Safety improvement action plans

The safety improvement plans:

- Identify safety deficiencies;
- Contain measures to improve safety:
 - Additional analyses, organizational arrangements (including development of necessary documentation and guidelines);
 - Technical measures to improve the NPP protection against external hazards;
 - Recommendations for the introduction of new or improvement of existing technology.

The proposed activities are accompanied by an indication of which safety deficiencies identified by stress test results were eliminated and its extent and what is the expected effect for each of the planned activities (or their combination) for the safety of the NPP. Justification for the adequacy of the proposed activities are provided.

The measures implemented at Russian NPPs in the light of lessons learned from the accident at NPP "Fukushima-Daiichi", including the activities undertaken or planned in order to withstand natural hazards, which exceed the design basis

The map of Russian nuclear power plants (existing, under construction and future) is shown in Fig. I-18-2.

The accident at the 2011 Fukushima Daiichi nuclear power plant had a significant impact on the maintenance of existing Russian nuclear power plants. Already in 2011, "stress tests "were conducted, including:

- Implementation of target reassessment of safety limits of NPP projects under extreme external natural impacts;
- Confirmation of sustainability of projects nuclear power plants in conditions of the extreme;
- Development of additional technical and organizational solutions aimed at improving the stability of nuclear power plants under extreme conditions.

Stress tests were performed not only for existing, but also for under construction, and for designed nuclear power plants.

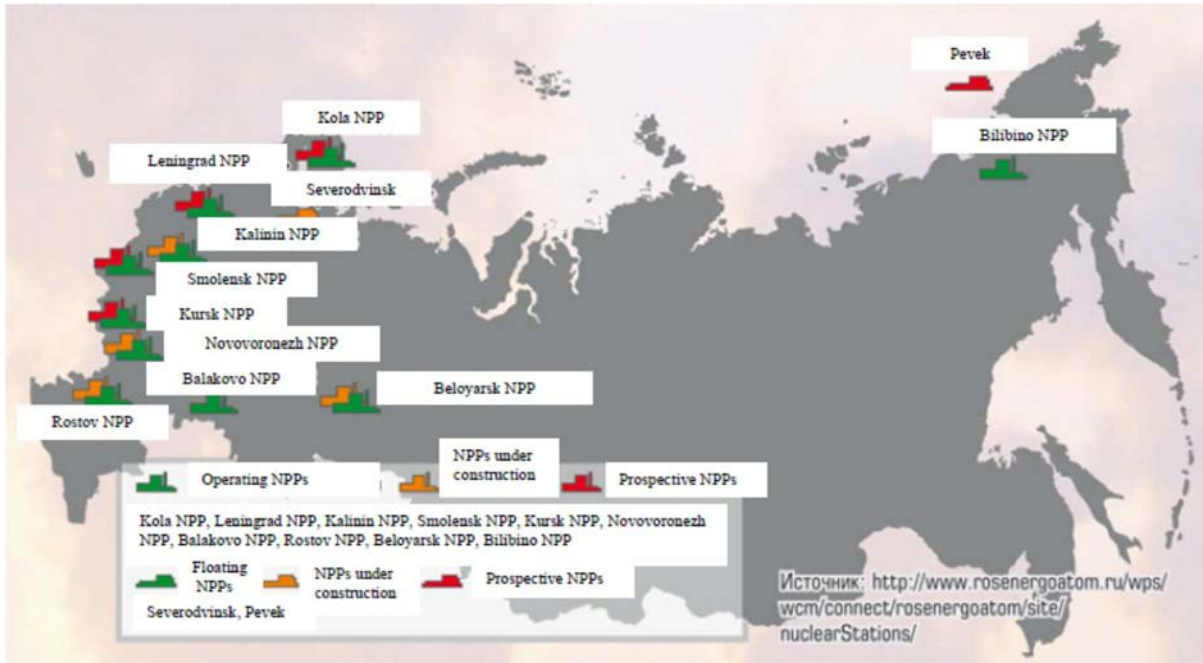


FIG. I-18-2. Russian NPP (existing, under construction and design).

Based on the results of the stress tests, "Reports on the safety analysis of nuclear power plants under extreme external events" were issued. In 2011, "Measures to reduce the consequences of beyond-design basis accidents at nuclear power plants" were developed and implemented.

These activities are updated annually taking into account the current state of NPP, obtaining additional information on the operating experience and the results of benchmarking of other NPP operating organizations.

The implemented measures are aimed to ensure the following types of safety of all operating and constructed nuclear power plants:

Water supply:

- Equipment of power units with mobile pumping equipment;
- Development and implementation of pumping equipment connection projects;
- Creation of reserve water storage tanks, use of fire water pipelines.

Energy supply:

- Equipment of power units with mobile diesel generator sets;
- Development and implementation of design to connect the mobile diesel generator.

Additional analysis of BDA beyond design accident:

- Accident analysis;
- Deterministic and probabilistic safety analysis taking into account external influences (DSA, PSA);
- Adjustment of safety reports;
- Correction of lists of beyond design basis accidents (BDA beyond design accident));
- Development of methods for assessing the parameters of ice formation (water supply in winter).

Hydrogen explosion safety:

- Analysis of a hydrogen explosion;
- Equipment procurement;
- Project development.

Accident gas discharge, accident sampling:

- Design development;
- Equipment purchase;
- Implementation.

Accident complementary C&I:

- Design development;
- Procurement of components;
- Implementation.

Seismic safety:

- Additional seismic microzonation at NPP sites;
- Verification of seismic resistance;
- Increase of seismic resistance of equipment;
- Implementation of seismic protection (where it required).

Accident management documentation:

- Implementation of operational staff action cards;
- Adjustment and development of new instructions and manuals;
- Development of guidelines for severe accident management (RUTA).

Emergency response. Creation of:

- Digital radio communication system of "TET-RA" standard» at nuclear power plants and crisis centres (CC);
- Radio soft and hardware complexes at NPPs and in the crisis centres;
- Mobile control points and mobile communication units at nuclear power plants;
- Regional Crisis Centre of WANO – Moscow Centre.

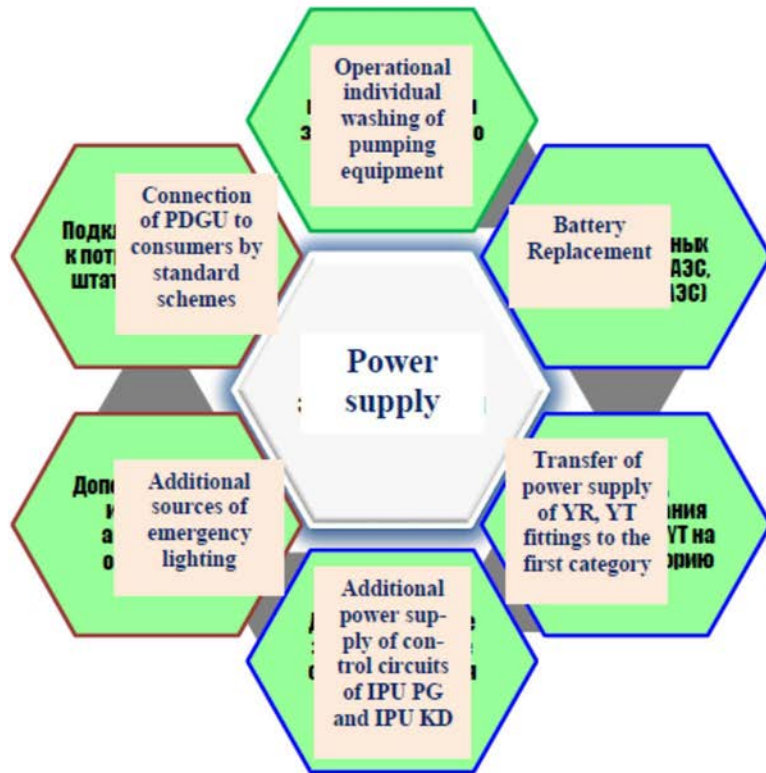
Equipment special equipment: cranes; trucks; fuel trucks; bulldozers; road scrapers.

- Positive practice;
- Implementation of a set of mobile emergency equipment on each NPP unit and implementation of design solutions ensuring its application within 1.5–2 hours;
- The scale of comprehensive emergency response exercises held annually at one of the NPP, with the participation of international observers.

The results of the measures developed on the basis of stress tests and implemented at Russian nuclear power plants are presented in Figs I–18-3 to I–18-9.



FIG. I-18-3. Additional technical equipment supplied to Russian NPPs.



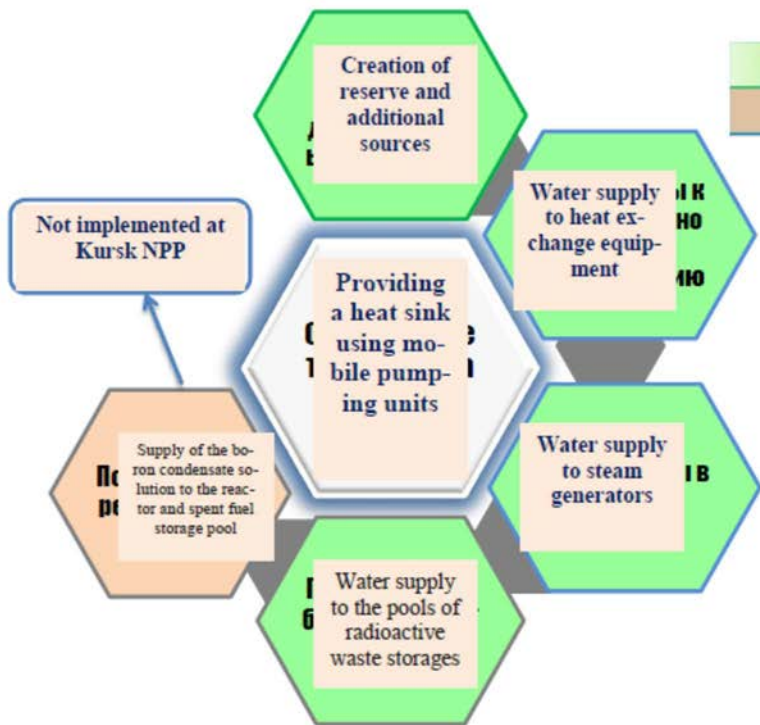
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Cabinets connecting PDGU-2.0 and PDGU-0.2



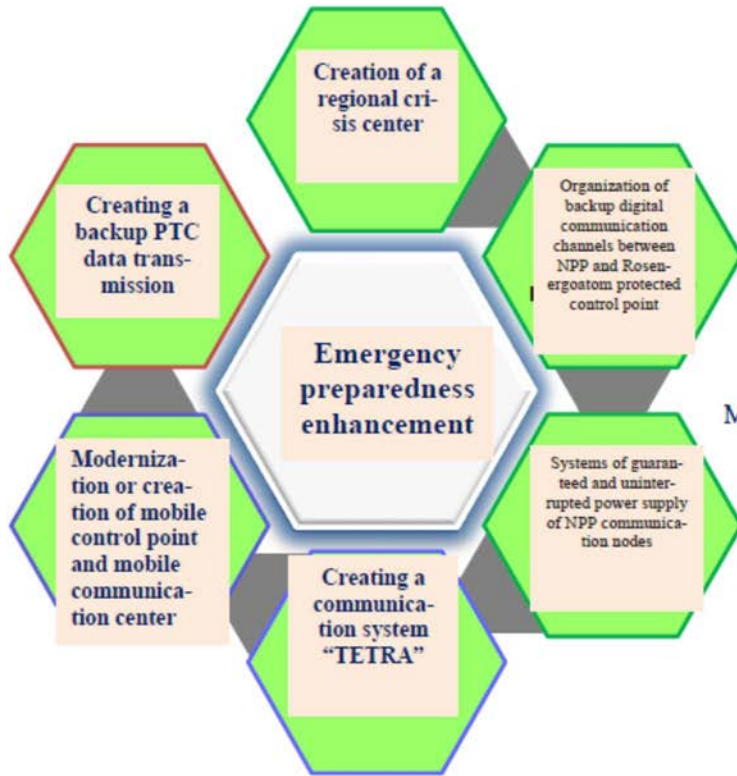
FIG. I-18-4. Measures to ensure the power supply of Russian NPPs.



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Completion of works in



FIG. I-18-5. Heat sink measures.



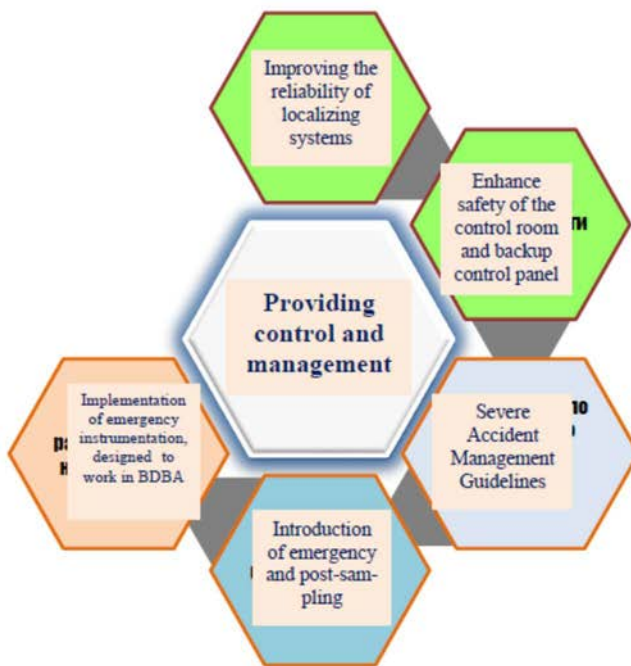
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Mobile communication center and control point



FIG. I-18-6. Emergency preparedness measures.



- Done
- Severe accident management guidelines: have been developed for WWER NPPs. For EGP and BN reactors there were developed the sections in BDBA accident procedures
- Introduction of a mobile module at a pilot NPP in 2019
- Introduction of emergency instrumentation at nuclear power plants in 2019-2021

Sampling and Dilution Module



FIG. I-18-7. Measures to ensure control and management.

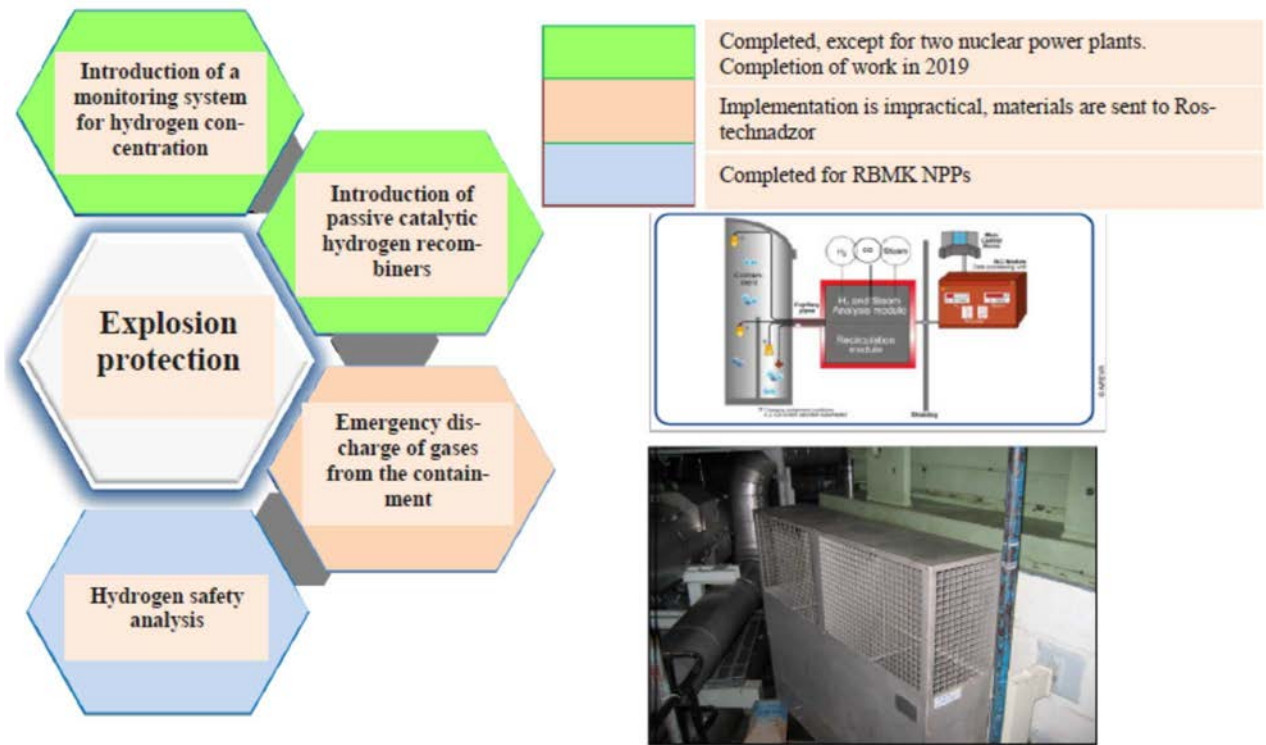


FIG. I-18-8. Measures to ensure explosion safety.

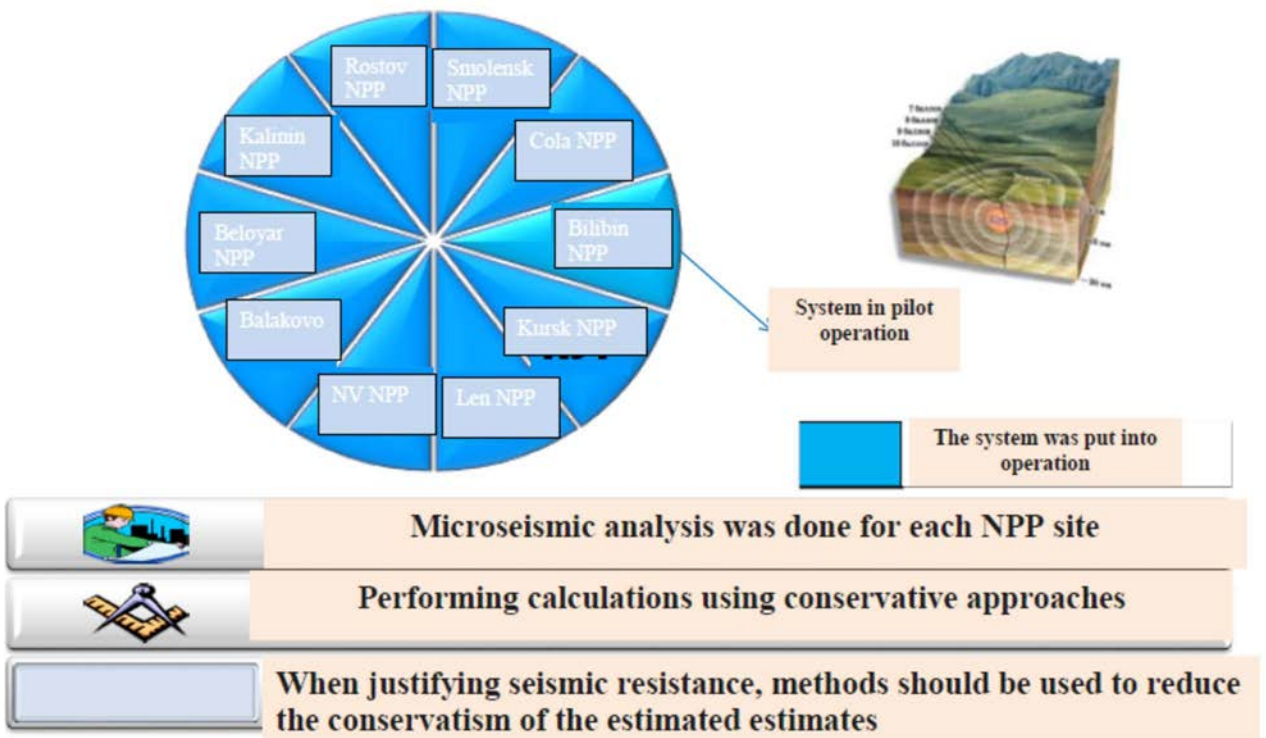


FIG. I-18-9. Implementation of seismic protection systems at nuclear power plants.

I-19. SPAIN

I-19.1. Introduction

Regarding the development of Spanish nuclear power plants (NPPs), the first generation NPPs started in the 1960s, with the construction of José Cabrera, Santa María de Garoña and Vandellós I. The second generation NPPs began in the early 1970s, with Almaraz I and II, Lemóniz I and II, Ascó I and II and Cofrentes. In the early 1980s, the construction of the NPPs Valdecaballeros I and II, Vandellós II and Trillo I started, and the preparatory studies for Trillo II were initiated. However, some of the projects and construction of Lemóniz, Valdecaballeros and Trillo II were halted during the 1980s.

At present, Spain has seven power reactors in operation in five sites: Almaraz I and II (PWR-3L-Westinghouse), Ascó I and II (PWR-3L-Westinghouse), Cofrentes (BWR-6-GE), Trillo (PWR-3L-KWU) and Vandellós II (PWR-3L-Westinghouse).

Three other power reactors have already been shut down: José Cabrera, Vandellós I and Santa María de Garoña.

The main entities and organizations with powers and responsibilities regarding nuclear power are the Nuclear Safety Council (CSN) and the Government through the Ministry for the Ecological Transition (MITECO). Figure I-19-1 shows the current institutional framework of nuclear energy in Spain.

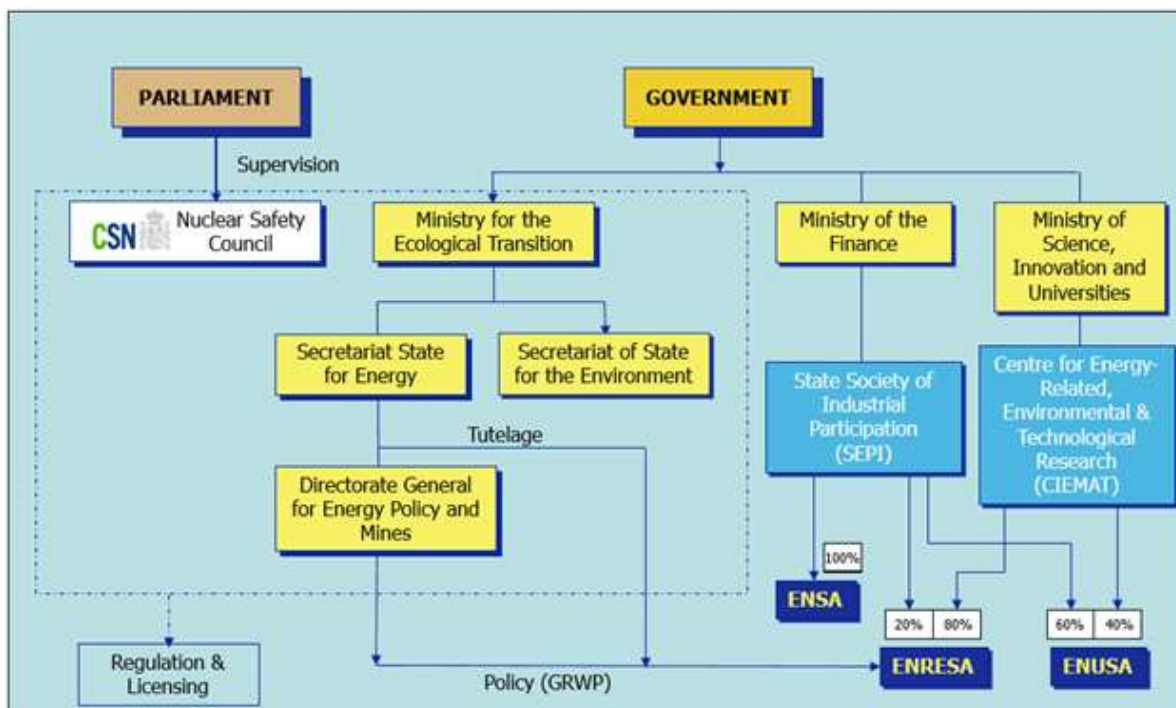


FIG. I-19-1. Organizational structure of nuclear power in Spain. [75]

I-19.2. Regulatory Framework

In Spain, two authorities undertake the regulatory function in nuclear matters: the Government, through the MITECO, and the CSN.

The Government is in charge of energy policy and of issuing binding regulatory standards. Specifically, MITECO is the Department of the General State Administration responsible for nuclear energy. The main tasks and duties of MITECO regarding nuclear energy include:

- Dictating norms and rules and proposing a radioactive waste management policy;
- Granting licenses for nuclear and radioactive installations, transporting radioactive materials and for the trade and commerce of nuclear materials;
- Suspending permits, in some specific cases, and sanctioning legal transgressions;
- Following up on the compliance of international commitments, such as non-proliferation, physical protection or civil liability;
- Managing the administrative registers on nuclear items.

The CSN is the sole organization competent in nuclear safety and radiological protection matters. It is governed by public law and by its charter. It is independent from the central Government and has its own legal personality and its own assets. The CSN's mission is to protect employees, the population at large and the environment from the harmful effects of ionizing radiation. It accomplishes this by ensuring that nuclear and radioactive facilities are operated safely and by establishing the preventive and corrective measures to apply in all radiological emergencies, no matter their source.

The main tasks of the Council are:

- To issue the required safety reports, prior to authorization by MITECO;
- To carry out all inspections with the capability to suspend activity in case of risk;
- To issue regulations concerning nuclear safety and radiological protection;
- To propose to the Government regulations concerning nuclear safety and radiological protection;
- To propose to MITECO sanctions in matters of nuclear safety and radiation protection;
- To grant licenses for operators of nuclear and radioactive installations;
- To inform the public about subjects of its competence;
- To report every year to Parliament about its activities.

Fig. I-19-2 shows the nuclear installation licensing process in Spain.

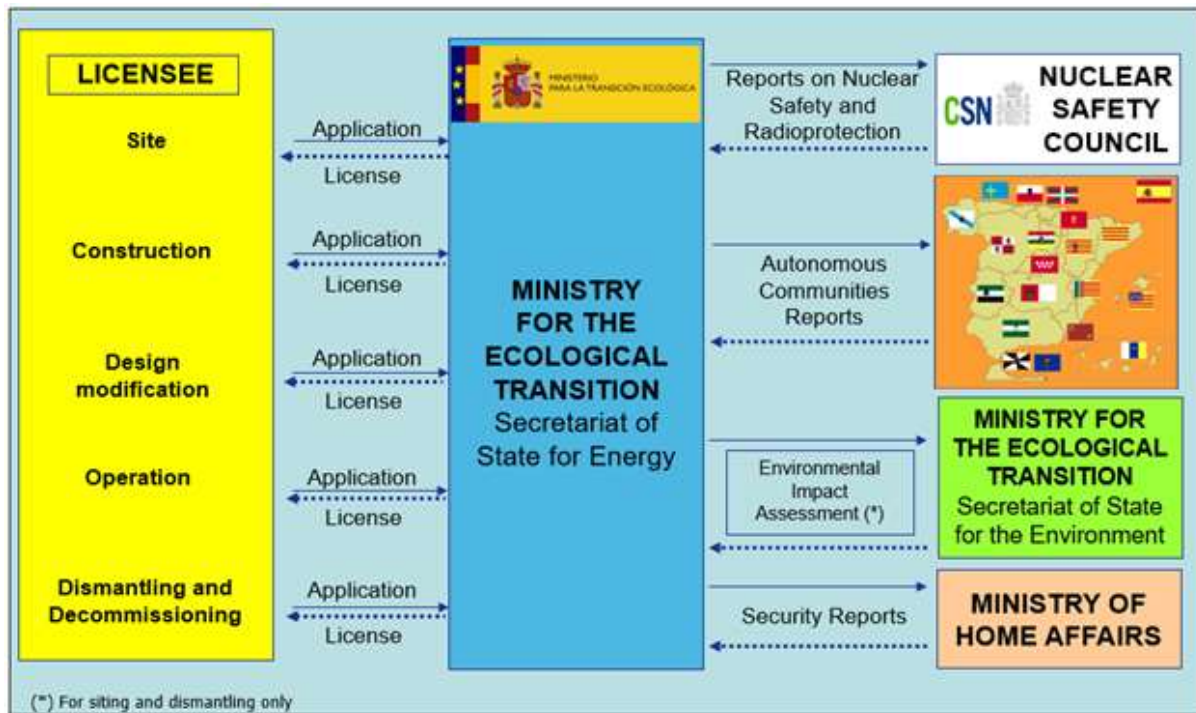


FIG. I-19-2. Licensing process of nuclear installations in Spain. [75]

The structure of Spanish nuclear safety regulations is shown in Fig. I-19-3. This national regulatory scheme legally support the capacity to enable the CSN to effectively driving safety improvements in the NPPs when considered necessary.

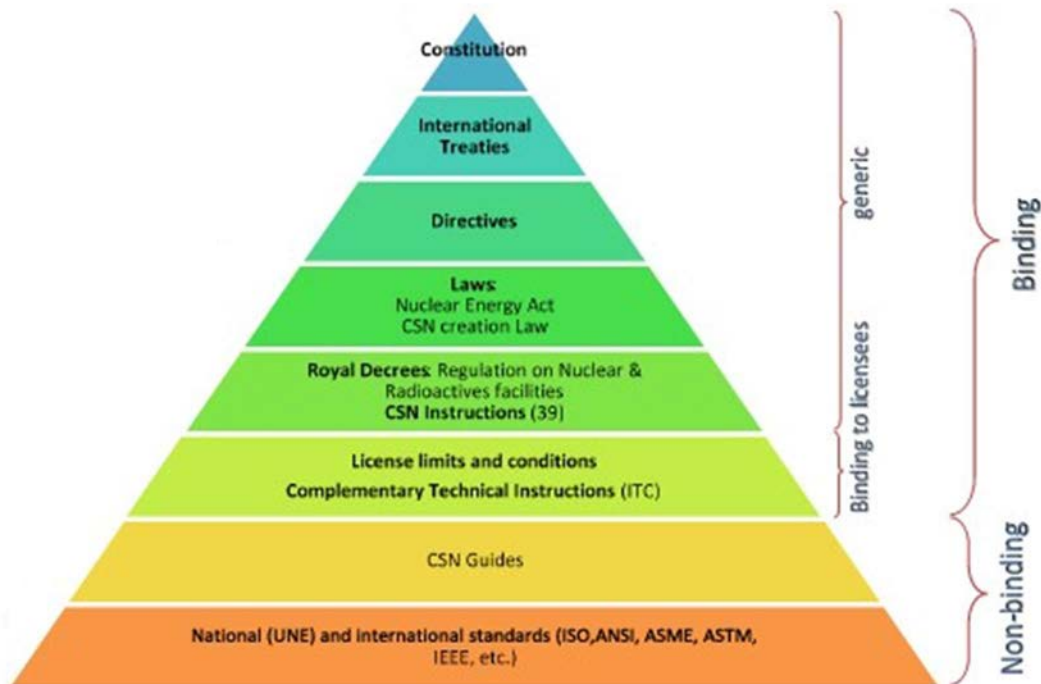


FIG. I-19-3. The nuclear regulatory pyramid in Spain. [76]

I-19.3. Fundamental Safety objectives and expectations of the Nuclear Safety Authority

Fundamental safety objectives and expectations are contained in existing national laws, mainly in the “Nuclear Energy Law” and in the Royal Decrees for “Regulation of Nuclear and Radioactive Facilities” and for “Nuclear Safety in Nuclear Facilities” (this last Decree, recently approved by the government).

More specific objectives and expectations (eventually developed as requirements) are also expressed in the prologues of the CSN (Spanish Regulatory Body) regulatory instruments, mainly the Safety Instructions (IS) and, sometimes, in the prologues of Technical Instructions (IT), Complementary Technical Instructions (ITC) or letters. All of these are binding technical standards necessarily to be adhered to by the addressees that are integrated into the legal system. Two of these IS (IS-26: Nuclear Safety Basic Requirements applicable to Nuclear Facilities) and (IS-27: NPPs General Design Criteria) include more generic requirements that can be understood as fundamental safety objectives.

Additionally, some expectations are included in The Nuclear Safety Council Safety Guides (GS). The content of guides is not mandatory, except in those cases in which a legal provision gives them compulsory status. Their objective is to achieve better compliance with regulatory forecasts and precepts, guiding the administrated entity in adequate decision-making rather than imposing solutions.

The CSN possesses a structured set of technical standards relating to the design, construction and operation of nuclear facilities that contemplates the principles of defence in depth. The specific requirements and criteria related to the operation of NPP are expressly contemplated in CSN Instructions (IS), Technical Instructions (IT), Complementary Technical Instructions (ITC) and letters, which include both the Spanish practices previously applied and the standards in force of the international organizations to which the Spanish State belongs (IAEA standards) and the standards available in the country of origin of the technology of each facility (USA and Germany), as well as the WENRA reference levels published in 2008.

I-19.4. Implementation of the Vienna Declaration on Nuclear Safety (VDNS)

On 9 February 2015, the Contracting Parties adopted INFCIRC 872, “Vienna Declaration on Nuclear Safety”, which is a commitment to certain principles to guide them in the implementation of the CNS objective to prevent accidents and mitigate their radiological consequences, if they occur.

The second principle of the VDNS is:

Comprehensive and systematic safety assessments are 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 above objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner. [77]

Spain address the application of the principles and safety objectives of the Vienna Declaration to existing NPPs in the following way:

- Implementing CSN IS-37 2015 on the DBA analysis at NPPs. This Instruction contributes to compliance with Council Directive 2014/87/EURATOM that requires that the national legal framework demand that the licensees “periodically assess and verify and permanently improve the nuclear safety of the nuclear facilities, to the extent that is reasonably possible and in a systematic and verifiable manner”. [78]
- Applying many improvements as the result of the PSRs and the performance of the stress tests and analyses such as: protection against fires, electrical separation of trains, containment isolation and standby gas treatment system, remote and alternative shutdown systems, calculation methodologies used in accident analysis, address situations of loss of major areas, complete loss of power supply and loss of the ultimate heat sink.
- Implementing Royal Decree 1836/1999, which establishes that the licensee shall continuously strive to improve the conditions of nuclear safety and radiological protection of the facility; analyse the best existing techniques and practices in accordance with the requirements established by CSN and implement those considered by this body to be appropriate. The CSN may at any time require the licensee to perform analyses for the implementation of improvements in nuclear safety and radiological protection, pursuant to the provisions of Law 15/1980 creating the CSN. [479]

These national requirements involve the performance of periodic comprehensive and systematic safety assessments of existing NPPs; and address reasonably practicable/achievable safety improvements to be implemented in a timely manner.

Most of the regulation and requirements were enacted before the establishment of the VDNS.

I-19.5. IAEA Safety Standards and other international Good Practices in the national requirements and regulations addressing the VDNS principles

The third principle of the VDNS is:

National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the Review Meetings of the CNS. [77]

Spanish national requirements and regulation take into account the relevant IAEA Safety Standards throughout the life-time of a nuclear power plant by the establishment of:

- CSN IS-26 on basic nuclear safety requirements applicable to nuclear facilities;
- IS-27 on general NPP design criteria, which include both the Spanish practices previously applied, the IAEA standards, and the country of origin standards (USA and Germany), as well as the WENRA reference levels of 2008.

Additionally, some IAEA Safety Guides are included and analysed under the scope of the Periodic Safety Reviews (PSR) or other regulatory documents.

The Spanish regime in relation to IAEA safeguards and non-proliferation is governed by the Euratom Regulation No. 302 of 2005. The Additional Protocol to the Safeguards Agreement, signed jointly by Spain, Euratom and the IAEA, is adapted by means of Royal Decree No. 1206 of 19 September 2003, for the application of the commitments undertaken by the Spanish State in the Additional Protocol to the Safeguards Agreement deriving from the Treaty on the Non-Proliferation of Nuclear Weapons.

I-19.6. Significant events that have resulted in large scope changes to the regulatory framework

Some significant events have resulted in large scope changes to the regulatory framework. These events are mainly the following:

I-19.6.1. TMI-2 Accident

Following the accident at Three Mile Island Unit 2, the Nuclear Regulatory Commission of the United States (NRC) developed an action plan, providing a comprehensive and integrated plan to improve safety at power reactors. These requirements were also applied to Spanish NPP reactors, as they were coming from the country origin of technology of most facilities.

I-19.6.2. WENRA harmonization project

In its study on the harmonization of reactor safety, published in January 2006, WENRA set out the conditions to be met by the requirements established by the different regulatory authorities in order for them to be considered “national requirements”. On the basis of this study, each WENRA member country drew up an action plan for the performance of the harmonization committed to CSN instructions are perfectly embedded in the Spanish regulatory framework and in addition fulfil the WENRA requirements for consideration as “national requirements”.

I-19.6.3. Fukushima Daiichi NPP accident

Based on the results of the Stress Tests, for which specifications were developed by ENSREG, the Spanish NPPs made a set of significant safety improvement proposals that were converted into CSN requirements by means of the issuance of several ITCs (Complementary Technical Instructions)

Council Directive 2014/87/EURATOM, of July 8th 2014, modifying Directive 2009/71/EURATOM, which established a community framework for the nuclear safety of nuclear facilities. The aim was to implement improvements in nuclear safety in the wake of the Fukushima Daiichi NPP accident, to be applied by the member countries. The consideration of the safety objectives as expressed in the Directive together with the concept of reasonably practicable, made the CSN to modify the rules to develop the PSR, in which framework, safety improvements are proposed.

Additionally, in September 2014, WENRA published new reference levels following their revision as a result of the accident at Fukushima Daiichi nuclear power plant. These new framework led to some changes in the national regulation.

I-19.7. Spanish approach for safety improvements at NPPs

I-19.8. Introduction

Spanish nuclear regulatory framework is essentially prescriptive. The principle of continuous safety improvement is addressed in several regulatory documents, like in the Law No. 15 of 22 April 1980, by which CSN was created, which grants to the regulatory body the capacity to issue binding orders to the license, and also in the Royal Decrees 1836/1999 “Regulation on Nuclear and Radioactive facilities” and 1400/2018 “Nuclear Safety in nuclear facilities”.

I-19.9. Identification of Safety Improvements

I-19.9.1. Drivers for the enhancement process

Measures to increase the level of safety at the NPP are gradually taken in accordance with new knowledge and experience. New knowledge and experience have emerged mainly from analysis of operational experience (internal and external), peer reviews, research and development, new knowledge and through the use of safety analyses and probabilistic safety analyses (PSA).

International accidents/incidents occurred at nuclear power plants, such as TMI NPP, Chernobyl NPP or Fukushima Daiichi NPP accident have had a major influence on these measures. Many of the previous mentioned improvements were converted by the regulator into requirements. Modernization programs and Power Uprate programs are also important mechanisms to identify safety improvements.

In Spain, a comprehensive and systematic safety assessment is carried out for existing NPP, referred to as Periodic Safety Reviews (PSR), in accordance with IAEA standards and other good practices acquired through the experience accumulated and exchange with peers. In the PSR, the plants identify reasonably and practicable safety improvements for implementation during the following period. These improvements are prioritized on the basis of their degree of benefit for safety.

Note: Related to PSR, the Spanish regulatory framework, before applying last IAEA Standard for PSR (SSG-25 [4]), required an additional process in order to compare current designs with state-of-the art standards. This process, nowadays incorporated into the PSR, was called Conditioned Application Standards (NAC).

Another differential fact of the Spanish Regulatory framework is that, because the country of origin of the technology of most facilities is USA, some NRC requirements are also mandatory (10CFR50 modifications, Generic Letters, Bulletins...) which could also result in improvements. Annually, each plant has to elaborate a report analysing the applicability of new regulations issued nationally and internationally, with special focus on the country of origin of the technology.

Changes to the regulatory framework and the national nuclear programme is also a measure to improve safety. Some examples are:

- Implementation of WENRA terms of reference. According to the latest harmonization status report, as of March 2018, of the 342 WENRA reference levels in force (including those revised following Fukushima Daiichi NPP accident), just 9 remain to be incorporated into the national standards, which means a high level of transposing, being one of the top countries.
- Establishment of secondary legislation (Royal Decree) for:
 - Responsible and safe management of spent nuclear fuel and radioactive waste;
 - Physical protection of nuclear facilities and materials and radioactive sources;
 - Nuclear safety in nuclear facilities.
- Establishment of CSN instruction (as legally binding technical standards): for example on the requirements of the NPP fire protection program, on emergency operating and severe accident management procedures at NPPs, on design basis accident analysis at NPPs, and on Technical Specifications.

I-19.10. Selection process of safety improvements

Methods used to select safety improvements mainly come from engineering judgement applied under a deterministic approach and from regulatory agreements between regulator and licenses.

No risk informed decision is used, nor is cost-benefit analysis used in the selection process. Although it is recognized that PRA is a good tool to select and prioritize plant modification needs and to compare the safety significance of alternative solutions it is not really used.

Concerning the concept of reasonably practicable safety improvements, it has not been applied so far and there is no standard set, or tick list, of specific engineering or operational improvements that could be applied to this concept. There is no any systematic method for assessing what are considered reasonably practicable safety improvements. They would be considered on a case-by-case basis mainly using engineering judgement and the process of deciding what is or is not reasonably practicable is generally iterative between the licensee and the regulator.

I-19.11. Implementation of safety improvements

All the safety improvements selected for implementation are performed under the scope of approved NPP procedures that follow regulatory requirements for their evaluation.

Times frames depends on the regulatory agreements achieved, the safety significance of the modification, the time needed for development and training, the status of the NPP necessary for implementation.

I-19.12. Main and recent safety improvements for existing nuclear power plants

I-19.12.1. Post-Fukushima safety enhancements

Post-Fukushima National Action Plan has been an important milestone in the process of continuous improvement of Spanish NPPs safety.

Spanish operating NPPs carried out the European Stress Tests (2011) and their associated peer review processes. In accordance with ENSREG agreement of July 2012 for European stress tests, the CSN published in 2012 a National Action Plan (NAcP) and has been subjected to two European Peer Reviews (2013/2015).

Additionally to the Stress Tests scope, the CSN launched a parallel process for improving the protection of NPPs against malicious acts (i.e. loss of large areas).

The plants were formally required (ITC) to perform Stress Tests and Loss of large areas evaluation. All the “findings” derived from the licensee and CSN analysis were required to implement through specific regulatory requirements (ITC). Major modifications required evaluation and “authorization” of the Ministry and CSN.

Post-Fukushima improvements could be classified as:

- Protective measures aimed at protecting SSCs against extreme natural phenomena:
 - Barriers against flooding;
 - Seismic resilience reinforced.
- Preventive measures aimed at avoiding fuel melt (in the reactor and the SFP):
 - Mobile equipment (pumps, generators).
 - Acquisition of 380 V AC portable diesel generators to provide feed for the minimum critical loads defined in the prolonged SBO scenario and for the installation of emergency connection systems for this equipment;
 - Acquisition of portable diesel pumps for the extinguishing of a major fire in the absence of off-site power supply or in the event of damage to the plant’s fixed fire-fighting systems, and for providing reactor or containment make-up water if necessary.
 - Instrumentation, procedures.
 - Capacity to fight large fires (well beyond DB).
- Mitigation measures: aimed at minimizing the impact on people and environment:
 - Improved plant capacity (new Alternative Emergency Management Centre (CAGE), human resources);
 - A common Emergency Support Centre (CAE) sharing resources among the Spanish NPP and capable of providing support in the event of an emergency at any of the sites;
 - Implementation of measures to address extensive damage accident scenarios, including interfaces with existing plant facilities, resources and portable equipment, equipment storage areas, heliport;
 - Drawing up of Extensive Damage Mitigation Guidelines (EDMG) and the new Extensive Damage Emergency Guidelines (EDEG);
 - Installation of Filtered Containment Venting System;
 - Installation of Passive Autocatalytic hydrogen Recombiners (PAR);
 - External Containment spraying & Radioactive liquid waste confinement;
 - New procedures to manage these potential situations have been developed.

All the improvements have been fully implemented by the Spanish NPPs in accordance with the initial challenging schedule (2012-2016), with minor delays to accommodate refuelling shutdowns. Figure I-19-4 shows major improvements.

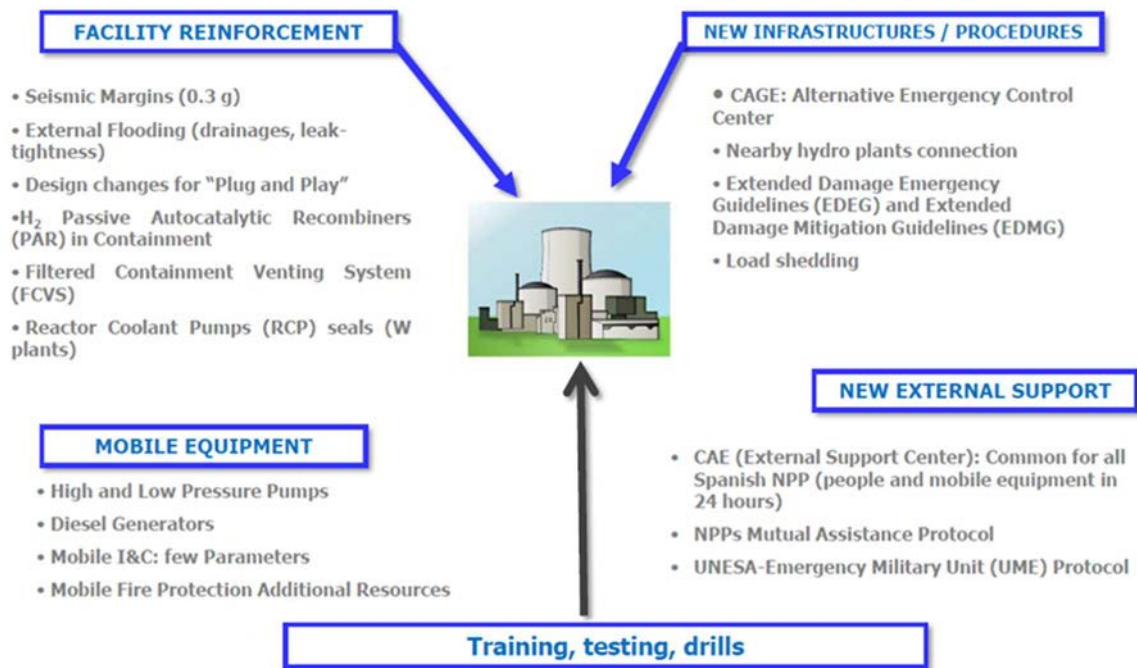


FIG. I-19-4. Major safety improvements at Spanish NPPs implemented after Fukushima Daiichi NPP accident. [76]

In addition, Fig. I-19-5 represents the safety margin increase achieved with these improvements.

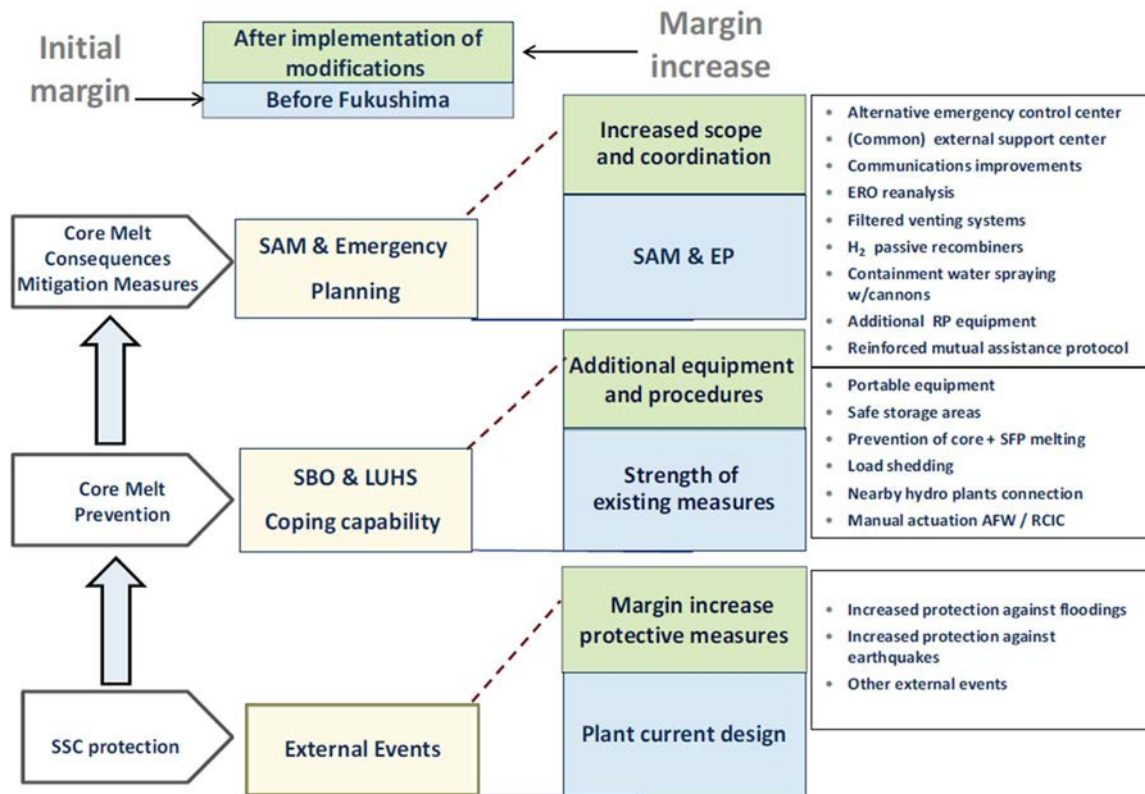


FIG. I-19-5. Safety margin increase in Spanish NPPs after Fukushima Daiichi NPP accident. [76]

I-19.13. Other safety enhancements

Other technical improvements and modification were also made lately, based mainly on the result of deterministic and probabilistic safety analysis.

Some NPPs have implemented specific improvements such as:

- Installation of alternative shutdown panels to guarantee safe shutdown following a fire in the cable room or control room;
- Filtering and ventilation systems of different plant buildings, including the installation of a new redundant train in each group for the spent fuel building;
- Systems for the protection of buildings against atmospheric discharges;
- Physical separation and electrical insulation of nuclear class circuits.

I-20. SWEDEN

I-20.1. Introduction

This provides information regarding safety improvements in Sweden following a reassessment after a new regulatory framework. The examples given, are from the Swedish PWRs, but in general the same are also valid for the BWRs.

I-20.2. Regulatory framework

In 2005 the Swedish Radiation Safety Authority issued a complete new set of regulatory framework.

A large scope change came with this new framework, that introduced new and modern requirements that the plants needed to fulfil. Below are examples of what the Swedish regulatory framework requires.

- Events and conditions that are relevant to radiation safety shall be identified and valued. Based on the valuation, measures shall be implemented to ensure that the operations are carried out in a safe manner.
- An established defence in depth with associated barriers and other measures that are adapted to the operation shall be implemented.
- There shall be a protection against man-made threats in order to prevent radioactive releases.
- Preparedness and management of radiological emergencies. I.e. an organization and instructions for radiological emergencies shall be established.

An IRRS review of the Swedish Radiation Safety Authority's activities were conducted in February 2012, it was found that the Swedish regulations for nuclear facilities have historically expanded as the need for regulation has arisen. The review report found that the IAEA safety standards have been used as a basis for the Swedish Nuclear Safety Code or referred to in these, but not systematically.

Requirements to address the stress tests are not incorporated in the regulatory framework yet, but there are requirements to install an Independent Core Cooling System for extreme external events. A new regulatory framework is being prepared, that will address this and the IRRS review findings, in accordance with the IAEA safety standards.

I-20.3. Methods for identifying safety improvements

To identify areas for safety improvements several drivers are used that are shortly described below.

Based on the new regulations issued in 2005, that was a “back fitting” set of requirements, the regulatory requirements has been a major driver for safety improvements. Other drivers are:

- PSR
 - The PSRs are conducted every 10 years in accordance with SSMFS 2008:1 that specifies 17 topics to address. It is judged that the 14 safety factors described in IAEA Safety Standards Series No. SSG-25 [4] are covered. The conclusion is documented in an overall report that refer to the areas where safety improvements are identified. This report is submitted to the regulator with an action plan to implement the improvement.
- Operational experience
 - A dedicated group are assigned to evaluate experience reported from various sources such as IRS, WANO and several owner's groups. If an experience is deemed applicable for the plant experts in that area are evaluating if actions are necessary. Based on the experts' recommendation a decision is made if the experience will lead to some corrective actions.
- LTO
 - LTO is performed prior to the original design estimated life is reached. IAEA Safety Reports Series No. 57 is followed, together with IAEA Safety Standards Series No. NS-G-2.12. IGALL [80] and GALL are supporting documents in the process. Several Time Limiting Ageing Analysis (TLAA) are performed covering seven different areas. The LTO is performed in parallel with the PSR and the conclusions are included in the PSR.
- Aging Management
 - In accordance with SSMFS 2008:1 an Ageing Management Programme (AMP) is implemented based on IAEA Safety Reports Series No. 15 and follows the IAEA Safety Guide No. NS-G-2.12. The AMP are divided into six areas, Primary systems, Mechanics, Electricity, Instrumentation, Civil and Fuel. Each area are managed by different groups within the maintenance department with support from several other departments. Obsolescence management are included in the programme.
- Internal and external review
 - Internal and external reviews are conducted on a regular basis with the aim to identify areas where an improvement may be needed. WANO, SALTO, peer-reviews and Technical Support Missions (TSM) have been conducted and provided several suggestions for improvements.

I-20.4. Selection process of safety improvements

When an area for improvement is identified, that is not a regulatory request, in one of the above drivers, a process starts with several steps before an implementation may be decided.

- First an evaluation is always performed. This evaluation is initially done by engineering judgment or a simplified analysis but may be supported by PSA.
- This evaluation is then reviewed and approved for further analysis.
- This further analysis investigate the level of the safety improvement and describe the work and time needed to make the improvement, together with an estimated cost. Here PSA, DSA and other analyses may be used.
- Based on this a comparison is made to other identified improvement and based on the budgetary situation the improvement that gives the highest improvement compared to the effort is chosen. An identified improvement is then planned for installation and a project starts in accordance with this plan.

For regulatory request the same process is followed but only to define the scope of work. Since the regulator has recognized that all requirements are not possible to fully implement in an old unit, there may be discussions regarding what measures are “Reasonable and Practicable”. However there is always improvements done that are approved by the regulator.

The new set of regulations did introduce some requirements that intended to improve the overall safety of the Swedish NPPs. For some of the areas the intended safety improvement required a more comprehensive effort. For this a specific organization (see Fig. I-20-1) was created with the aim to identify and propose necessary modifications in the plant and to all affected documentation, such as the operating procedures, DSA, PSA and safety analysis report. This organization was given an economic frame for the identified improvements with the benefit of being able to move resources between different projects, which is normally not possible.

The Programme Director had the overall responsibility of the programme with the support of a steering committee. All contacts with the regulator went through the director. The Programme Manager had the support of a group of experts to develop the safety concept to meet the new requirements. If it was judged that the consequences of a specific requirement were not reasonable or impossible to implement, a compensatory solution was developed and presented to the regulator for discussion. When the safety concept was established with a new licensing basis, several improvements were required. An action plan to implement these improvements was presented and accepted by the regulator.

The programme manager delivered specifications for each of these improvements to the director. The director decided to start projects based on these specifications supervised by the Key Account Managers. The projects followed the plants normal procedures with backup from the programme manager's expert team.

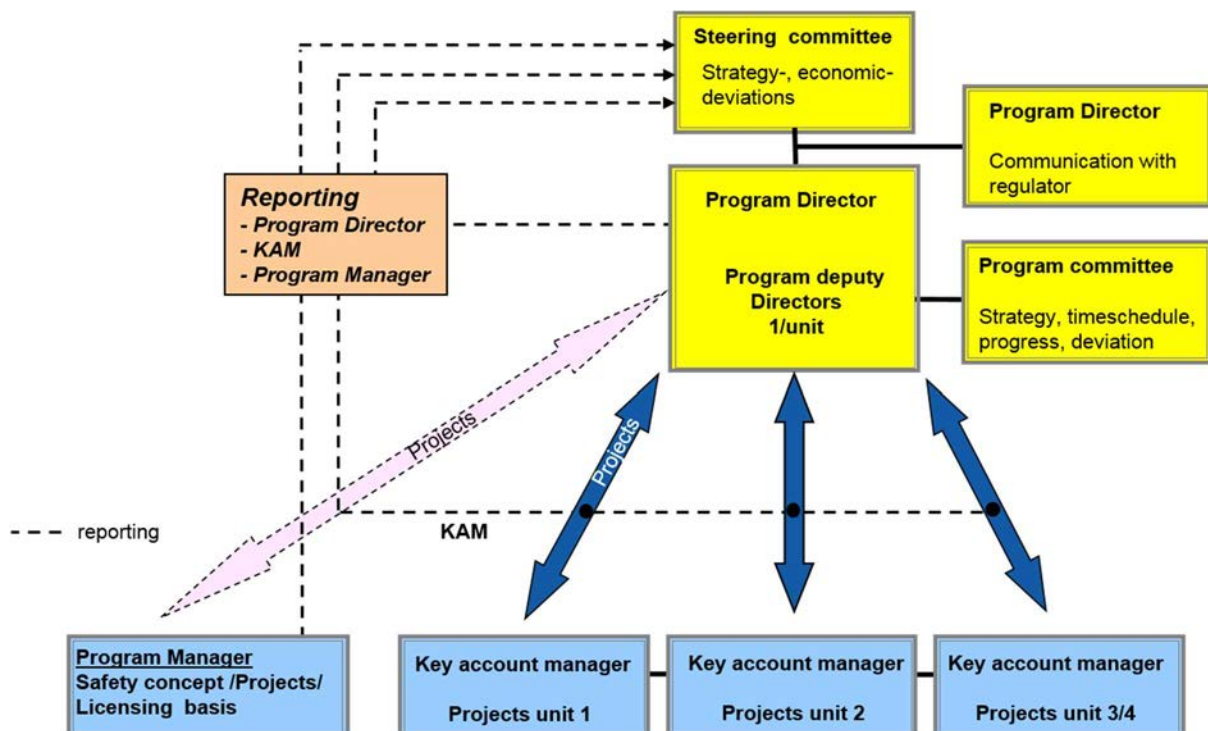


FIG. I-20-1. Organization created for implementation of the new regulation framework.

The new regulatory framework was issued in 2005 and due to the large impact, the action plan did not end until 2015.

I-20.5. Safety improvements

I-20.5.1. Safety improvements after major events implemented in the 1980s

Examples of earlier safety improvement implemented after major events are:

- After the TMI accident:
 - Reactor Vessel Head Ventilation;
 - Improved Post Accident Monitoring.
- Following the Chernobyl NPP Accident:
 - Filtered Vent System was installed;
 - Severe Accident Mitigation Guidelines.

I-20.6. Safety improvement following the new regulatory framework

- A new definition of the safe state, (under critical, temperature below 93°C and no pressure), e.g. Operation Mode 5.
 - Originally the safe state was Operation Mode 3, (under critical and full pressure and temperature). This change required that new parts of the plant had to be credited to reach the safe state.
- New internal and external Events was to be included in the design basis.
 - The events Earthquake and Fire was two events that had a large impact on the safety improvement installed. Originally, an earthquake was not a part of the licensing basis since Sweden were considered as a low seismic area (only Forsmark 3 and Oskarshamn 3 had that originally).
- Separation and Environmental qualification.
 - Based on the new definition of the safe state, parts outside the Containment that was required in order to reach the Safe State, needed to be qualified/protected for the environmental condition following a Pipe Break or a Fire. The older plants had some areas where the structure did not provide for a full separation, that required special attention.

The above paragraphs did lead to the most efforts and resulted in nearly 100 different projects. Some of the major improvements are illustrated in Fig. I–20-2. Environmental qualification was an area that did require a large effort, primary in the areas outside containment, e.g. replacement of components, installation of new dampers to strengthen the fire cell integrity and installation of fire resistant cables.

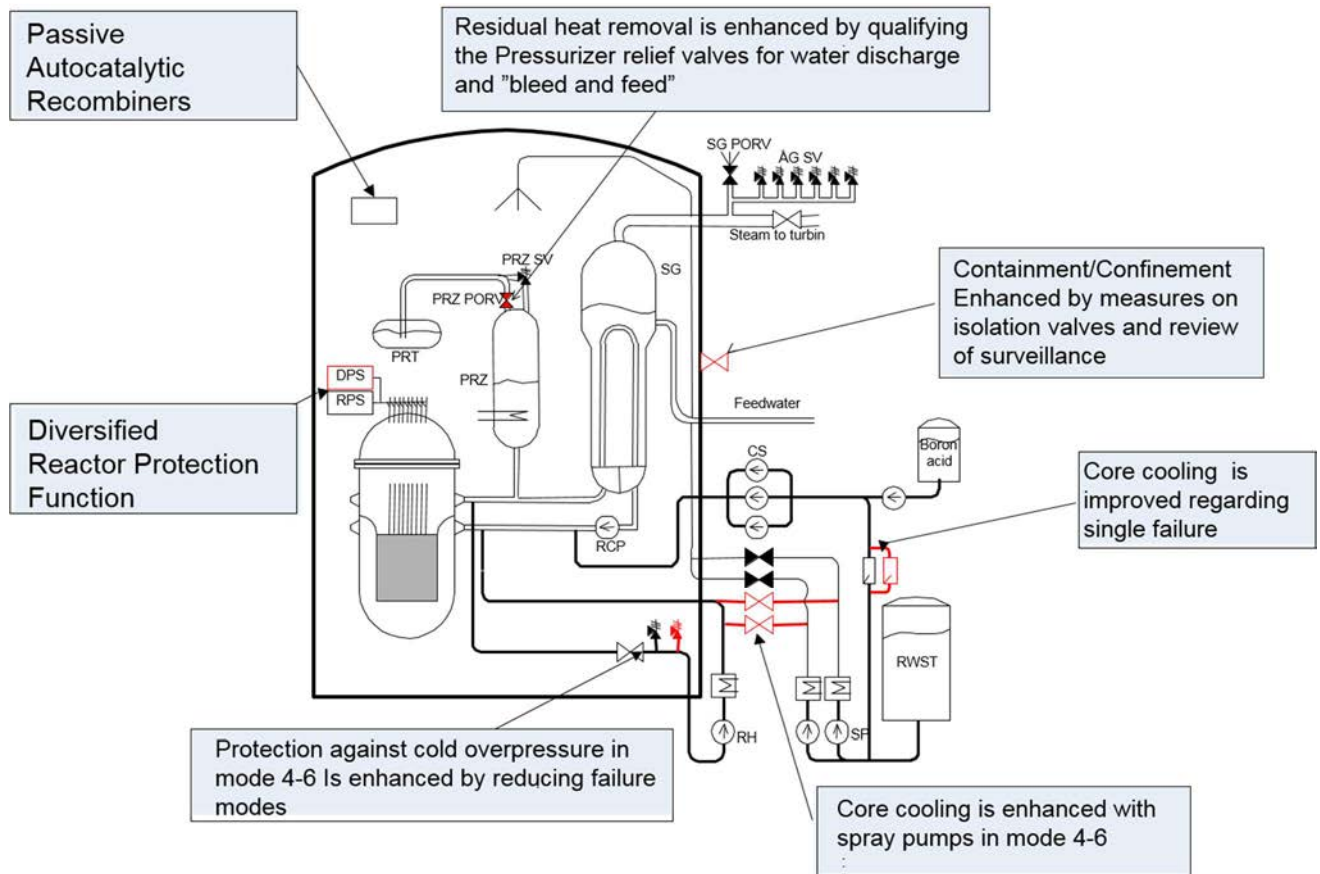


FIG. I-20-2. Examples of some of the major safety improvements.

For the seismic requalification the SMA methodology accordance with EPRI-NP-6041 was applied. External experts were engaged together with internal SQUG qualified personnel to perform walk-downs. The outcome of the SMA requalification resulted in replacement of several electrical equipment as well as strengthening of supports, among other things.

I-20.7. Swedish National Action Plan (SNAP) following the stress tests.

Following the stress tests an action plan were established that included the following improvements (see Fig. I-20-3)

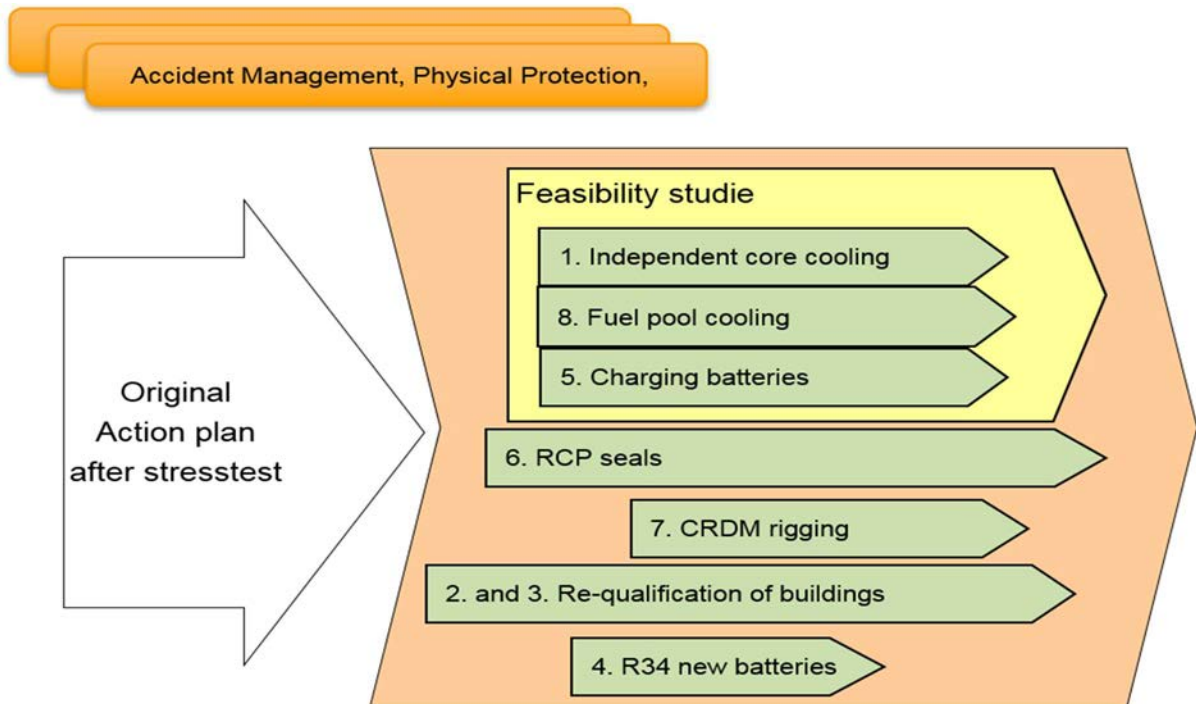


FIG. I-20-3. The improvements identified by the Swedish National Action Plan.

Of the above areas only the CRDM rigging are not to be modified since new analysis shows that they will meet the new requirements. All others are installed or will be implemented before the end of 2020.

RCP passive seal are implemented with the purpose to prevent loss of reactor coolant following different black-out scenarios. New batteries with extended capacity to 8h are installed. New diesel generators with the capacity to charge the batteries and support the start of the emergency diesels are in place.

Following the feasibility studies the regulator has required that an Independent Core Cooling (ICC) system to be in place in 2020 that manage the DEC scenarios Extended Loss of Altering Power (ELAP) and Loss of Ultimate Heat Sink (LUHS) together with extreme external conditions.

The strengthening of the fuel pool cooling is due to practical reason included into the solution for the above DEC

I-20.8. Improvement for the DEC (planned)

The solution for the above mentioned DEC scenarios the following criteria shall be taken into account:

- Extended Loss of AC Power for at least 72 hours;
- Loss of normal access to Ultimate Heat Sink (LUHS) for at least 72 hours;
- Extreme external hazards (a frequency of 10⁻⁶/year to be taken into account);
- Independency requirements;
- Physical protection (man-made events).

In order to meet this criteria a fixed solution was required by the regulator. A separate building designed to manage extreme external hazards is to be erected for each unit. This building will contain enough water for 72 hours. For the PWR's two tanks are provided. One for the steam generators and one for make-up of the reactor coolant system. Each tank have external connections to facilitate refilling, if needed, after the stipulated 72 hours. The pumps are driven by an diesel engine or by electrical motor that receives power from a diesel generator located inside this new building. Diesel storage is also provided that last at least 72 hours with the possibility to refill if needed.

Figure I-61 illustrates a schematic view of the planned installation of the independent feed water system that connects to current auxiliary feedwater system outside of the containment. All existing parts credited for the DEC scenarios has been evaluated to meet the same criteria as the new installations. For the new equipment's a single failure is not considered.

To strengthening the fuel pool cooling a separate pump will provide water to the pool with the Feed and Boil principle. Upon loss of all electrical power the current cooling system will stop, and the temperature will rise until boiling starts. This pump will provide make-up water to ensure the continuous cooling of the fuel.

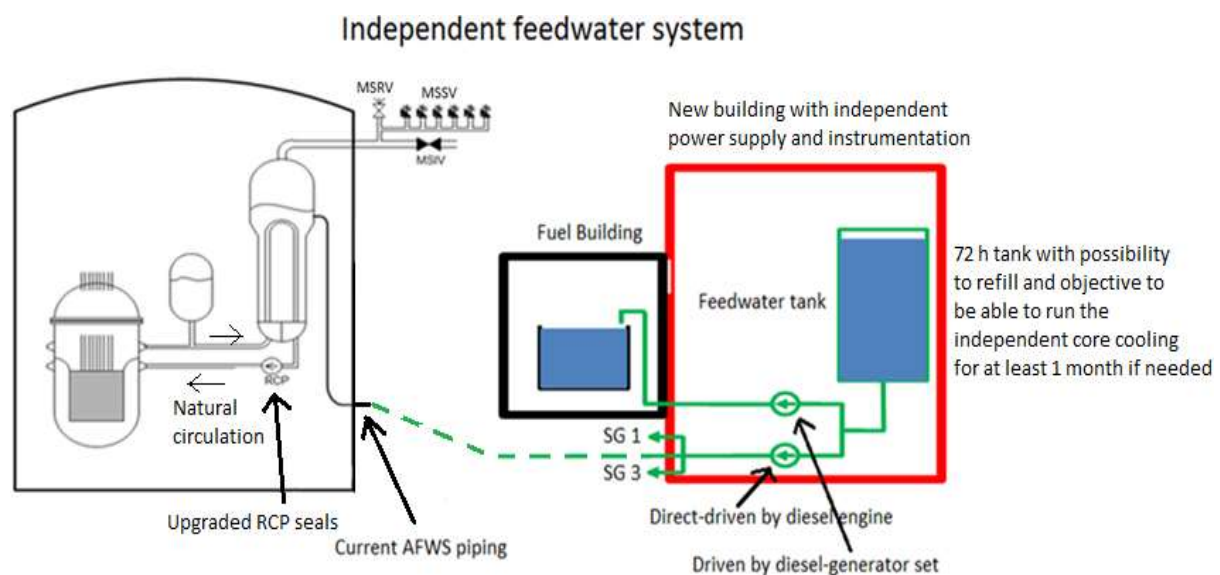
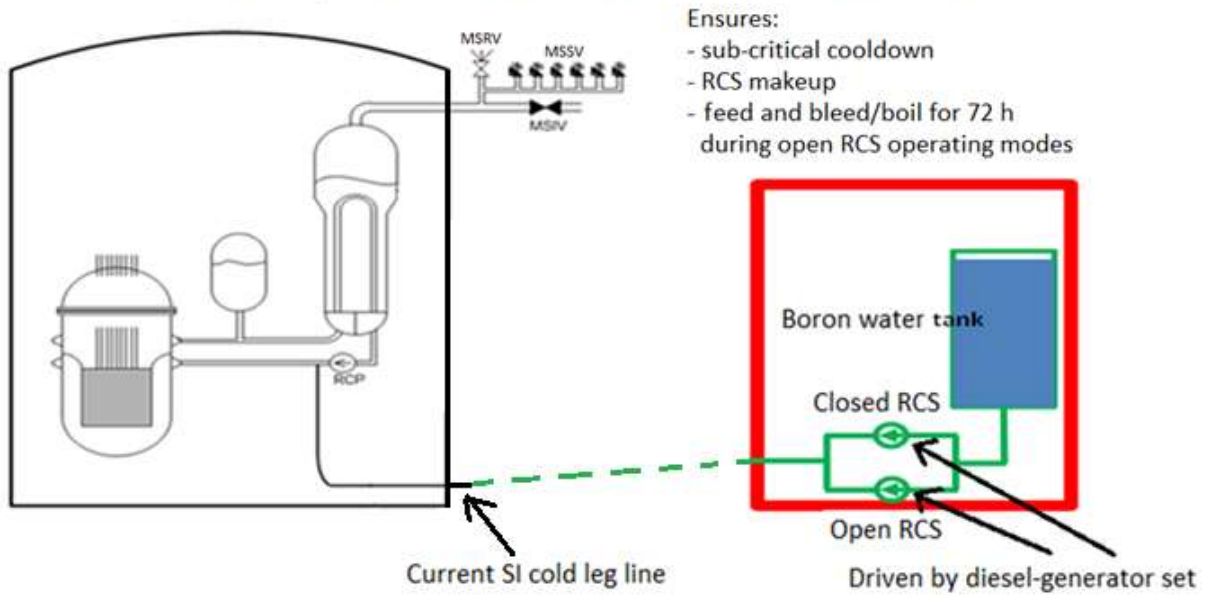


FIG. I-20-4. The independent feed water system including water supply to the spent fuel pool.

For the independent volume control system (see Fig. I-20-4) the same principles are valid but both pumps are driven by an electrical motor that receives power from the diesel generator, one is a displacement pump that provides water with high pressure to the RCS for make-up of shrinkage and small leakage. The other pump is a centrifugal pump that provides water if the event occurs during outages in mode 5 and 6 when RCS is open to the containment atmosphere. The independent volume control system connects to the current safety injection system outside the containment.

Independent volume control system



- Ensures:
- sub-critical cooldown
 - RCS makeup
 - feed and bleed/boil for 72 h during open RCS operating modes

FIG. I-20-5. Schematic view of the independent volume control system.

I-21. SWITZERLAND

I-21.1. Regulatory framework

The statutory and regulatory framework for the peaceful use of nuclear energy is stipulated by the Swiss constitution (first level), Federal legislation (second level), the ordinances (third level) and the ENSI guidelines (fourth level). Legislation regarding the use of nuclear energy and radiation protection is enacted solely at national level. The Federal Parliament and the Federal Council have the sole right to enact laws in this area. The material provisions regarding authorisation and regulation, monitoring and inspection are based on the Nuclear Energy Act (NEA), the Federal Law on Radiological Protection (RPA) and the ENSI Act (ENSIG).

The fundamental provisions of the Nuclear Energy Act regarding the principles of nuclear safety and the operators' responsibilities for the safety of their nuclear power plants apply, as well as the fundamental requirements in the Nuclear Energy Ordinance (NEO) and in the Ordinance of the Federal Department of the Environment, Transport, Energy and Communications (DETEC) on Hazard Assumptions and the Evaluation of Protection Measures against Accidents in Nuclear Installations (DETEC-O).

Articles 7, 8 and 10 of the NEO contain internationally recognised principles to guarantee the safety of nuclear facilities. The strategy specified in Article 7 to ensure the nuclear safety of nuclear facilities at four levels (the "defence in depth" concept) is stated in more practical detail in Articles 8 and 10. In accordance with Article 8, protective measures for nuclear facilities have to be implemented against accidents that originate both inside and outside the facility. In addition, those accidents that have to be brought under control without an inadmissible release of radioactive substances are explicitly stated.

Article 10 defines principles for the design of the safety functions of nuclear power plants. These include, in particular, the single failure criterion, the principles of redundancy and diversity, the functional and physical separation, the automation principle and the conservatism in design. As regards compliance with these design requirements, however, the applicable principle, in accordance with Article 82 of the NEO, is that existing nuclear power plants need to be backfitted to the extent that is necessary on the basis of experience and the state-of-the-art in backfitting technology and, beyond that, insofar as this results into a further reduction of risk and is appropriate.

The DETEC-O stipulates hazard assumptions for accidents that originate inside and outside the plant, as well as the radiological and technical criteria for proof of adequate protection against accidents. Accordingly, hazards due to natural events, in particular earthquakes, flooding and extreme weather conditions, have to be determined with the help of probabilistic hazard analysis. For proof of adequate protection against natural events, account has to be taken of hazards with a frequency greater than or equal to 10^{-4} per year. ENSI is responsible for drawing up guidelines, which are support documents that formalise the implementation of legal requirements and facilitate uniformity of implementation practices. While compliance with the laws and ordinances by the operators is mandatory, ENSI may allow deviations from the guidelines in individual cases, provided that the suggested solution ensures at least an equivalent level of nuclear safety or security.

In addition to the IAEA and the OECD Nuclear Energy Agency, WENRA is a major driving force in efforts to harmonise nuclear safety requirements at the European level. Switzerland was one of the founding members and has held the chair of WENRA since 2011. WENRA

provides regulatory authorities with a single forum at which they can share their years of experience in regulating a range of nuclear facilities as well as in elaborating and implementing standards. Based on this expertise so called Safety Reference Levels (SRLs), which are based on the IAEA safety standards, are issued. The SRLs may be adopted and incorporated into national legislation. The implementation is monitored by the corresponding WENRA working group.

The Inspectorate participates in the following WENRA groups: Reactor Harmonisation Working Group and Working Group on Waste and Decommissioning. The Swiss self-assessment in the area of Reactor Harmonisation identified a number of SRLs to be incorporated into the Swiss regulatory framework. All WENRA SRLs for spent fuel and waste storage as well as for decommissioning are implemented in the Swiss regulatory framework, the latter by the guideline ENSI-G17, "Decommissioning of Nuclear Installations," which was published in April 2014. Besides considering the WENRA SRLs, the guideline ENSI-G17 also respects the IAEA safety standards on decommissioning. The Inspectorate participates in several IAEA committees to promote high international standards in nuclear safety. On the other hand, the Inspectorate harmonises its guidelines with IAEA safety standards. Therefore, when issuing a new guideline or revising an existing one, the Inspectorate analyses the IAEA Safety Fundamentals and Safety Requirements relevant to the topic of the guideline. Every guideline is accompanied by an explanatory report. This report shows also for each IAEA Safety Requirement where in the Swiss legislation or the Inspectorate's guidelines it is implemented.

In addition, the Inspectorate has committed itself to implementing all safety reference levels issued by WENRA. In the explanatory reports, it is shown for each guideline if and how each safety reference level is implemented.

The Inspectorate has published its Regulatory Framework Strategy consisting of five guiding principles:

- ENSI's regulatory framework is harmonised with the relevant international requirements and is comprehensive.
- ENSI's regulatory framework is based on existing, tried-and-tested regulations, insofar as they are suitable for application within its supervisory scope.
- ENSI issues its own guidelines only when it is necessary to do so.
- ENSI's guidelines are drawn up transparently, with the involvement of all stakeholders.
- The level of detail of ENSI's regulatory framework is based on the hazard potential and the risk.

I-21.2. Regulator`s activities after major accidents

The accidents in Three Mile Island NPP and Chernobyl NPP lead to large scope changes in the regulatory framework as from that point the safety layer 4 became more important and requirements were defined regarding that level of defence.

The accident in Fukushima Daiichi NPP however lead to comprehensive activities to improve the performance in case of a beyond design basis accident but entailed no major adjustments in regulation.

Directly after the reactor accidents in TEPCO Fukushima Daiichi NPP on 11 March 2011, ENSI ordered measures for a review of the safety of the Swiss nuclear power plants. The measures were set out in several formal orders issued by ENSI. The first three orders (dated 18 March, 1 April and 5 May 2011) called for immediate measures and additional reviews.

The immediate measures comprised the establishment of a joint external emergency storage facility (Reitnau storage) for the Swiss nuclear power plants, including the necessary plant-specific connections for accident management (AM) equipment, and the backfitting of feeds for the injection of water into the spent fuel pools from the outside. The additional reviews covered the in-depth design reassessment of the Swiss NPPs in respect of earthquakes, external flooding and a combination thereof. A review of the coolant supply for the safety and auxiliary systems and the spent fuel pools was also requested.

In respect of the external hazards, ENSI requested the operators to update the hazard assumptions making use of the latest scientific results and state-of-the-art analysis techniques. Thus, for the seismic hazard, intermediate hazard curves as determined from a SSHAC (Senior Seismic Hazard Analysis Committee) Level 4 process (the so called PEGASOS Refinement Project) were adopted; for the flooding hazard, the most updated simulation and transport techniques were used in order to take into account phenomena such as debris transportation, clogging. For extreme weather hazards, a request was issued to perform a probabilistic hazard analysis in order to determine the $1E-4/y$ (mean) hazard curve, as is necessary for all external hazards, in accordance with the Swiss legal and regulatory framework.

In parallel with these investigations by the operators, ENSI carried out topical inspections, which in 2011 included reviews of the existing cooling systems for the spent fuel pools, protection against external flooding and the systems for filtered containment venting. Topical inspections were continued during 2012; they covered the plants' strategies in case of a prolonged loss of the power supply, the processes and documented requirements for assessing external events, and the emergency rooms available in the Swiss plants. The radiation protection equipment available on site, which is a basic prerequisite for coping with a severe accident, was inspected at all the nuclear power plants during 2013. Radiation protection equipment is also essential so that the emergency rooms can be used by the emergency response organization in the longer term.

The results of ENSI's reviews have confirmed that the Swiss nuclear power plants have a high degree of protection against the effects of earthquakes, flooding and combinations thereof, and that appropriate precautions have been taken against loss of the power supply and the heat sink. All the analysed accidents are brought under control, taking into account the most updated hazard assumptions. This means that the basic statutory requirements for fulfilling the fundamental safety functions (control of reactivity, cooling of the fuel assemblies and containment of radioactive substances) are guaranteed. With a view to further improvements to safety, ENSI nevertheless specified a series of additional requests for substantial backfitting measures. For example, ENSI concluded that all Swiss NPPs shall have groundwater wells as part of their (bunkered) special emergency systems as alternate cooling water sources for severe accidents, except for Mühleberg NPP. The operator was therefore asked to propose a solution for a diverse ultimate heat sink. Further examples of backfitting include: temperature and level measurements for the spent fuel pools (SFPs); new feeds for water injection into the SFPs from the outside; for the older Swiss NPPs new safety-grade SFP cooling systems; several AM diesel generators (mostly in fixed installation) and water pumps; PARs for non-inertized containments; seismic isolation of several emergency and special emergency diesel generators; increase of the seismic capacity of numerous components (especially electrical cabinets). Also

the Wohlensee dam, around 1.5 km upstream of the Mühleberg NPP was reinforced against sliding effects in the event of earthquakes, thus significantly reducing the risk of seismically induced flooding at the NPP site.

In 2013, ENSI started in-depth re-assessments concerning extreme weather hazards and NPP provisions against them, as well as concerning hydrogen management in containment. Studies on the extreme weather safety case were submitted by the licensees by the end of 2014. On the basis of the hydrogen management analyses of the licensees, ENSI concluded that for those plants where no containment inertization is in place (i.e. for all NPPs except Mühleberg), additional passive means of hydrogen control are necessary. At the same time, issues related to the safety demonstrations were identified that needed more detailed consideration by the plant operators, including the necessary modifications to the SAMG (severe accident management guideline). The backfitting of (mainly) passive autocatalytic recombiners in the plants is being monitored by ENSI within the framework of the normal permit-issuing process (on-going). Additionally, the knowledge obtained from analysis of the events of the accident at Fukushima Daiichi NPP was reviewed to determine its applicability to Switzerland, and a summary of insights was compiled in an ENSI report entitled «Lessons Learned» in the form of a series of checkpoints. Further points were added on completion of the analyses for the EU stress tests. The processing and implementation of the identified points were updated annually in the Fukushima Action Plan. The last Fukushima Action Plan was released in February 2015 and most of the identified checkpoints were implemented by the end of 2015. By the end of 2016, ENSI published a summary report on all activities that have been performed within the framework of the Fukushima Action Plan, thus concluding ENSI's post-Fukushima activities.

I-21.3. Identification of safety improvements

I-21.3.1. Drivers for the enhancement process

Basically ENSI is using all the applicable drivers listed before in this document. For existing nuclear power plants, a Periodic Safety Review (PSR) is required at least every ten years. Important elements of a PSR are an update of the Safety Analysis Report (SAR), an assessment of design basis accidents, an assessment of the ageing surveillance programme, an update of the Probabilistic Safety Analysis (PSA) and an evaluation of operating experience over the last 10 years. The details (scope and process) of a PSR are defined in the Inspectorate's Guideline ENSI-A03.

Changes in the organization, modifications or backfitting of components and documents (e.g. Technical Specifications) related to safety have to be approved by the Inspectorate. The Inspectorate's associated review may involve inspections. Data from inspections, event assessments and safety indicators provide a foundation for ENSI's systematic assessment of operating safety, carried out annually. In addition, the licensees have to perform annual safety assessments in accordance with the requirements given in the guideline ENSI-G08, and probabilistic evaluations of their operating experience in accordance with the guideline ENSI-A06. Further reviews and assessments of the design basis are mandatory if events of INES 2 and higher occurred in a national or international NPP.

Apart from that ENSI is involved in various multinational organizations especially the IAEA and WENRA and Bilateral Commissions with neighbouring countries and uses all the tools they offer as drivers for further safety improvements.

Furthermore ENSI takes part in IRRS missions. In April 2015, an IRRS follow-up mission was conducted in Switzerland. The mission concluded that the four recommendations and 16 suggestions for which ENSI was primarily responsible had been implemented but that the Swiss government should give ENSI, as the technical nuclear safety authority, the ability to issue legally binding technical safety requirements and licence conditions on nuclear safety, nuclear security and radiation safety.

Also, an OSART follow-up mission to the Mühleberg NPP was completed in June 2014. Switzerland participated in the European Stress Test and its follow-up activities. Furthermore, in December 2013, Switzerland tabled a proposal to amend Article 18 of the Convention on Nuclear Safety and participated in the ensuing Diplomatic Conference. Switzerland contributed actively to the development of the Vienna Declaration on Nuclear Safety.

I-21.4. Selection process of safety improvements

The Nuclear Energy Act defines the general responsibilities of a licensee, including the responsibility for the safety of the installation, the obligation on NPPs to conduct systematic and periodic safety reviews and to backfit installations to the necessary extent. Also further reviews and assessments of the design basis are mandatory if events of INES 2 and higher occurred in a national or international NPP.

There is a dynamic requirement and precautionary principle also for existing NPPs. Article 22, clause 2, letter g of the NEA requires that the licensee shall backfit the installation as far as necessary, in accordance with operational experience, and to be in line with the current state of backfitting technology. Additionally obligatory the licensee has to upgrade the plant if it is appropriate and results in a further reduction of risk to humans and the environment.

The Ordinance on the Methodology and the General Conditions for Checking the Criteria for the Provisional Taking out of Service on Nuclear Power Plants (SR.732.114.5) defines a set of minimal criteria for the existing NPPs to fulfil. In detail, for all internal events up to an initiating frequency of 10^{-6} per year and for all natural initiating events up to a hazard of 10^{-4} per year (mean value) the plants have to fulfil the radiological and technical acceptance criteria for design base accidents. If these criteria are not met the plant has to be taken out of service and back-fitted.

Additional further measures or back-fittings can become necessary if the CDF (Core Damage Frequency)/FDF (Fuel Damage Frequency) is greater than 10^{-5} per year (for LERF greater than 10^{-6} per year). Further measures for improvement have to be identified if an initiating event category contributes more than 60% to the mean CDF and its contribution is more than $6 \cdot 10^{-6}$ per year. All these measures or back-fittings to reduce the risk further have to be appropriate and proportional.

Towards continuously safety improvement, licensees are obligated to show the safety for long term operation, which explicitly comprises all planned backfittings and technical or organizational improvements for the following operating decade.

I-21.5. Outcomes identified of safety improvements

As an example for the PSA as a driver for safety improvements the new bunkered special emergency core cooling and residual heat removal systems were back-fitted in the older Swiss NPP. These back-fitting measures were also driven by the fact that the newer Swiss NPPs had been constructed with such bunkered special emergency systems. During that time bunkered special emergency core cooling and residual heat removal systems were also back-fitted in Germany, so it was a state of the art back-fitting measure to improve safety. Further examples for lessons learned from an accident, the new developed filtered containment venting systems had to be installed in all Swiss NPPs (Chernobyl NPP). The new T-Quencher in the depressurisation chamber of the BWRs and the enlarged sump strainer of the containment of the PWRs and BWRs were installed as a reaction by the Würgassen and Barsebäck events.

The main reaction of the Fukushima Daiichi NPP accident was the implementation of one remote storage for mobile equipment in combination with an organization consist of staff from all Swiss NPP to assist each other during a severe accident.

As a result of the oversight process of the Swiss Federal Nuclear Safety Inspectorate the following safety improvements have been identified and implemented.

Tables I-21-1 to I-21-4 give an overview of further backfitting measures from 1980 to now.

TABLE I-21-1. SIGNIFICANT BACKFITTINGS MÜHLEBERG NPP

| Significant Backfittings Mühleberg NPP | |
|---|--|
| 1980-1990 | <ul style="list-style-type: none"> • Installation of T-Quencher in the torus • Bunkered special emergency core cooling and residual heat removal system (SUSAN) |
| 1991-2000 | <ul style="list-style-type: none"> • Filtered containment venting system • Replacement of reactor protection SCRAM initiation system • Drywell spray and flood system • Alternate Reactor Shutdown and Isolation System • Inertization of the Containment • Replacement of the torus suction strainers • Prevention of initiation of the automatic depressurisation system (ADS) in case of ATWS • Improved reactor level measurement |
| 2000– Now | <ul style="list-style-type: none"> • Implementation of Severe Accident Management Guidance (SAMG) • Procurement of 2 new large mobile diesel generators and 3 additional large fire water pumps (10 m³/min) • Mobile flood walls to enhance flood protection • Emergency spent fuel pool refill system • Additional emergency water supply to restore ultimate heat sink • Additional automatic low pressure core injection system • Additional seismically qualified emergency spent fuel pool cooling system |

TABLE I-21-2. SIGNIFICANT BACKFITTINGS BEZNAU NPP

| Significant Backfittings Beznau NPP | |
|--|--|
| 1980-1990 | <ul style="list-style-type: none"> • Extension of the accident instrumentation • Start of the seismic requalification program • New bunkered building for the borated water tanks |
| 1991-2000 | <ul style="list-style-type: none"> • Filtered Containment Venting System • Replacement of the Steam Generators • Bunkered special emergency core cooling and residual heat removal system • Replacement of the pressurizer safety valves for “primary feed and bleed” • Alternative system to cool the spent fuel pool • New Emergency Feedwater System |
| 2001-2010 | <ul style="list-style-type: none"> • Replacement of the reactor protection and control system by a digital system • Replacement of the secured uninterruptible AC Power Supply • Implementation of SAMG • Several new fire water hook up points • Passive hydrogen recombiners inside the Containment |
| 2011 - Now | <ul style="list-style-type: none"> • Additional water supply system for the spent fuel pool • Procurement of additional fire water pumps stored in tents and 2 mobile diesel generators • Replacement of the reactor pressure vessel head • Replacement and extension of the emergency power supply system • Additional well water supply to the steam generators • Additional emergency seal water supply system • Additional, seismically qualified heat removal system for the spent fuel pool • 2 additional passive hydrogen recombiners inside the containment (planned) |

TABLE I-21-3. SIGNIFICANT BACKFITTINGS LEIBSTADT NPP

| Significant Backfittings Leibstadt NPP | |
|---|---|
| 1990-2000 | <ul style="list-style-type: none"> • Alternate rod insertion system • Replacement of the suppression pool suction strainers • Filtered Containment Venting System • Active hydrogen igniter system • Additional external power supply for the bunkered special emergency heat removal system • Automatic reduction of the feedwater flow and automatic blockage of the ADS in case of an ATWS • ATWS recirculation pump trip |
| 2000-Now | <ul style="list-style-type: none"> • Procurement of 3 mobile diesel generators • Additional, remote connection point to feed water into the RPV • Implementation of SAMG • Seismic strengthening of the FCVS • Passive hydrogen recombiners (12) and passive hydrogen igniters (8) inside the containment <u>and</u> annulus (planned) |

TABLE I-21-4. SIGNIFICANT BACKFITTINGS GÖSGEN NPP

| Significant Backfittings Gösgen NPP | |
|--|--|
| 1990-2000 | <ul style="list-style-type: none"> • Seismic strengthening of electrical cabinets • Enhancing the robustness of safety significant motor operated valves under accident conditions • Filtered Containment Venting System • Additional independent SFP cooling system • Extension of the accident instrumentation |
| 2001-Now | <ul style="list-style-type: none"> • Replacement of the containment sump suction strainers • Seismic strengthening of non-reinforced masonry inside the electrical building • Replacement of the pressurizer safety valves for “primary feed and bleed” • Implementation of SAMG • Replacement of operational reactor I&C system by a digital system • External flood protection wall • Procurement of 2 new mobile diesel generators and mobile fire water pumps • Seismic strengthening of the emergency and special emergency diesel generators • Seismic strengthening of several electrical components (e.g. batteries, cable tray supports) • Additional iodine filter in the FCVS • Seismic strengthening of the special emergency system and implementation of additional safety functions • Passive hydrogen recombiners (56 inside containment <u>and</u> 2 inside annulus, planned) |

Detailed Design of Safety Improvements

All the Swiss nuclear power plants have been backfitted intensively and in some cases the backfitting costs exceeded several times the initial construction costs of the plant. One of the most important features of the power plants is surely the bunkered special emergency safety systems realized in all the NPPs as well as an independent secondary heat sink (ground water well) realized in all NPPs except Muehleberg.

As a result of the Fukushima Daiichi NPP accident NPP-Muehleberg had to improve the existing water intake of the bunkered special emergency core cooling and residual heat removal system (SUSAN). In July 2011 after a detailed analysis of failure modes the NPP had to install new suction (periscope) pipes in the intake (safety level 3) to protect against extreme river bed load (installed in 2 months): see Figs I-21-1 to I-21-4.

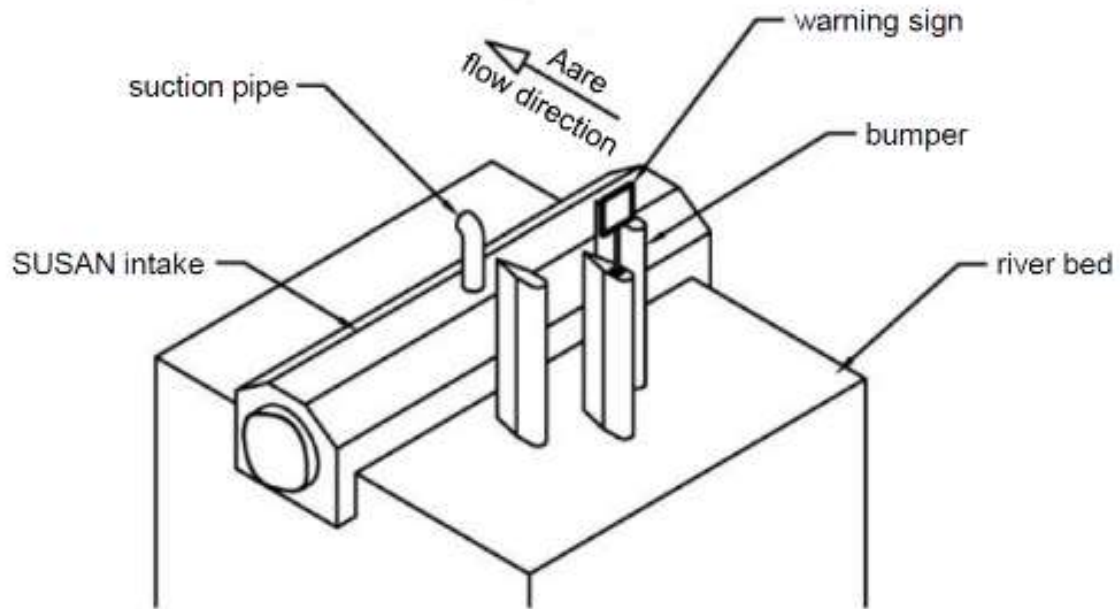


FIG. I-21-1. Modification of the water intake.



FIG. I-21-2. Simulated river bed load (increased hazard, modified intake).



FIG. I-21-3. Suction pipe.



FIG. I-21-4. Protection of the suction pipe (bumper).

As a first step for a diverse cooling water supply of the bunkered special emergency core cooling and residual heat removal system (SUSAN) an injection point was realised, to ensure a cooling water supply by fire engine pumps (2011): see Figs I-21-5 and I-21-6.



FIG. I-21-5. Cooling water injection point.



FIG. I-21-6. Fire water pump, one necessary, three stationed at the plant.

Furthermore, NPP Mühleberg installed a pipe connection to the drinking water network, to feed the SUSAN cooling water system via the drinking water network (high water reservoir Runtigenrain), from a remote water well (REWAG, 2,4km) and from additional remote water reservoirs. At the plant material is stored for a 2,4 km fire water pipeline for further diverse water-supply: see Figs I-21-7 and I-21-8.

Water supply by drinking water network

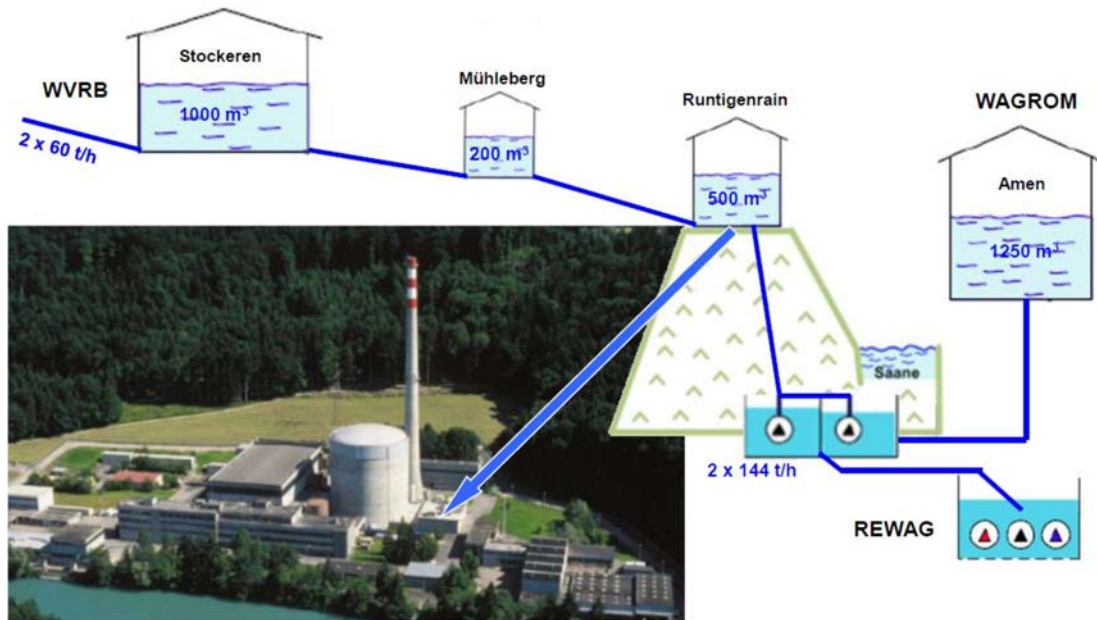


FIG. I-21-7. Cooling water supply via drinking water network.



FIG. I-21-8. 2.4 km fire water pipeline.

I-22. UNITED STATES

The mission of the U.S. Nuclear Regulatory Commission (NRC), as presented in the agency's Strategic Plan (NUREG-1614, Volume 7) [81], is to license and regulate the Nation's civilian use of radioactive materials to provide reasonable assurance of adequate protection of public health and safety and to promote the common defence and security and to protect the environment. The NRC's strategic goals are to ensure the safe and secure use of radioactive materials. The agency achieves its safety goal by ensuring that licensee performance is at or above acceptable safety levels. The NRC's licensees are responsible for designing, constructing, and operating nuclear facilities safely, while the NRC is responsible for the regulatory oversight of the licensees. As of October 2018, 98 nuclear power plants (NPPs) were operating in the United States; these NPPs began operation as early as 1969 and as recently as 2015.

This annex highlights NRC policy and practices, as well as industry experiences, on three topics: (1) safety goals for NPPs, which relates to Section 2 of this TECDOC on national regulatory frameworks and safety objectives; (2) processes for evaluating potential new requirements, which relates to Chapter 5 on integrated decision-making; and (3) examples of comprehensive evaluations of continued plant safety, which relates to Chapter 4 on assessment of the current design (with reference to some of the drivers noted in Chapter 3). Links are provided to detailed information for readers interested in more information beyond this short summary; additional information can be found in the U.S. National Report for the Convention on Nuclear Safety (NUREG-1650) [82] and the NRC's public website²⁶.

I-22.1. Safety goals for nuclear power plants

In 1986, the Commission published a policy statement on safety goals for NPPs (51 FR 30028) [83], focused on broadly defining an acceptable level of radiological risk. The Commission made it clear that it was discussing acceptable risks, not acceptable deaths—no death attributable to nuclear power plant operation will ever be 'acceptable' in the sense that the Commission would regard it as a routine or permissible event. The policy includes two qualitative safety goals:

- Individual members of the public should be provided a level of protection from the consequences of NPP operation such that individuals bear no significant additional risk to life and health.
- Societal risks to life and health from NPP operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The Commission also defined two quantitative health objectives related to prompt and latent cancer fatality risks, as shown below. The Commission specifically noted that, given the uncertainties in risk assessments, "quantitative objectives should be viewed as aiming points or numerical benchmarks of performance ... not a substitute for existing regulations."

²⁶ NRC public website can be access in the following link < <https://www.nrc.gov/>>

- The risk to an average individual in the vicinity of an NPP of prompt fatalities that might result from reactor accidents should not exceed 0.1 percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near an NPP of cancer fatalities that might result from NPP operation should not exceed 0.1 percent of the sum of cancer fatality risks resulting from all other causes.

The Commission also expressed its view of the importance of accident prevention, not only severe accident mitigation. To avoid consequences including life-threatening releases, evacuation, contamination of public property, and erosion of public confidence, the Commission noted that the objective of its regulatory programme would continue to be “providing reasonable assurance, while giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. [NPP].” Finally, the Commission emphasized defence in depth as part of accident prevention and mitigation, including features such as containment, siting in less populated areas, and emergency planning.

These safety goals have been integrated into many agency decision making processes. In these processes, the NRC typically employs subsidiary objectives for core damage frequency (less than 10^{-4} per reactor-year) and large early release frequency (less than 10^{-5} per reactor-year) that can be calculated more easily than the quantitative health objectives. For example, they are applied in the review of licensee-requested changes that are characterized as “risk-informed” (as discussed in Regulatory Guide 1.174) [84]. In addition, the NRC conducts regulatory analyses of most of its planned actions, considering the safety or security benefits and costs that would result from the actions. As described in NUREG/BR-0058, Revision 4, [85] the safety goals provide a “safety first” test, focusing NRC activities on improvements that will provide important safety benefits.

I-22.2. Processes for evaluating potential new requirements

For NPPs, the NRC has a formal regulatory process called “backfitting” that is codified in 10 CFR 50.109 (as well as other regulations for certain other classes of licensees) [86]. When a new issue is raised that affects public health and safety or the common defence and security, backfitting requirements provide a structured approach for the NRC to consider a proposed action—specifically, imposing a changed or new requirement or staff position. When implementing backfitting requirements, the NRC is focused on physical or design changes to NPP systems, structures, or components, as well as changes to organizations or procedures for design, construction, or operation of NPPs.

In evaluating a proposed action, the default process is a backfit analysis. A backfit analysis begins with a determination whether the proposed action would provide a substantial increase in radiological public health and safety or common defence and security. If it would, the NRC continues to assess whether the cost of implementing the proposed action would be justified in light of the increase in safety or security. A backfit analysis need not be conducted if the action is associated with *adequate protection* of public health and safety or common defence and security—in these cases, the NRC has to act under the Atomic Energy Act of 1954, as amended [87], and only a documented evaluation of the issue is needed. In addition, a backfit analysis is not needed in certain circumstances when the action is needed to correct an error or omission made at the time of an earlier NRC approval (the “compliance exception,” as further discussed in a recent NRC memorandum). [88]

Examples of safety improvements required under the NRC’s backfitting requirements include:

- Orders issued after the Fukushima Daiichi NPP accident, such as Order EA-12-049 [89] on mitigation strategies, discussed further below, which were needed to provide adequate protection to public health and safety and issued under 10 CFR 50.109(a)(4)(ii);
- The station blackout rule (10 CFR 50.63) [90], which was issued in 1988 as a cost-justified substantial safety enhancement given the risk associated with a station blackout (53 FR 23203);
- The anticipated transient without scram (ATWS) rule (10 CFR 50.62) [91], which was issued in 1984 (49 FR 26036); while the backfitting requirements were changed after this rule (in 1985 and 1988), in the proposed rule, the NRC acknowledged its statutory obligation to prescribe regulations necessary “to protect health or to minimize danger to life or property” and the importance of defence in depth to achieving this objective.

More details on backfitting and the related topic of NPP licensing bases are presented in training slides [92] from Summer 2018, available on the NRC’s public website.

I-22.3. Examples of comprehensive evaluations of continued safety

I-22.3.1. Systematic Evaluation Program

In the mid-1970s, the NRC recognized the importance of assessing the adequacy of the design and operation of operating, as well as understanding the safety significance of deviations from safety standards that were approved after those plants were licensed. It also recognized the importance of providing the capability to make integrated and balanced decisions about the need for modifications at those plants. Consequently, in 1977, the NRC initiated the Systematic Evaluation Program (SEP), in which it compared the designs of 10 older NPPs to the licensing criteria delineated in the then-recently issued Standard Review Plan. After further review, the staff determined that 27 issues required some corrective action at one or more NPPs and that resolution of those issues could lead to safety improvements at other operating plants built at about the same time. Of these 27 issues, 4 were completely resolved as part of the SEP (reactor coolant boundary leakage detection, organic materials, water purity in the reactor coolant system, and containment isolation system). One issue was of such low safety significance that it required no additional action, and the remaining 22 needed no immediate action and were resolved through the NRC’s Generic Issues Program. Additional information on these evaluations is available in Generic Letter 95-04 [93] and NUREG-0933.[94]

I-22.3.2. Individual Plant Examination

The NRC and licensees also evaluated severe accident vulnerabilities at all operating plants through the Individual Plant Examination (IPE) and IPE—External Events (IPEEE) projects in the 1980s and 1990s. These projects built on Commission policy in the 1985 policy statement on severe reactor accidents (50 FR 23138) [95] that “systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements.” Licensees evaluated leading contributors to severe accident sequences and potential improvements, and the programme served as a catalyst for licensees to improve further the overall safety of NPPs. As part of the IPE, licensees identified over 500 improvements, including changes to power systems, coolant injection systems, and decay heat removal systems. Furthermore, over 90 percent of licensees identified plant improvements as part of the IPEEE, including strengthening anchorages, replacing vulnerable electrical components, and implementing transient-combustible procedures. This programme is

described further in Generic Letters issued in 1988 [96], as well as in NUREG-1560 and NUREG-1742. [97]

I-22.3.3. Containment Performance Improvement Program

The NRC also conducted a containment performance improvement program, described in Supplements 1 [98] and 3 [99] to Generic Letter 88-20 and Generic Letter 89-16 [100]. As part of this program, all licensees of boiling-water reactors (BWRs) with Mark I containments installed hardened vent capabilities, which were later updated after the Fukushima Daiichi NPP accident. In conjunction with their IPEs, licensees made further improvements to their containments. These improvements included alternate water supplies and enhanced vessel depressurization capability for BWRs with Mark I containments, additional heat removal capability for BWRs with Mark II containments, hydrogen igniter improvements for BWRs with Mark III containments and pressurized-water reactors (PWRs) with ice condenser containments, and evaluation of localized hydrogen combustion for other PWRs.

I-22.3.4. Fukushima Daiichi NPP accident response

On March 11, 2011, the Great East Japan Earthquake cut off offsite electrical power to the Fukushima Daiichi NPP, and the associated tsunami inundated portions of the plant site. Critical plant equipment flooded, causing the extended loss of onsite electrical power and the loss of reactor monitoring, control, and cooling functions in multiple units of the six-unit site. In response to the lessons learned from the accident, the U.S. nuclear industry has significantly enhanced NPP safety. Two aspects of these enhancements are discussed below; additional details of the U.S. response to the Fukushima Daiichi NPP accident can be found at the NRC website Japan Lessons Learned page. [101]

I-22.3.5. Mitigation strategies order

In March 2012, the NRC issued an order to NPP licensees requiring, in general, a three-phased approach for mitigating beyond-design-basis external events. The initial phase requires licensees to use installed equipment and resources to maintain or restore core cooling, containment, and spent fuel pool cooling. In the transition phase, licensees have to provide sufficient portable onsite equipment and consumables to maintain or restore these functions until they can be maintained with offsite equipment and support. The final phase requires licensees to obtain sufficient offsite resources to sustain those functions indefinitely.

In response to the order, the nuclear industry proposed what became known as the “FLEX” strategy—diverse and flexible mitigation strategies that would increase defence in depth for beyond-design-basis scenarios, to address a loss of all alternating current (ac) power and loss of normal access the ultimate heat sink occurring simultaneously at all units on a site. FLEX consists of the following elements:

- **Both installed plant equipment and portable FLEX equipment** that provide means of obtaining power and water to maintain or restore key safety functions for all reactors at a site. This could include equipment such as pumps, generators, batteries and battery chargers, compressors, hoses, couplings, tools, debris clearing equipment, temporary flood protection equipment and other supporting equipment or tools.
- **Reasonable staging and protection** of FLEX equipment from beyond-design-basis external events applicable to a site. The FLEX equipment would be reasonably protected from applicable site-specific severe external events to provide reasonable assurance that

there is sufficient equipment for each reactor on a site. These protections were later reconfirmed in consideration of updated hazard information discussed below.

- **Procedures and guidance to implement** FLEX strategies. FLEX Support Guidelines, to the extent possible, will provide pre-planned FLEX strategies for accomplishing specific tasks in support of Emergency Operating Procedures and Abnormal Operating Procedures functions to improve the capability to cope with beyond-design-basis external events.
- **Programmatic controls** that assure the continued viability and reliability of the FLEX strategies. These controls would establish standards for quality, maintenance, testing of FLEX equipment, configuration management and periodic training of personnel.

As of June 18, 2018, all operating power reactor units comply with this mitigation strategies order.

I-22.3.6. Seismic and flooding hazard evaluations

In March 2012, the NRC issued a [letter \[102\]](#) that requested licensees to take three actions under Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f), two of which related to seismic and flooding hazards. (The other related to emergency preparedness for multi-unit large scale events.) Specifically, licensees were asked to examine existing seismic and flood protection measures (i.e., conduct walkdowns) and re-evaluate seismic and flooding hazards at each site using present-day methods.

During the flooding walkdowns, some sites found deficiencies in their mitigation capabilities. In each case, the licensee took corrective action to remedy the deficiencies, and the NRC inspected the licensee's actions. The NRC issued [Information Notice 2015-01 \[103\]](#) to communicate the issues that were found and corrected.

The seismic and flooding hazard re-evaluation request recognized that the licensing framework has evolved over time as new information regarding site hazards and the potential consequence has become available. Each operating NPP, therefore, may have different design and licensing requirements, as well as analytical assumptions, based on its time of licensing. Therefore, given the demonstrated consequences of the Fukushima Daiichi NPP accident, the NRC found it necessary to confirm the appropriateness of the hazards assumed for U.S. plants and their ability to cope with and protect against them. All operating NPP licensees re-evaluated their hazards in response to the 10 CFR 50.54(f) letter referenced above. Based on these results, licensees have independently made safety improvements, and the NRC is determining and documenting whether any additional actions are necessary to provide additional protection against the updated hazards. For example, 13 licensees made seismic plant modifications such as replacing relays, strengthening anchorages, or removing structural interferences.

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ANNEX II. MULTINATIONAL EXPERIENCE ON IMPLEMENTING SAFETY IMPROVEMENTS AT VVER440

The purpose of this annex is to summarize the information on the practical use of individual drivers, the development of strategies, the evaluation of the design, and the implementation of projects to improve the level of nuclear safety.

For simplicity, presentations of countries operating VVER213, (Finland operating Loviisa NPP, Slovakia operating Mochovce and Jaslovské Bohunice NPPs, Hungary operating Paks NPP, and Czech Republic operating Dukovany NPP) were selected.

PSR, lessons learned from significant events, and PSA studies were selected as safety improvement drivers for the NPPs.

II-1.1. Severe accident management implementation

A common starting point for all NPPs was activities after TMI NPP and Chernobyl NPP accidents to include in the VVER 440 defence in depth level 4 (severe accident management (SAM)) and requirements for SAMG development. This was the starting point for analyses for SAM phenomena and common projects like PHARE, VERSAFE. The leader for SAM strategies implementation was Finland. Results of analyses showed that basic design of VVER213 could be modified for SAM and based on the final results, a common severe accident management strategy was developed: see Table II-1.

TABLE II-1. COMMON SEVERE ACCIDENT MANAGEMENT STRATEGY

| | Base case | Strategy I | Strategy II |
|---|------------------|------------------------|---|
| Prevention of RPV failure | ECCS recovery | ECCS recovery | ECCS recovery + reactor cavity flooding |
| Hydrogen treatment | - | 30 recombiners | 30 recombiners |
| Limitation of radioactive releases | Spray recovery | Spray recovery | Spray recovery |
| Prevention of containment overpressurization | - | Filtered venting | Filtered venting or NOT for in vessel phase |
| Safe integrity of the reactor cavity | - | Isolation of room A004 | Solved by cavity flooding |
| (External cooling of the molten material) | - | - | (Not challenged) |

The target of the accident management is the overall capability of the plant to respond to and recover from a severe accident situation. This capability is increased by hardware modifications and with a guide to use the available resources in an optimal way.

After creation of strategy all NPPs were ready to develop SAMGs connected to existing Symptom Based Emergency Operating Procedures (SBEOPs). In addition, new implemented systems and procedures, the new SAM control system (located in MCR, ERC, or new room) was incorporated in to existing SAMGs.

Strategies calculated with BD HW modification and driver for its implementation was:

TABLE II-2. D

| | Finland | Slovakia | Hungary | Czech |
|---------------|----------------------|-----------------|----------------|--------------|
| Driver | Regulator request | PSR | PSA L2 / LTO | PSR |

HW modifications necessary for mitigation of severe accidents agreed by national regulatory bodies are shown in Table II-3.

TABLE II-3. HW MODIFICATIONS NECESSARY FOR MITIGATION OF SEVERE ACCIDENTS.

| Country /Implementation of Safety Upgrading Measures | Finland | Slovakia | Hungary | Czech |
|--|----------------|-----------------|----------------|--------------|
| External coolability of RPV – IVR strategy | yes | yes | yes | yes |
| Controlled depressurization of RCS in the onset of severe accident | yes | yes | yes | yes |
| Containment Hydrogen management - Installation of PARs | yes | yes | yes | yes |
| Containment Vacuum breaker | - | yes | - | yes |
| Alternative coolant system – RPV corium flooding | - | yes | - | yes |
| Alternative coolant system – inside containment spray | - | yes | design | yes |
| Containment spray (external) | yes | - | - | - |
| Alternative coolant system -Spent fuel flooding | | yes | yes | yes |
| Alternative electric power system | yes | yes | yes/design | yes |
| I&C capabilities needed for severe accident management | yes | yes | yes | yes |
| Long term heat removal – to ultimate heat sink | yes | yes | design | yes |
| Venting system | - | - | - | - |
| Ice condenser (mixing of containment atmosphere, temperature and pressure control) | yes | - | - | - |
| Hydrogen igniters | yes | - | - | - |
| Containment isolation enhancement | yes | yes | yes | yes |

Implementation of SAM safety mitigation measures depend on NPPs specifics, results of analyses and strategies.

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