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Joint European Canadian Chinese development of Small Modular Reactor Technology

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Abstract:

This report collects the most important generic and design specific safety criteria and requirements judged to be applicable for the conceptual design developed within the ECC-SMART project. This has been done through a critical review of the relevant documents published by IAEA, WENRA, GIF and national legislations of Finland, Canada, Czech Republic, UK and China. The driving principles and main approaches for the refinement and further development of the safety requirements and criteria have also been identified, substantiated and documented.

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Executive Summary

The purpose of this deliverable is to identify generic safety principles, objectives and criteria (hereafter simply named 'safety elements') and specific safety elements applicable to ECC-SMART.

The driving principles for the application of generic safety elements are discussed and substantiated in Section 2. These are then implemented in the review of safety elements applicable to all ECC-SMART family of designs at the international level, including IAEA, GIF and European Commission (EC), in Section 3. Section 4 discusses safety requirements at the national level in Canada, China, the Czech Republic, Finland and the United Kingdom. Finally, Section 5 closes the report by summarizing the identified safety elements applicable to Supercritical Water-Cooled Small Modular Reactor (SCW-SMR) designs and provides recommendations for the refinement and further development of the safety requirements and criteria with the future developments of the respective conceptual designs.

In this deliverable, it was assumed that the ECC-SMART family of designs results from the combination of three big nuclear power plant families of designs: new (3rd generation) reactors, small modular reactors and 4th generation reactors including the Supercritical Water-Cooled Reactor (SCWR). Such a combination is the conjunction of the three sets of safety elements as long as compatible with ECC-SMART.

The determination of safety elements of ECC-SMART is made based on: (1) the identification of existing safety elements based on a structured literature review and applicability analysis and (2) the development of safety elements from identified gaps related to specific fields of safety where accommodation of existing safety elements is not feasible. A critical review of the relevant documents of international legal or guidance frameworks (IAEA, GIF, EC) and national legislations (Canada, China, Czech Republic, Finland and UK) has been performed.

In the summary of specific safety elements for SCW-SMR, guiding principles for review and selection of relevant safety elements, safety elements and recommendations are summarized.

All compatible safety elements of new 3rd generation reactors apply to ECC-SMART, while safety elements specific to small modular reactor (SMR) designs apply to ECC-SMART depending on the design, hence on a case-by-case basis. All the top-level safety requirements identified by GIF apply to ECC-SMART, in particular, the elimination of DEC-B scenarios. No new challenging safety requirements specific to SCW-SMR in the accident domain are foreseen or expected.

No safety elements developed specifically for the SCW-SMR have been found. The majority of the IAEA high-level safety requirements apply to the ECC-SMART. IAEA SSR-2/1 standard for design should be used during the SCW-SMR development as a starting point for the design criteria development. As the SCWR is among the selected Generation IV International Forum (GIF) designs, ECC-SMART should meet the relevant GIF safety goals. In summary, the frequency of DEC-A scenarios shall be very low and its damage extension limited and DEC-B scenarios shall be eliminated. Also, with the exception of Canada and China, no specific regulations for SMRs appear to be in force in the countries reviewed within this project. Specific requests for SCW-SMR are also not yet available in the Chinese and Canadian SMR regulations.

This is why safety criteria covering particular aspects of the ECC-SMART conceptual designs will have to be developed in parallel with the specific design features featured by the designs under ECC-SMART.

Some relevant aspects to be considered in the future refined conceptual design and in the parallel development of safety criteria, respective research and development (R&D) programmes and Phenomena Identification and Ranking Table (PIRT), are already identified in the deliverable. However, it is recommended to carry out an exhaustive identification of the specific design and performance features in the future. Within the scope of this project, the necessary preparations to develop the PIRT are being planned to support the implementation of the Integrated Safety Assessment Methodology (ISAM) in future projects.

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List of acronyms and abbreviations

AFCEN	Association Française pour les règles de Conception, de construction et de surveillance en exploitation des matériels des Chaudières Electro Nucléaire
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BEPU	Best Estimate Plus Uncertainty
BOP	Balance Of Plant
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CNSC	Canadian Nuclear Safety Commission
DBA	Design Basis Accident
DBC	Design Basis Conditions
DEC	Design Extension Conditions
DiD	Defence-in-Depth
DPA	Deterministic and Phenomenological Analyses
EC	European Commission
ECCS	Emergency Core Cooling System
ECC-SMART	(Joint) European Canadian Chinese development of Small Modular Reactor Technology
ENISS	European Nuclear Installations Safety Standards
EPZ	Emergency Planning Zone
EUR	European Utility Requirements
FINSP	FINnish Support Programme
FOM	Figure-Of-Merit
GDI	Guidance Document for Integrated Safety Assessment Methodology (ISAM)
GIF	Generation IV International Forum
HPLWR	High Performance Light Water Reactor
HTGR	High Temperature Gas-cooled Reactor
IAEA	International Atomic Energy Agency
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INSAG	International Nuclear Safety Advisory Group
ISAM	Integrated Safety Assessment Methodology
IVR	In-Vessel Retention
JAEA	Japan Atomic Energy Agency
JSFR	Japanese Sodium-cooled Fast Reactor
JRC	Joint Research Centre
KTA	Kerntechnischer Ausschuss (Nuclear Safety Standards Commission)
LFR	Lead-cooled Fast Reactor
LOCA	Loss-Of-Coolant Accident
LOP	Line Of Protection
LoERF	Large and/or Early Release Frequency

LP&S	Low Power and Shutdown
LWR	Light Water Reactor
LW-SMR	Light Water SMR
MEAE	Ministry of Economic Affairs and Employment
NNSA	National Nuclear Safety Administration
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OECD/NEA	Organisation for Economic Co-operation and Development/Nuclear Energy Agency
OPT	Objective Provision Tree
PCD	Pre-Conceptual Design
PCT	Peak Cladding Temperature
PIRT	Phenomena Identification and Ranking Table
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
QSR	Qualitative Safety Features Review
RCC-E	Règles de Conception et de Construction des matériels Electriques des îlots nucléaires REP (in English 'Design and construction rules for electrical equipment of PWR nuclear islands')
RCC-M	Règles de Conception et de Construction des matériels Mécaniques des îlots nucléaires REP (in English 'Design and Construction Rules for the Mechanical Components of PWR Nuclear Islands')
RCP	Reactor Coolant Pump
R&D	Research and Development
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RSWG	Risk and Safety Working Group
SA SDG	Safety Design Guidelines on Safety Approach and Design Conditions
SASS	Self-Actuation Shutdown System
SCWR	Supercritical Water-cooled Reactor
SCW-SMR	Supercritical Water-cooled Small Modular Reactor
SDC	Safety Design Criteria
SDC-TF	Safety Design Criteria Task Force
SDG	Safety Design Guidelines
SMR	Small Modular Reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SSA	Supercritical-Water-cooled Reactor (SCWR) System Safety Assessment
SSC SDG	Safety Design Guidelines on Structures, Systems and Components
SSG	Specific Safety Guide
SSR	Specific Safety Requirements
STUK	Säteilyturvakeskus (Finland's Radiation and Nuclear Safety Authority)
SÚJB	Státní úřad pro jadernou bezpečnost (State Office for Nuclear Safety of the Czech Republic)
U.S.	United States
VDNS	Vienna Declaration on Nuclear Safety

VTT	Technical Research Centre of Finland
VVER	Vodo-Vodyanoi Energetichesky Reactor (Water-Water Energetic Reactor, WWER)
YVL	Regulatory Guides on Nuclear Safety and Security
WENRA	Western European Nuclear Regulators Association
WP	Working Package

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1 Introduction

1.1 Background

The Joint European Canadian Chinese development of Small Modular Reactor Technology (ECC-SMART) is a multi-international project focused on the development and licensing of the future Supercritical Water-cooled Small Modular Reactor (SCW-SMR).

In this document we assume that the ECC-SMART family of designs results from the combination of three large nuclear power plant (NPP) families of designs: new (3rd generation) reactors, SMRs, 4th generation NPPs including the SCWR. Such combination is the conjunction of the three sets of safety elements as shown with the help of Venn's diagrams in Figure 1 as long as compatible with ECC-SMART.

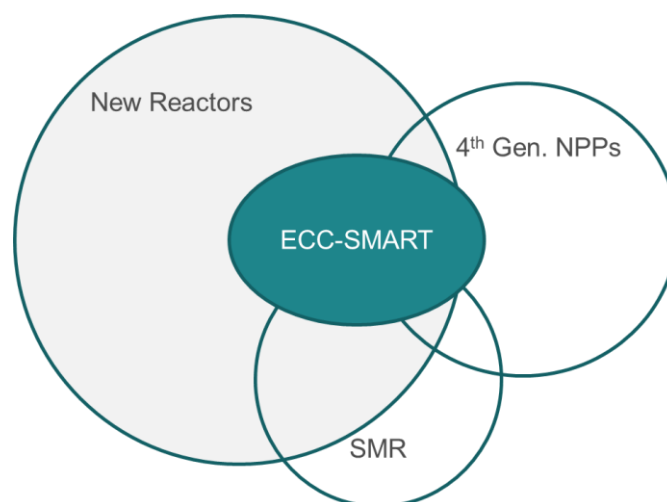


Figure 1. Safety Elements applicable to ECC-SMART

ECC-SMART takes the family of new reactors (with light grey colour) as the reference family of designs against which it departs from, and where most of the safety elements applicable to ECC-SMART are located (hence the bigger circle compared to the others). This absolute reference to analyse the applicable safety elements to ECC-SMART dwells on the fact that the new plants have already achieved a consolidated, agreed and robust set of safety elements, so that in principle no further research and development is required for successful licensing. The collection of safety elements applicable for new plants is well reflected by many international organisations, such as EC, IAEA and WENRA, and national regulatory frameworks. The current project therefore addresses the task of identifying safety elements to ECC-SMART through a biased approach based on first stating the common safety elements with so-called new plants applicable to ECC-SMART, and then stating the differences that need dedicated application.

1.2 Purpose of the document

This document is the first among the four deliverables to be developed within the WP5 Licensing and safety of the ECC-SMART project. It documents the work performed in the WP5 task 5.1 and aims at two major objectives:

- Identification of generic safety principles, objectives and criteria (hereafter simply named 'safety elements') applicable to ECC-SMART.
- Identification of specific safety elements applicable to ECC-SMART.

The information collected and developed for this document will be further developed, complemented and refined in the follow-up deliverables:

- D5.2 Safety related features of the SCW-SMR concept, aiming at synthesizing the main safety related findings and conclusions of the WPs 2-4;
- D5.3 Pre-licensing study; aiming to demonstrate the feasibility of the conceptual design to be licensed;
- D5.4 Guidelines for the demonstration of the safety in the further development stages of the SCW-SMR concept.

1.3 Safety of nuclear installations

It is assumed throughout the document that the reader is familiar with the top-level concepts and principles of nuclear safety including Defence-in-Depth (DiD), fundamental safety functions and principles (control the chain reaction, heat removal, containing the radioactive materials), resistance to single failure, redundancy, diversity, independence, etc.

Nevertheless, it appears to be useful to depict some definitions of levels of defence, which are at the core of diverse national and international implementations of the DiD concept. The general idea of the DiD concept is discussed in the document “Defence in Depth in Nuclear Safety” [1] published in 1996 by the International Nuclear Safety Advisory Group (INSAG). This document introduces five levels of DiD, outlined in Table 1 and is consistent with the definitions of the five levels of DiD in IAEA SSR-2/1 [2].

Table 1. Levels of DiD according to INSAG-10 [1]

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

The Western European Regulators Association (WENRA) called, for new plants, towards reinforcement of DiD, and for the improvement of the independence between the levels of DiD. As a result, WENRA derived a refined structure of the DiD levels [3] depicted in Table 2. Even though no new level of defence was suggested, a subdivision of the level 3 was created, with a clear distinction between means and conditions for sub-levels 3a and 3b was proposed.

Table 2. The refined levels of DiD according to WENRA [3]

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

As a rough approximation, the first three levels of DiD were traditionally considered as anticipated events, transients, hazards and accidents, termed also plant “design conditions”, which were together with the respective safety requirements, defined in the plant “design bases”. The licensing process required to demonstrate successful defence against the threats collected in the design bases.

The events, transients, hazards and accidents leading to Levels 4 and 5 were traditionally termed “beyond design”. However, they received augmented attention after the Fukushima-Daiichi accident resulting in:

- Design Extension Conditions (DEC-A) without core melt. These may include for example multiple failure events, which were traditionally excluded from the design bases. The goal here appears to be in control of escalation of accidents to prevent core melt. This relates DEC-A with Level 3 (INSAG, Table 1) and Level 3b (WENRA, Table 2).
- Design Extension Conditions (DEC-B). These include mitigation of anticipated core melt event, especially mitigating the (large early) off-site releases of radionuclides. This relates the DEC-B with Level 4 (INSAG, Table 1, and WENRA, Table 2).

WENRA allows for a graded approach, including probabilistic assessment, while demonstrating defence against level 3b and level 4 events.

1.4 Security and safeguards of nuclear installations

At this time the conceptual design of the ECC-SMART does not appear to open any specific security and safeguards challenges as compared to other 3rd and 4th Gen reactors. In addition, the SMRs with decisively smaller inventories of radionuclides than typical large 3rd and 4th Gen reactors appear to pose also comparably smaller security and safeguards risks.

As a consequence, this document focuses predominantly on the requirements for nuclear safety.

It is recommended to review these assumptions regularly with the future developments of the ECC-SMART Conceptual Design.

1.5 Organisation of the document

The driving principles for the application of generic safety elements are discussed and substantiated in Section 2. These are then implemented in the review of safety elements applicable to all ECC-SMART family of designs at international level, including IAEA, GIF and European Commission, in Section 3. Section 4 discusses safety requirements at national level in Canada, China, Czech Republic, Finland and United Kingdom.

Section 5 closes the report by summarizing the identified safety elements applicable to SCW-SMR designs and provides recommendations for the refinement and further development of the safety requirements and criteria with the future developments of the respective conceptual designs.

2 Guiding Principles for the application of Generic Safety Elements in ECC-SMART

The SCW-SMR designs comprised by ECC-SMART (hereafter referred to as 'ECC-SMART') includes some specific design features, e.g. high-pressure tubes versus reactor pressure vessel (RPV). This is why specific safety elements identified for ECC-SMART should be further broken down into design-specific safety elements. However, this is out of this report's scope (but would become necessary if a concrete SCW-SMR design becomes available).

The determination of safety elements of ECC-SMART is made up of two steps:

- 1) Identification of existing safety elements based on a structured literature review and applicability analysis.
- 2) Development of safety elements from identified gaps related to specific fields of safety where accommodation of existing safety elements is not feasible.

This two-step process is informed by four guiding principles as subsequently described.

2.1 First Guiding Principle derived from common aspects with New Reactors (3rd Gen. NPPs)

Safety elements at general level mainly derive from the plant design and performance, on one hand, and from the characterization of the hazard sources, on the other: if two different nuclear systems share these aspects, top-level safety approaches allow similar application.

Similar Design / Hazard / Performance → Similar general safety elements

Furthermore, similar designs lead to similar accident phenomena and plant response, which in turn leads to similar safety systems.

Similar Design / Hazard / Performance → Similar accident phenomena & plant response → Similar safety systems

The **first guiding principle** for safety application in ECC-SMART states:

All compatible¹ safety elements of new reactors apply to ECC-SMART.

The underlying common **fields of design and performance** allowing the application of a similar safety approach stem from the following similarities²:

- Fluid systems;
- Nuclear steam supply system (NSSS) and balance of plant (BOP) layout and interface;
- Key safety functions;
- Representative threats (initiating events, accident categories).

¹ Within the current context, 'compatible' is intended as applicable to ECC-SMART in terms of design and performance.

² 'Similarity' does not mean 'equivalency', in fact, emphasizing that many aspects regarding each of these fields of the plant design and performance match between the design families, e.g. nuclear reactors, and ECC-SMART, hence at the same time differing in some others.

The underlying common **fields of hazard sources** allowing the application of a similar safety approach stem from the following similarities:

- Encapsulated fuel material in solid state;
- Arrangement of physical barriers between the toxic, i.e. nuclear and chemical, hazard, and the people and environment: fuel pellet, fuel rod, reactor coolant pressure boundary, containment;
- Hazard thermodynamic and chemical response;
- Hazard location.

Depending on the ECC-SMART specific design, e.g. design with a RPV or a pressure tube design, a different compatibility, as set of generic safety criteria, applies.

Differences between new reactors and ECC-SMART are mostly quantitative: the phenomena falling under the accident domain are the same, i.e. differences are more on magnitude rather than on different figures of merit. For instance, limits on cladding oxidation or peak cladding temperature (PCT) should hold yet perhaps at different thresholds; core melting also applies with the same definition than for reference plants; containment thermal-hydraulics and challenging phenomena only differ quantitatively; emergency core cooling system (ECCS) should play the same role against loss-of-coolant accidents (LOCAs); loss of residual heat removal (RHR) system in low power and shutdown (LP&S) conditions as initiating events apply for new reactor designs; etc.

2.2 Second Guiding Principle derived from common aspects with SMRs

Despite the broad spectrum of SMR designs, all of them share some basic features:

- Modularity;
- Power less than 300 MW_e;
- Passive safety systems.

These three factors might be potentially³ beneficial for safety, as they should lead to higher system reliability; lower hazard magnitude thereby lower potential radioactive / toxic releases; and lower dependency of key safety functions on supporting systems or actions respectively.

On one hand, SMRs share many aspects with existing and new reactors – though strongly depending on the SMR design, at least when dealing with light water (LW)-SMRs. On the other hand, the plant response may depart from the new plants because of substantial modifications due to different key ratios – e.g. containment volume versus thermal power– or passive safety performance.

The **second guiding principle** on safety application in ECC-SMART states that:

Safety elements specific to SMR designs apply to ECC-SMART depending on the design, hence on a case-by-case basis. Safety demonstration dealing with passive safety systems performance and plant accident response need conservative safety margins.

³ 'Potentially' because the majority of SMRs still have to provide demonstration of compliance with more stringent safety objectives consistently derived from these features.

Higher uncertainties due to limited operating experience dealing with SMRs and passive safety systems should be offset with larger safety margins. Experimental testing should provide enough evidence to support the use of analytical tools involved in any qualification activity and safety demonstration.

2.3 Third Guiding Principle derived from common aspects with 4th Gen. NPPs

4th Gen. NPPs constitute a family of innovative designs specifically aimed at solving some of the most relevant issues of existing NPPs. It is precisely this problem-solving feature, shaping and selecting the viable designs, where to invest research and development efforts and justifying their innovative nature.

Because of their goal-based design, 4th Gen. NPPs already come with a set of nuclear safety elements in all aspects dealing with nuclear energy production, such as sustainability, proliferation or safety.

4th Gen. NPPs present substantial differences both in design and performance – contrary to 3rd Gen. NPPs – preventing the application of similar specific safety requirements. This is why safety elements of 4th Gen. NPPs are necessarily generic, while specific safety elements only apply for each design under the family of 4th Gen. NPPs.

Given the more ambitious nature triggering 4th Gen. NPPs, predetermined goals imposed for all the designs are equal or more challenging than those of new plants or SMRs.

Safety elements for 4th Gen. NPPs mainly come from OECD/NEA-sponsored Generation IV Forum (GIF). The SCWR is one of the selected designs by GIF. GIF safety principles therefore apply to ECC-SMART, among which one in particular, if reformulated, leads to the elimination of DEC⁴-B scenarios, as further discussed in Section 3.2 (refer to [2] for further information on categorization of plant states or design basis conditions).

The **third guiding principle** is stated as follows:

All the top-level safety requirements identified by GIF apply to ECC-SMART, in particular, the elimination of DEC-B scenarios.

2.4 Fourth Guiding Principle derived from specific aspects of ECC-SMART

The key design differences between SCWRs and 3rd Generation reactors are the followings:

- Higher energy conversion efficiencies;
- Higher operating temperatures and pressures;
- Differences in core design, e.g. two-pass / three-pass core configurations;
- No phase change (in normal operating conditions) compared to BWRs;
- Absence of separators and dryers compared to BWRs;
- Absence of reactor coolant pumps (RCPs) compared to BWRs.

These key design differences lead to the following key design aspects departing from existing reactors in normal operating conditions:

- Heat transfer correlations;
- Harsher corrosion environment;

⁴ Design extension conditions

- Changes in the water physicochemical properties;
- Neutron physics parameters and neutron – thermal-hydraulic coupling;
- Harsher mechanical loads on structural materials.

These key design differences lead to the following key design aspects departing from existing reactors in accident conditions (DBC⁵ and DEC):

- Refilling / Reflooding patterns;
- Higher primary system depressurization rate in the RPV;
- Higher containment pressurization rate;
- Stronger High Pressure Melt Ejection;
- Stronger Direct Containment Heating;
- Stronger attack to the RPV lower plenum in case of corium relocation at high pressure.

Except for the lack of phase change under normal operating conditions, the list of phenomena only departs quantitatively and not qualitatively from existing reactors, in line with what is stated at the end of Section 2.1. As for the listed phenomena above, quantitatively more challenging phenomena compared to reference plants', each of them should be addressed in detail in the ECC-SMART to check that they are practically eliminated, in consistency with safety principles for new reactors, as stated in, e.g. [2], [3], and [4].

The **fourth guiding principle** for safety application in ECC-SMART states that:

No new challenging safety requirement specific of ECC-SMART in the accident domain is foreseen. Quantitatively more demanding phenomena affecting safety need robust safety demonstration and conservative safety margins.

⁵ Design basis conditions

3 International legal frameworks for nuclear safety, security and safeguards

This section aims at reviewing the existing safety elements applicable to ECC-SMART and assigning the applicability scope of their elements upon the guiding principles as set forth above⁶.

- Safety elements belonging to new reactors apply fully to ECC-SMART, as derived from the first principle – abbreviated as 1P.
- Safety elements dealing with SMRs apply on a case-by-case basis depending on the design, as derived from the second principle – abbreviated as 2P.
- Generic safety elements dealing with 4th Gen. NPPs apply fully as derived from the third principle – abbreviated as 3P.
- Specific safety elements dealing with ECC-SMART come from the fourth principle – abbreviated as 4P.

There are four key sources of information concerning international legislation or guidance applicable to nuclear reactors: IAEA, OECD/NEA GIF, European Union legal acts and international fora at European level. They are further discussed below.

3.1 IAEA

IAEA safety dispositions are arranged at hierarchical order in terms of both relevance – from higher to lower – and scope of application – from more to less general –, at the top the fundamental safety principles, followed by requirements and concluding on guidance, the first two levels are mandatory whereas the latter are recommendations. Below this level, a set of technical documents provide further insight into specific topics, e.g. Safety Reports, Technical Documents (so called TECDOCs), etc.

The majority of the documents concerning safety of nuclear installations for energy production are either technology-neutral or targeted to specific reactor coolants (i.e. water or gas). Several documents also deal with SMRs and 4th Gen. designs. There is no IAEA document dedicated specifically to safety aspects of SCWRs⁷.

The main applicable IAEA safety reports with their scope of application to ECC-SMART are listed in Table 3.

⁶ Please note there is no need of reproducing the safety elements here since they are available at the source.

⁷ The IAEA has published a number of TECDOCs on specific technical issues of the SCWR, e.g. thermal-hydraulics and modeling of SCW flow phenomena, fuel cladding and reactor materials and water chemistry of SCWRs.



Table 3. Main applicable IAEA safety standards and reports with their scope of application to ECC-SMART

IAEA REPORT	CLASSIFICATION	APPLICATION	COMMENTS
Fundamental Safety Principles, Safety Fundamentals, SF-1 [6]	1P	Full application	Top-level principles apply to ECC-SMART fully.
Safety of Nuclear Power Plants: Design, Specific Safety Requirements, SSR-2/1 [2]	1P	Full application	Top-level requirements apply to ECC-SMART fully.
Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment, TECDOC-1936 [7]	2P and 3P	Full application	The LWR part requirements apply to ECC-SMART fully.
Design and Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident, TECDOC-1785 [8]	2P and 3P	Partial application	Applicability of lessons learned and recommendations on passive safety systems depends on the ECC-SMART design.
Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants, Specific Safety Guide, SSG-56 [9]	1P	Full application	Guidelines to fulfil the top-level requirements drawn from SSR-2/1.
Design of the Reactor Containment and Associated Systems for Nuclear Power Plants, Specific Safety Guide, SSG-53 [10]	1P	Full application	Guidelines to fulfil the top-level requirements drawn from SSR-2/1.
Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants, Specific Safety Guide, SSG-62 [11]	1P	Full application	Guidelines to fulfil the top-level requirements drawn from SSR-2/1.
Protection against Internal Hazards in the Design of Nuclear Power Plants, Specific Safety Guide, SSG-64 [27]	1P	Full application	Guidelines to fulfil the top-level requirements drawn from SSR-2/1.



3.2 Review of GIF guidance with respect to SCWR

As it is well known, the Technology Roadmap of Generation IV International Forum (GIF) identifies four technology goals for the six different types of Generation IV nuclear energy systems:

- the first is enhanced safety and reliability,
- the second is improved economics,
- the third is enhanced sustainability,
- while the fourth is the strengthening of proliferation resistance and physical protection.

It is worth to note, that both the plant physical protection and the operational reliability have been significantly improved from the level of Generation III and III+ nuclear systems. Consequently, the focus of further improvement for Generation IV nuclear systems falls primarily on the economics, the safety, the sustainability, and the proliferation resistance. Additionally, after the Fukushima Daiichi accident in 2011, the improvement of plant safety is the prime focus in order to secure the public confidence and acceptance of nuclear energy.

3.2.1 Assessment of Generation IV Goals [12]

The SCWR is among the selected GIF designs. As such, ECC-SMART should meet the GIF safety goals. In particular, GIF Safety and Reliability 2 goal states [12]:

“Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.”

GIF Safety and Reliability 3 goal states:

“Generation IV nuclear energy systems will eliminate the need for offsite emergency response.”

First goal looks at significantly reducing both the probabilistic and deterministic aspects of DEC-A scenarios whereas the second goal looks at preventing DEC-B scenarios. Reformulating these goals:

The frequency of DEC-A scenarios shall be very low and its damage extension limited.

This first GIF goal on safety is flexible enough not to be translated into higher demanding safety criteria and / or goals for Generation IV designs in general and ECC-SMART in particular. In fact, it could be argued that current plants already feature low core damage frequency (CDF) values. The same applies to the damage extension concept embedded in DEC, which is limited to whether dedicated systems meant to cope with DEC events perform as expected – just like with new plants. Therefore, this GIF criterion, even if looking at 4th Gen. NPP designs, is actually shared with new reactors, and hence falls under 1P applicability.

Furthermore, DEC-A frequency depends upon two aspects: frequency of initiating events and failure probability of safety systems in bringing the plant to a safe state. Low frequencies mean low frequency of initiating events and / or reliable plant response in overcoming the threat posed by the initiating event.

Concerning the initiating events, external and internal hazards for ECC-SMART do not significantly differ from reference plants. As for the internal events, they depend on the reactor

design and working operating conditions. In terms of design, ECC-SMART should feature lower parts and a lower number of support systems subject to failure; however, higher working pressures and temperatures may lead to more frequent initiating events.

Concerning the safety systems responding to the threat posed by the initiating event, passive safety means characteristic ECC-SMART designs should decrease the number of components whose performance is necessary to fulfil the safety function. Nevertheless, passive devices bring a new source of dependence on the boundary conditions necessary for the driving force under the laws of physics to perform as expected. If these boundary conditions significantly depart from a certain range, e.g. higher heat losses, higher amount of non-condensable gases, etc., the driving force, e.g. condensation flow rate, can be impaired. This is why safety criteria depending directly or indirectly on passive safety features shall undertake higher safety margins in order to compensate for the lower operating experience in this field.

Based on the arguments given above, no conclusion on CDF values of ECC-SMART compared to reference plants can be derived. Since this criterion does not lead to higher demanding safety criteria or objectives, it can be argued that ECC-SMART is already in the position of meeting it.

As for the second reformulated objective:

DEC-B scenarios shall be eliminated.

This safety principle is even stronger for the new plants as reflected in [3], etc., where only so-called Low Early Release (LoERF) categories have to be eliminated. Instead, 4th Gen. NPPs shall eliminate all *significant* radioactive releases. The applicability scope of this safety principle is 3P.

Accidents leading to *significant* offsite radioactive releases – high enough to achieve dose thresholds so that countermeasures to protect people and environment must be deployed – fall under the field of severe accidents. If no need for any offsite emergency countermeasure, the distances taken by new reactors to quantitatively convert statements such as to limit countermeasures ‘in space’, usually taken 3 to 5 km away from the plant, apply at the site boundary in ECC-SMART (as 4th Gen. NPP), i.e. approx. 0.2 km away from the radioactive source. This drastic reduction of the distance where radioactive releases must be kept below dose thresholds can only be achieved if any DEC-B whatsoever is eliminated, since any unmitigated (i.e. unstopped) DEC-B able to progress will highly likely lead to offsite dose values exceeding dose reference values triggering countermeasures at 0.2 km away from the source.

There are only two possible design paradigms to eliminate all DEC-B scenarios: (1) integrity of the latest static and dynamic confinement barriers and minimization of driving force for uncontrolled radioactive releases; (2) no DEC-A events able to take place.

As for (1), if the last static confinement barrier, e.g. usually containment, keeps its integrity, and the pressure difference with respect to the outside environment is kept low, offsite radioactive releases may be low enough so that no cumulative dose threshold is ever reached at the site boundary. In order for the last confinement barrier to hold the energy stored by the fuel without exceeding the safety limits, usually maximum failure pressure, temperature and corrosion, while keeping the pressure low, a set of mechanisms shall be put in place, such as heat removal outside the containment. These mechanisms shall feature nearly full reliability, i.e. there shall be no

credible scenario making them fail⁸. Therefore, this means that (1) points at so-called ‘inherent safety’ linked to DEC-B: **the last static confinement barrier and support systems** are such that there is no credible scenario, i.e. chain of events, leading to the loss of the last static barrier. Full reliability of safety systems belonging to the last onsite DiD barrier. This design paradigm relies also on nearly fully trusting the dynamic safety systems once DEC-A has already taken place: **nearly full reliability of dynamic safety systems belonging to the 4th level of DiD**.

As for (2), the second way of achieving inherent safety linked to DEC-B is that **DEC-A scenarios are practically eliminated**. This is based either (i) also on nearly full reliability of dynamic safety systems belonging to the 3rd level of DiD, or (ii) because of the physicochemical properties of the hazard sources that naturally prevent any significant radioactive release, irrespective of the condition of the 3rd and 4th static and dynamic safety barriers.

Both (1) and (2) are different ways of fulfilling the inherent safety concept linked to DEC-B (and offsite consequences). It is worth noting that if inherent safety relies on dynamic passive safety systems, they shall compensate their lack of reliability of the passive driving force through different mechanisms such as redundancy, very large grace periods for recovery systems, active portable equipment, etc. This is in line with a more robust safety demonstration of SMR passive safety means (see the second guiding principle).

3.2.1.1 *In conclusion:*

- The first GIF safety goal does not impose any extra burden on ECC-SMART.
- The second GIF safety goal is transposed into challenging safety criteria dealing with the last static and dynamic safety barriers or physicochemical properties of the nuclear hazards. If relying on passive safety systems, compensation for the lack of reliability through redundancy, diversity or large grace periods should constitute an additional safety criterion.

3.2.1.2 *Further reviewed GIF materials:*

Next to the online available GIF information [12], the following five selected GIF documents were reviewed in order to collect further SCW-SMR (ECC-SMART) relevant safety criteria and requirements:

- Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System (Rev. 1) (2017) [13];
- Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-cooled Fast Reactor Systems (Rev. 1) (2019) [14];
- An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems (Version 1.1) (2011) [15];
- Super-Critical Water-cooled Reactors (SCWR) Risk and Safety Assessment White Paper (Rev. 1) (2017) [16];
- Supercritical-water-cooled reactor system (SCWR) System Safety Assessment (2018) [17].

⁸ ‘Nearly full’ leaves the door open to residual risk, in recognizing that plant response is always against a set of scenarios that, even if extreme, still do not comprise all possible scenarios, indeed, only those credited as reasonable. Nonetheless, the list of credible scenarios has been significantly expanded to comprise complex sequences and DEC-A and DEC-B, including cliff-edge effects, which make the list become wide enough to have high confidence in having captured even the most extreme –yet reasonable– events.

All safety criteria and requirements were selected and mentioned in the following five subchapters which are directly applicable or applicable after adaptation to SCW-SMR designs.

3.2.2 Safety Design Criteria for Generation IV SFR document [13]

The document „Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System“ [13] is the second highest level document in the regulation system created by GIF (see Figure 2). On the top of the hierarchy of safety standards there are the general safety goals (SG) representing fundamental safety principles and common safety goals for GIF reactors (discussed in the previous sub-chapter). These principles reflect the basic fundamental requirements of the IAEA [18] and the safety goals of the GIF.

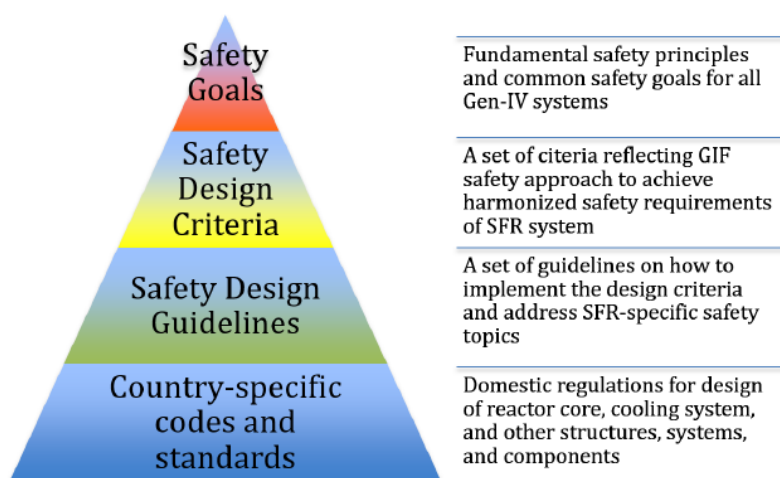


Figure 2. Hierarchy of Safety Standards of GIF [13]

The “Safety Design Criteria” (SDC) document is on the second level of this hierarchy. It has been developed by the safety design criteria task force of the GIF in 2017 for one selected GIF design (Sodium-cooled Fast Reactors, SFRs). The document comprises a set of design criteria in order to establish a harmonised requirement system for all SFRs. (It is worth to mention that recently the document Safety Design Criteria for LFR has been published as well.)

The next level of the hierarchy is for the safety design guidelines (SDG). These are more technical documents establishing a set of requirements for different fields in order to meet the criteria set up in SDC. So far, two different guidelines have been released by the GIF:

- Safety Design Guidelines on Safety Approach and Design Conditions (SA SDG) [14];
- Safety Design Guidelines on Structures, Systems and Components (SSC SDG) [19].

On the lowest level of the hierarchy the national codes and standards can be found. These country-specific standards regulate the design of the different systems in details.

During the review the criteria and requirements of the SDC document have been evaluated and compared with the IAEA SSR-2/1 [2]. The results of the review can be seen in Annex 1. The colour code shows, which requirements could be fitted without modification into a possible high level SCW-SMR regulation document. The comments are available in Annex 2.

3.2.2.1 Main findings of the review:

1. The basis of the development of high level SFR-related regulation is the IAEA SSR-2/1 document (Safety of Nuclear Power Plants: Design) [2]. The 82 criteria and the connecting requirements of the [2] were evaluated by the GIF Task Force⁹ one-by-one and modified, if necessary. However, this modification is largely based on the special features of SFR reactors (such as the special requirements concerning the high fast neutron flux, the problems with possible reactivity transient, or the issue of possible interaction between sodium coolant and water). **This means that most of the modifications made by the GIF Task Force cannot be accepted for SCWR regulation.** The SDC document therefore cannot be treated as a basis for the development of design criteria for SCWR/SCW-SMR but as a good example how the IAEA standards and guidelines can be applied for a Generation IV system through suitable adaptation of IAEA documents.

Recommendation: IAEA SSR-2/1 [2] should be used during the SCW-SMR development as a starting point for the design criteria development.

2. The SDC document has been developed based on the IAEA standard for design of nuclear power plants. This is absolutely acceptable, as Generation IV SFRs are planned in a large power range, up to 1500 MW_e capacity. However, SCW-SMR is a small modular reactor, which needs special standards because of the different systems and arrangement of SMRs. Such open questions are for example the requirements concerning multiple unit NPPs (see Criterion 33 or Requirement 5.63). Being aware of this difference, IAEA has published some safety standards concerning SMRs. These are:
 - Applicability of Design Safety Requirements to Small Modular Reactor Technologies Intended for Near Term Deployment (for LWRs and HTGRs) (IAEA-TECDOC-1936, 2020) [20];
 - Design Features to Achieve Defence in Depth in Small and Medium Sized Reactors (No. NP-T-2.2, 2009) [21];
 - Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident (IAEA-TECDOC-1785, 2016) [22].

For multiple units, [20] introduces the term “multi-module unit” emphasizing the differences between the small and large multi-unit NPPs. *“In some of the designs available, multiple reactor modules share some safety systems, safety features for design extension conditions, or supporting services. The potential for design approaches using multiple modules introduces new safety considerations in areas such as common-cause failures, internal hazards and human factors (e.g. shared control room design). For multi-module units, it is important to ensure that the safety of the NPP is not negatively impacted by the adoption of a modular design, and this provides the driving force for the formulation of changes to the safety requirements established in SSR-2/1 (Rev. 1).”* [2]

Recommendation: Besides SSR-2/1 [2], the available SMR regulation standards and guidelines should be used as a basis in order to take into account the specific safety features of SMRs.

⁹ GIF Task Force on Safety Design Criteria (SDC-TF). See more about the Task Force here: https://www.gen-4.org/gif/jcms/c_93020/safety-design-criteria

3.2.2.2 *Criteria and requirements of SDC that need deeper evaluation¹⁰:*

- **“Criterion 15: Design limits**
A set of design limits consistent with the key physical parameters for each item important to safety for the nuclear power plant shall be specified for all operational states and for accident conditions.
5.4. The design limits shall be specified and shall be consistent with relevant national and international standards and codes, as well as with relevant regulatory requirements”

Requirements concerning design limits are of main importance in case of SCWR reactors. Some non-conformance can exist between detailed – national – technical regulation and the possible SCWR operational parameters. The mention of the very different design limits of SCWRs should be considered here. (In Req. 5.4 rather than in the criterion itself.)

- *“5.15B. For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously.”*

Important requirement for SMRs where up to 5-10 reactors can operate at the same site. For some SMR types (e.g. NuScale¹¹) the reactors are in the same building sharing the decay heat removal water pool, which creates a much tighter connection between the units, including some interconnections of the safety systems. This special feature could be reflected in the requirement (see at general remarks).

- *“5.26. The design basis accidents are preferably analysed¹² in a conservative manner. This approach involves postulating certain failures in safety systems, specifying design criteria and using conservative assumptions, models and input parameters in the analysis. The design basis accidents could also be analysed in a best estimate manner, together with adequately analysed and evaluated uncertainties.”*

Modification¹³ not accepted. The detailed evaluation of different deterministic approaches is necessary. By IAEA the conservative manner means performing deterministic safety analyses with different levels of conservatism associated with the computer code used (conservative computer code type or conservative assumptions about system availability or initial and boundary condition not fully best estimate). The realistic approach (or realistic analysis) means use of best estimate code (without uncertainty quantification), best estimate assumptions about system availability and best estimate initial and boundary conditions. The application of BEPU (best estimate plus uncertainty) approach (also called Option 3) is based on the development of physical and mathematical models in the last decades (i.e. our better understanding and extended knowledge) of the considered phenomena. (For Generation III LWRs, this approach is used in the design process and for DEC calculations, but for the analysis of DBC accidents usually the conservative (Option 1) or the combined method (Option 2) - best estimate computer codes with conservative parameters and assumptions - is applied, see IAEA SSG 2 [23]). Also, it

¹⁰ Deeper evaluation means that modification of IAEA requirements may be needed. For example, the definitions may need to be modified or new ones may need to be defined (for example: multi-unit NPP).

¹¹ See more about the NuScale here: <https://www.nuscalepower.com/>

¹² Underlined text marks modifications of IAEA SSR-2/1 in GIF SDC. See also Annexes.

¹³ The underlined text presents added text, when comparing 5.26. to IAEA SSR-2/1 Rev.1 requirement 5.26.

should be noted, that in U.S. the large-break loss of coolant accidents have been typically calculated using BEPU (alternative is conservative Option 1).

According to IAEA SSG-2, Rev. 1 [23] for DBC one of the conservative methods (Option 1 – 3) should be used. In case of DEC-A Option 2 and 3 may be used. However, to be in line with the general rules for analysis of design extension conditions, best estimate analysis without a quantification of uncertainties may also be used. Analysis of severe accidents (DEC-B) should be performed using a realistic approach (Option 4 in Table 1, Section 2 of [13]) to the extent practicable. In case of SCWRs, the deep understanding of accident phenomena are still missing.

- *“Criterion 33: Safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant
Each of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions.
5.63. To further enhance safety, means allowing interconnections between units of a multiple unit nuclear power plant shall be considered in the design.”*

The above Criterion 33 differs from IAEA SSR-2/1 Rev. 1 Requirement, which starts: "Each *unit* of a multiple unit...". The criterion shall be re-evaluated considering the SMR safety approaches. The term 'multi-module units' is applied by IAEA documents for SMRs instead of multiple units. However, the specific features of such multi-module unit type SMRs means new type of interconnections between the reactors compared to the traditional LWRs (e.g. common containment building or decay heat removal water pools). See at general remarks.

- *“6.9. The means for shutting down the reactor shall consist of at least two diverse and independent systems. For design extension conditions, inherent power reduction with complementary shutdown method and/or passive shutdown capabilities shall be provided to prevent severe core degradation and to avoid re-criticality in the long run.”*

The modification requires re-evaluation. For DEC conditions, usually no additional shutdown systems are required in case of Generation III reactors. Apart from ATWS, diverse shutdowns systems are capable of stopping the chain reaction.

- Insertion of additional requirement should be considered for SCW-SMR:
“6.19C. The design of containment shall ensure the proper arrangement of structures, systems and components, making possible to perform their safety functions. (Especially for passive safety systems where the proper elevations are key factors for natural convection development.)”
- Insertion of a new requirement concerning the radioactive contamination of steam supply system could be considered for SCW-SMR.
“6.58A. The possible radioactive contamination of the turbine generators shall be taken into consideration during the design of the system and for the planning of radiation protection of plant personnel.”

3.2.3 Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV SFR document [14]

The document (“SDG document”, [14]) has been prepared as well by the GIF SDC-TF and was published in 2016, with an update (Rev.1) in August 2019.

Being published in 2013 (rev. 2017), the Safety Design Criteria (SDC) [13] established the high-level requirements for sodium cooled Generation IV reactors. The aim of the SDG document [14] is to give practical recommendations on how to meet the SDC and address SFR-specific safety issues. The SDG document has been reviewed by the NEA and the IAEA. The wording of the SDG is “should”, showing that the recommendations are not obligatory, but give a method to comply with the “shall” requirements.

The document does not cope with all safety concerns of the SFRs – for example, it does not cover the fuel handling and storage systems of a reactor.

3.2.3.1 Structure of the SDG document:

In the first part (Section 1.) a short introduction about the regulation framework and the scope of the Guideline is given.

Section 2. (“Main Characteristics of Gen-IV SFR Systems”) presents the SFR technology systems and fundamental characteristics of four current Gen-IV SFR designs.

Section 3. “General Approach” describes the different plant states taken into consideration in the design basis. It introduces the concept of “practical elimination” and “residual risk”. The definition of practical elimination is similar as in case of Generation 3 reactors (basically the events with unacceptable consequences should be practically eliminated), however, among the situations to be practically eliminated in Section 3.5.2 “situations leading to failure of the containment with a risk of fuel damage” include for example “Complete loss of the decay heat removal functions, leading to core damage and failure of the reactor coolant boundary” and “Core uncovering due to sodium inventory loss”. It is not really clear, why would these events lead to early / large release into the environment, as there is an IVR (in-vessel retention) approach for core damage.

The section describes the general approach to Normal Operation, AOO and DBA and the general approach to DEC as well (see Figure 3). It describes the possible inherent and passive safety systems for DEC states, the Decay Heat Removal systems and the concept of the IVR. For DEC states, the following events are described:

- Anticipated Transient Without Scram (ATWS);
- Loss of Safety Systems for Decay Heat Removal;
- Reactor Coolant Level Reduction.

This section describes the situations to be practically eliminated in details. However, as mentioned above, not the same situations should be subject of practical elimination in case of water-cooled reactors (such as SCWRs).

Section 4. (“Guidelines for Application of Safety Design Criteria”) contains the recommendations for meeting the requirements of Design Criteria. The scope of the Guidelines is the following:

- “reactivity characteristics of the reactor core;
- reactor shutdown system;

- containment function; and
- decay heat removal system, including important design parameters, postulated events and design limits, as well as testability and safety design demonstrations”.

For all the issues recommendations are grouped based on the safety function to be performed (such as prevention of core damage / mitigation of the core damage etc., see Figure 3).

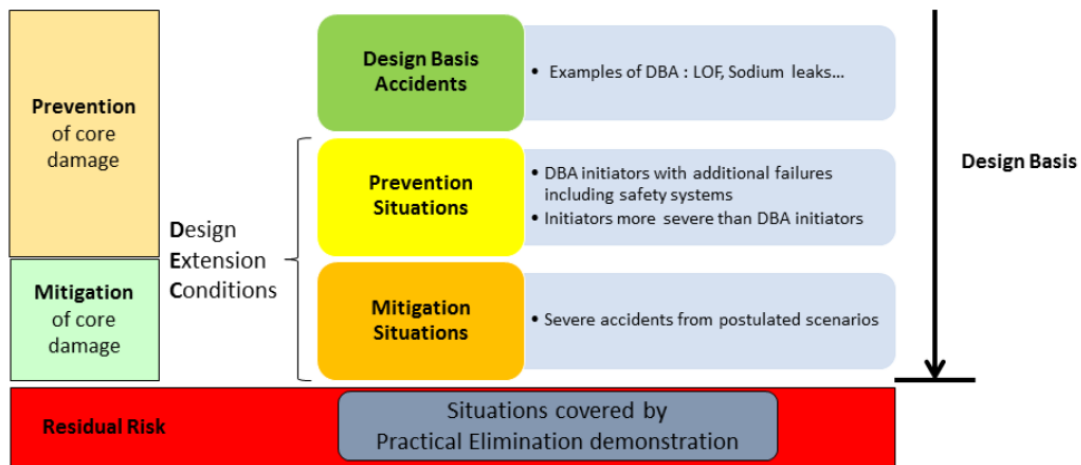


Figure 3. Design basis and residual risk [14]

For the given area the SDC identifies the relevant SDC requirements and the design concept. For the area specific functional requirements and design considerations are given.

Section 5. gives a detailed description of the reactivity characteristics (that is a very specific SFR issue because of the positive void factor of SFRs). It shows the considerations concerning reactivity accidents in DBA or DEC states.

Appendix A of the SDG document [14] gives very useful insight into the use of Objective Provision Tree (OPT) method through the example of determination of the situations to be practically eliminated performed for the Japanese SFR (JSFR). Appendix B of the SDG document [14] gives a preliminary list of some AOO and DBA initiating events.

3.2.3.2 Main findings of the review:

The SDG document [14] gives a good basis for the development of the set of regulation requirements. It can be used as a practical tool for the fulfilment of SDC requirements.

However, as a rather technical document, the SDG is not really suitable to be used as a direct basis for the development of the similar documentation for SCWRs (for example to the SCW-SMR) because the technical differences are extremely large between the different Generation IV reactor concepts. After the development of the Design Criteria document for SCWRs/SCW-SMRs, a similar guideline should be issued. See more comment in Annex 3.

3.2.4 An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems document [15]

The document (“ISAM document”, [15]) has been prepared by the Risk and Safety Working Group (RSWG) of the GIF¹⁴ and was published in 2010, with an update (version 1.1) in June 2011.

3.2.4.1 Integrated Safety Assessment Methodology (ISAM):

The ISAM methodology has been developed in the last decades for a systematic evaluation of safety issues of different nuclear systems. The method is a PSA-based procedure which ensures the thorough safety evaluation of nuclear reactors starting from qualitative evaluation up to a detailed probabilistic and / or deterministic analysis of the affected systems (the use of deterministic and probabilistic analysis conforms to the IAEA GSR Part 4 standard assessment process, see Figure 4 below). The aim of this systematic approach is to prevent the necessity of design backfits (i.e. the subsequent modification of the design of the reactor or the safety systems to meet the safety requirements). This is in line with IAEA GRS Part 4 standard, which in paragraph 1.8 defines stages in the lifetime of the facility including safety assessment and "Development of the design".

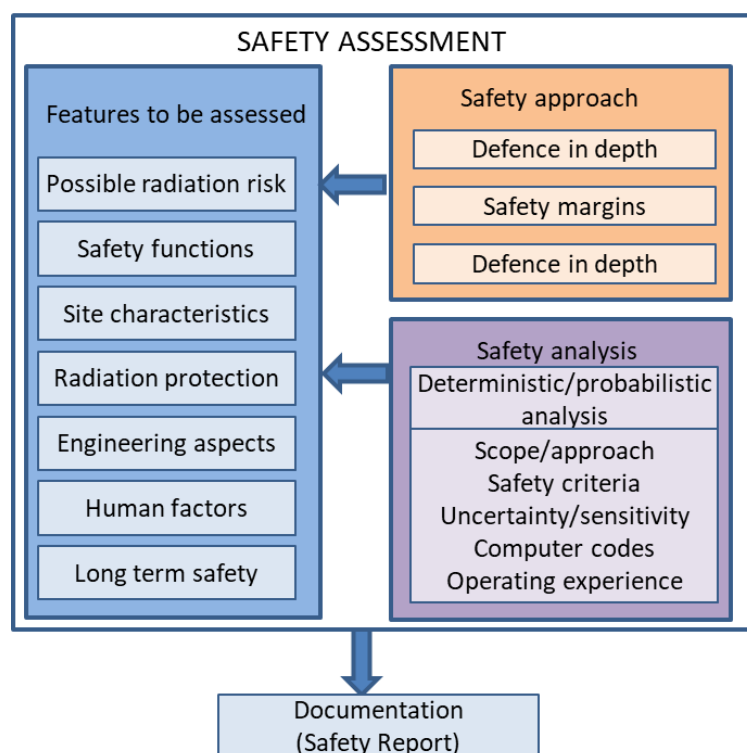


Figure 4. Overview of the safety assessment process (adapted per FIG. 1. of [59])

The **ISAM methodology can be applied at the very early phase of the design process, so that it can help to achieve that safety is “built-in” rather than “added on” by influencing the direction of the concept and design development from its earliest stages**, according to the ISAM document [15].

¹⁴ For more information about the link between SDC-TF and RSWG see the 5th paragraph here: https://www.gen-4.org/gif/jcms/c_92818/task-force-on-safety-design-criteria

Recommendation: Apply the ISAM methodology as early as possible in the design process of the SCW-SMR concept. Its application may be prepared during the lifetime of the ongoing ECC-SMART project then started in the next project by the application of first¹⁵, second¹⁶, third¹⁷, fourth and fifth¹⁸ steps of ISAM.

ISAM helps to identify the safety vulnerabilities of the concept, so these possible weak points can be improved or eliminated at a very early phase in the development process, decreasing development costs and time needed for the design process, and preventing the posterior modification of the design. According to the ISAM document [15] the methodology can be used for Generation IV reactors in three principal ways:

- *The ISAM is intended for use throughout the concept development and design phases with insights derived from the ISAM serving to influence the course of the design evolution. In this application of the methodology, the ISAM is used to develop a more detailed understanding of safety related design vulnerabilities, and resulting contributions to risk. Based on this detailed understanding of safety vulnerabilities, new safety provisions or design improvements can be identified, developed, and implemented relatively early.*
- *Selected elements of the methodology will be applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision makers.*
- *The ISAM can be applied in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria. In this way, the ISAM will allow evaluation of a particular Generation IV concept or design relative to various potentially applicable safety metrics or “figures of merit”. This post facto application of the ISAM will be especially useful for decision makers and regulators who require objective measures of safety for licensing purposes, or to support certain late-stage design selection decisions.*

ISAM, as a general methodology for safety development is a quite new, risk-informed method, developed by the GIF-RSWG. **It includes the qualitative evaluation of safety and technology gaps** identified and evaluated by expert judgement made by experts of the given technology. **The identification of these gaps can help to determine the fields of further development and the needs for further experimental activities.** In the quantitative steps of ISAM the deterministic and probabilistic analysis of the selected systems and / or processes is performed. The ISAM methodology consists of the following five consecutive steps:

- *Qualitative Safety Features Review (QSR);*
- *Phenomena Identification and Ranking Table (PIRT);*
- *Objective Provision Tree (OPT);*
- *Deterministic and Phenomenological Analyses (DPA);*
- *Probabilistic Safety Analysis (PSA).*

¹⁵ QSR can be iteratively applied from the pre-conceptual phase (prerequisite is a reasonable pre-conceptual design (PCD)).

¹⁶ Prerequisites for the first application / iteration of PIRT are the existence of a PCD, a set of relevant accident scenarios and a complete set of phenomena influencing each accident scenarios.

¹⁷ OPT can be iteratively applied from the pre-conceptual phase (prerequisite is a reasonable pre-conceptual design (PCD)).

¹⁸ DPA and PSA can be applied rather in more matured design phases (from late PCD), up to the licensing of the Generation IV system.

QSR, PIRT and OPT are basically qualitative methods, while DPA and PSA are well-known quantitative methods using different system codes and PSA analysis tools.

ISAM, as a whole concept of an integrated safety evaluation (see Figure 5) is not really widely used yet, however some examples can be found in the literature (e.g. by the JAEA for an SFR design, or for DEMO fusion project in Korea). However, the single parts of the methodology are quite well known, for example PIRT method is widely used for the safety evaluation of specific systems or phenomena in LWRs as well.

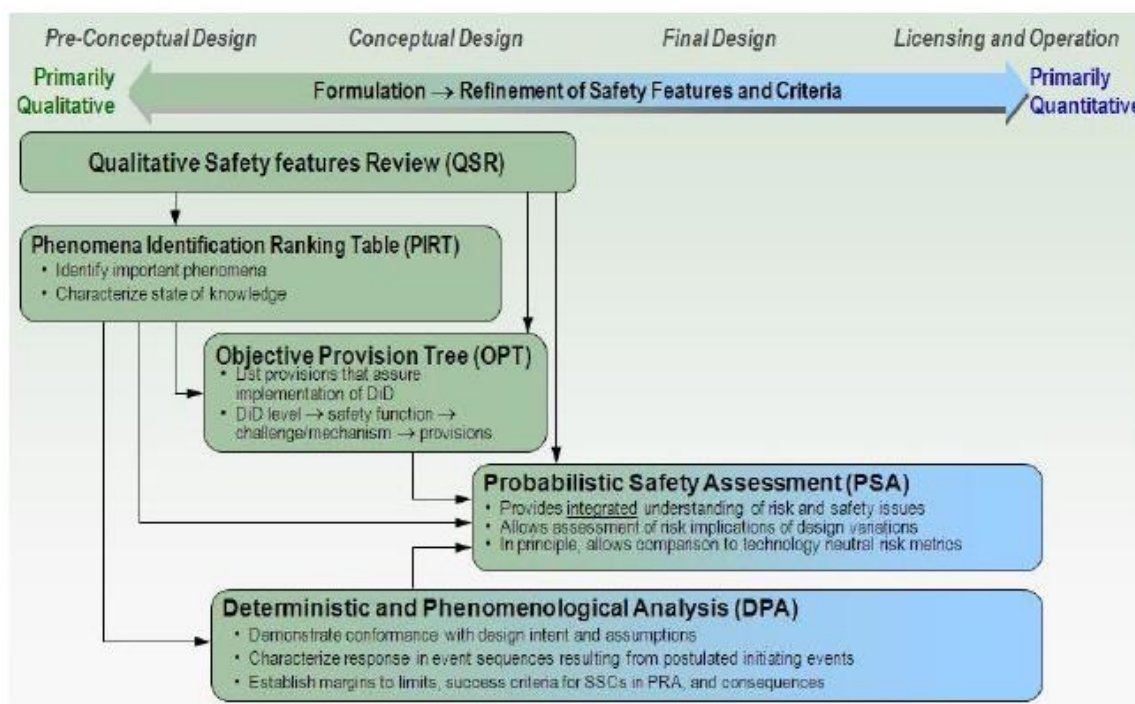


Figure 5. Proposed GIF Integrated Safety Assessment Methodology (ISAM) Task Flow [15]

3.2.4.2 Parts of Integrated Safety Assessment Methodology:

Different elements of ISAM methodology can be applied for different stages of design maturity (see Figure 5).

3.2.4.2.1 Qualitative Safety Features Review (QSR):

The Qualitative Safety Features Review (QSR) provides a systematic means to identify and discuss the safety-related attributes and characteristics of the given Generation IV design and to evaluate the relevant references for it (IAEA, GIF, INPRO guidelines, etc.).

QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety is "built-in, not added-on" through the early phases of the design of Generation IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of DiD, such as transparency, high reliability, minimization of sensitivity to human error, and other important safety characteristics, might best be incorporated.

It provides a measure of discipline to help ensure that certain desirable characteristics are incorporated into the design at its earliest phases. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the design under

development in terms of safety characteristics, i.e., assets or vulnerabilities that will be analysed in more depth in those other analytical steps.

The method groups the applicable requirements and recommendations for different DiD levels (see also Table 1) and different safety functions into classes as it follows:

➤ *Class 1 – Generic & Technology neutral (i.e., applicable to all the technologies implemented by the innovative systems)*

○ *Class 2 – Detailed & Technology neutral*

▪ *Class 3 – Detailed & Technology neutral but applicable to a given safety function*

• *Class 4 – Detailed, applicable to a given safety function, and technology specific, i.e., applicable to a given reactor technology.*

As a result, design objectives are determined based on RSWG and other references and compared with the given Generation IV design characteristics.

3.2.4.2.2 Phenomena Identification and Ranking Table (PIRT):

PIRT is a well-known qualitative process to identify the importance of safety issues and to determine the knowledge of them.

The method is based largely on expert elicitation. The **process involves selecting hardware** (i.e., nuclear power plant), **selecting an accident scenario**, and then **identifying all plausible phenomena impacting on the outcome of the accident. Each phenomenon is then ranked in order of relative importance and its state of knowledge.** The PIRT provides a structured means of identifying and analysing a wide variety of off-normal scenarios that potentially challenge the viability of complex technological systems. **The PIRT methodology brings into focus the phenomena that dominate, while identifying all plausible effects to demonstrate completeness.**

PIRT has been applied to different Generation III reactors and Generation IV concepts as well. An example is the application to the Japan Sodium Fast Reactor – JSFR, for the evaluation of the operation of the planned self-actuation shutdown system (SASS).

During the process important phenomena are identified and ranked based on their effect on a selected key parameter (called figure-of-merit, FOM), chosen by a panel of experts. The ranking process consists of the classification of importance and knowledge level of the phenomena (see Figure 6). The typically used categories for importance are High, Medium, Low, Insignificant. For the knowledge level evaluation, the level of uncertainty is also evaluated.

Rank	Definition	Application Outcomes
High (H)	<i>Phenomenon has controlling impact on figure-of-merit</i>	<i>Experimental simulation and analytical modelling with a high degree of accuracy is critical</i>
Medium (M)	<i>Phenomenon has moderate impact on figure-of-merit</i>	<i>Experimental simulation and/or analytical modelling with a moderate degree of accuracy is required</i>
Low (L)	<i>Phenomenon has low impact on figure-of-merit</i>	<i>Modelling must be present only to preserve functional dependencies.</i>
Insignificant (I)	<i>Phenomenon has no, or insignificant impact on figure-of-merit</i>	<i>Modelling must be present only if functional dependencies are required.</i>

Table 1: Most Often Used Phenomena Ranking Scales

Rank	Meaning
4	<i>Fully known, small uncertainty</i>
3	<i>Known, moderate uncertainty</i>
2	<i>Partially known, large uncertainty</i>
1	<i>Very limited knowledge, uncertainty cannot be characterized.</i>

Table 2: Most Often Used Knowledge Based Ranking Scales

Figure 6. Possible ranking of phenomena importance and knowledge level [15]

Based on the PIRT evaluation process, important knowledge gaps can be identified. The results can be used for the determination of further analytical and / or experimental investigations, determining a priority R&D effort (shown red in Figure 7).

Knowledge Base Gap Determination				
Adequacy of knowledge	Rank of Phenomenon			
	H	M	L	I
(4) Fully known; small uncertainty				
(3) Known; moderate uncertainty				
(2) Partially known; large uncertainty	GAP	GAP		
(1) Very limited knowledge; uncertainty cannot be characterized	GAP	GAP	GAP	

Figure 7. Results of PIRT process [15]

3.2.4.2.3 Objective Provision Tree (OPT):

The OPT is a relatively new analytical tool, developed mainly by the IAEA. It provides an exhaustive overview of the safety related architecture and allows the identification, for each level of the DiD, of all provisions that contribute to the achievement of safety functions as well as their mutual interrelations.

Based on the results of the PIRT evaluation, it identifies the design provisions necessary for preventing, controlling or mitigating the consequences if the given phenomena occurs. OPT can be applied from the pre-conceptual phase through the whole design process in an iterative manner.

OPT uses a tree structure for the visualization of safety structures (see Figure 8). This tree structure is prepared for all DiD level. For the given level the safety objectives and barriers to be protected are identified, and for them the connected safety functions are set. After determination of the possible challenges and their damaging mechanism for a safety function provisions are listed, which are able to cope with the given challenges.

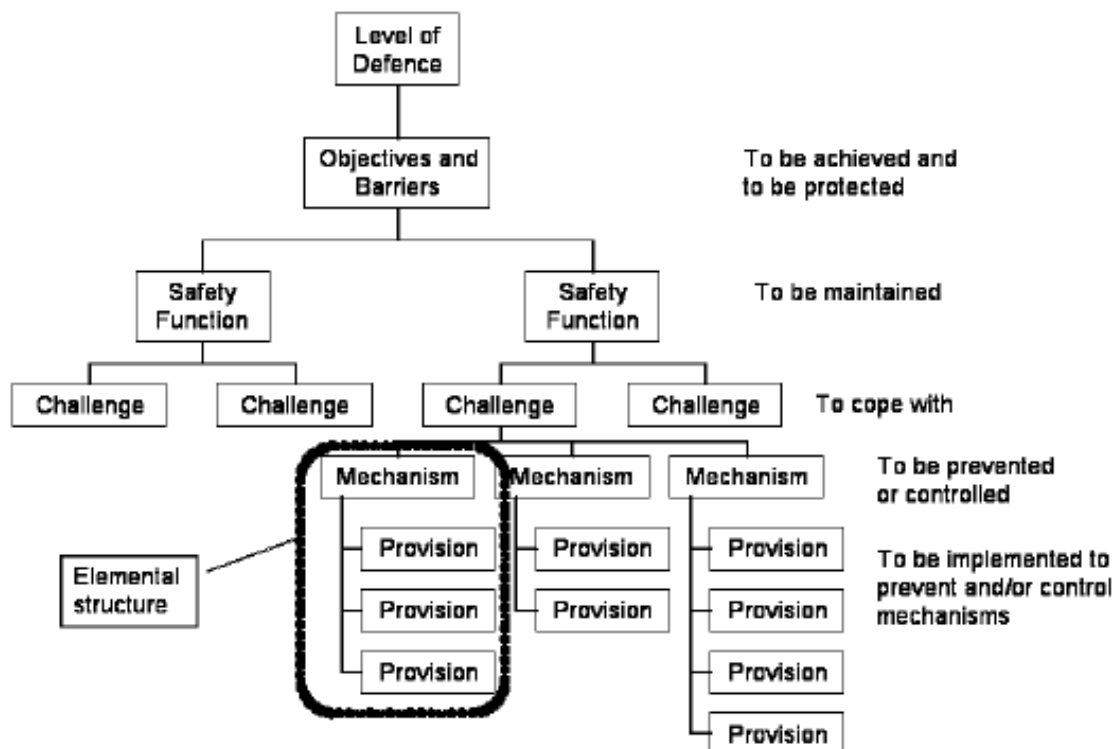


Figure 8. Hierarchy structure of OPT [15]

The result of the OPT gives a so-called line of protection (LOP), which is the elemental structure shown in Figure 8. This LOP includes the necessary provisions that together can cope with the given deterioration mechanism of the challenge.

3.2.4.2.4 Deterministic and Phenomenological Analyses (DPA):

DPA consists of the use of “conventional” deterministic and phenomenological analyses, as thermal-hydraulic, computational fluid dynamics (CFD), reactor physics analyses, structural models, etc. The **aims of DPA are:**

- To help to understand the phenomena important from safety viewpoint;
- To supply input data for PSA analyses;
- To determine the uncertainties of phenomenon description.

These quantitative analyses are expected to be performed by traditional deterministic system codes, CFD or other finite element codes, etc. **DPA can be applied rather in more matured design phases** (from late pre-conceptual design), **up to the licensing of the Generation IV system.**

3.2.4.2.5 Probabilistic Safety Analysis (PSA)

The basis of the ISAM methodology is the full-scope PSA. **PSA can be applied rather in more matured design phases** (from late pre-conceptual design), **up to the licensing of the Generation IV system**. However, development of earlier application is in progress with the help of the so-called living PSA applications (updated frequently for the evaluation of design changes).

PSA is widely used in different nuclear systems, and is obligatory for new and operating nuclear reactors in many countries. **For new designs it can help to understand and manage different types of uncertainties.**

3.2.4.3 Evaluation of ISAM methodology for concept development of ECC-SMART project:

Integrated Safety Assessment Methodology can be evaluated as an important tool for the development of SCW-SMR design in the ECC-SMART project. The advantages of the ISAM methodology are the followings:

- Applicable for whole design or only for specific systems or phenomena (see examples of JSFR SASS system);
- The method has been developed further since the release of ISAM document [15] in 2011, new guidance created by JRC is available, and some examples of the application are available;
- The method has been already applied for Generation IV reactors (e.g. SFRs). Canada even has an experience with ISAM concerning SCWRs;
- Parts of ISAM (as the PIRT method) are well established, with decades of experience with application for Generation III and IV reactors;
- Can help to identify safety gaps of different designs and to evaluate the needs for further evaluations and research activities (such as the necessity of experimental facilities etc.);
- Can take advantage of number of international experts participating in the ECC-SMART project.

3.2.4.3.1 Disadvantages of the ISAM methodology are:

- **For the application of ISAM methodology a preliminary design is needed** in order to determine the phenomena and systems being subject of evaluation, thus some design decisions have to be made concerning the size and technology of the reactor (see Figure 9). (However, it is also possible to use the ISAM for the selection of the favourable design from competing ones, but that needs more resources.);
- At the present pre-pre-conceptual design phase only the first and second (partly the third) steps can be used;
- The qualitative analysis is based on the judgement of an expert panel, so the adequacy of the result relies on the members of this expert panel – appropriate number and qualification of experts is necessary;
- The quantitative analysis (OPT, deterministic analysis of selected phenomena and PSA analysis) needs experts with large experience, mainly in field of PSA;
- The ISAM, as a whole approach has not an exhaustive literature. However, some parts of it (mainly the PIRT) have been used for a wide range of nuclear technologies.

Development stage	QSR	PIRT	OPT	DPA	PSA
Selection of a reactor type	X	X	X		
Definition of high safety issues	X	X	X		
Definition of safety provisions		X	X	X	
Definition of safety systems initial design			X	X	X
Definition of safety systems final design			X	X	X

Figure 9. Applicability of different ISAM steps during the design evolution [15]

3.2.4.3.2 Examples of ISAM and / or PIRT application:

- Guidance Document for Integrated Safety Assessment Methodology (ISAM) – (GDI), EC JRC report prepared for GIF RSWG, 2014, https://www.gen-4.org/gif/jcms/c_65330/guidance-document-for-integrated-safety-assessment-methodology-isam-gdi
- Phenomena Identification and Ranking Table, R&D Priorities for Loss-of-Cooling and Loss-of-Coolant Accidents in Spent Nuclear Fuel Pools, OECD NEA No. 7443, 2018, <https://www.oecd-nea.org/upload/docs/application/pdf/2020-01/dir1/csni-r2017-18.pdf>
- Kyemin Oh, Myoung-suk Kang, Gyunyoung Heo, Hyoung-chan Kim: *Safety studies on Korean fusion DEMO plant using Integrated Safety Assessment Methodology*, Fusion Engineering and Design, Volumes 98–99, October 2015, Pages 2152-2156
- David J. Diamond: *Experience Using Phenomena Identification and Ranking Technique (PIRT) for Nuclear Analysis*, PHYSOR-2006 Topical Meeting, Vancouver, British Columbia, Canada, September 10-14, 2006; <https://www.bnl.gov/isd/documents/32315.pdf>

3.2.5 Super-Critical Water-cooled Reactors (SCWRs) Risk and Safety Assessment White Paper document [16]

This document has been prepared by three authors delegated by the GIF Risk and Safety Working Group (RSWG), the GIF Super-Critical Water-cooled Reactor (SCWR) System Steering Committee (SSC), and the GIF as a whole institution [16]. This White Paper provides an overview of activities conducted by participating members of the GIF in the application of the ISAM¹⁹ developed by the GIF RSWG. Based on the proper application of the ISAM, the GIF SCWR SSC has elaborated the future Research and Development (R&D) needs for the conceptual design of SCWR systems. Several areas have been identified which do require additional R&D and analysis in order to improve the risk and safety performance of the different SCWR concepts. The White Paper provided a summary of the identified areas for improvement [16].

The objective of this White Paper was to perform an assessment of the adequacy of safety provisions incorporated in the conceptual design completed so far for the SCWR systems. While the expected outcomes of the White Paper were improvements required to demonstrate DiD and additional R&D as required. It presented an overview of the SCWR concept and described the application of some of the ISAM tools and their outcomes.

¹⁹ The RSWG documented a set of five distinct analytical tools as an ISAM to assess Gen-IV systems under development to ensure that safety is “built-in” rather than “added-on”.

The White Paper focused on the safety of the Canadian SCWR concept and only a supplementary description has been provided on the High-Performance Light Water Reactor (HPLWR or European SCWR). The reason was that the HPLWR project ended in 2010 before the first version of the ISAM became available. Thus, **the ISAM methodology is expected to be applied in the next available European project on risk and safety of SCWRs** [16]. Consequently, the **application of the ISAM methodology or at least a suitably selected portion of the five analytical tools should be performed in the framework of the next EU financed research project, if the pre-conceptual design of the concept will be available.**

Recommendation: *It is suggested that suitable preparation should be carried out during the ongoing ECC-SMART project to enable an immediate start of the application of the ISAM methodology in case of a possible next EU project on the SCW-SMR concept.*

Figure 10 shows a comparison on the general features of the simplified control systems of a BWR and a SCWR [16]. As it can be seen on the left part of Figure 10, the so-called once-through system of a BWR has an important safety advantage in terms of general safety system requirements: it has a closed coolant loop inside the reactor (partially depicted and partially not in Figure 10) during emergency situations. Thus, in case of a BWR the residual heat can be removed by natural convection (driven by the rising steam in the core and above) and the safety system has to ensure sufficient amount of coolant inventory into the pressure vessel to keep the core covered with water. On the other hand, in case of a SCWR (right part of Figure 10) the residual heat can be removed only by forced convection inside the reactor pressure vessel (RPV), which forced convection may be driven by a natural convection loop outside the RPV. Thus, the requirement for the safety system in case of a SCWR is to ensure sufficient coolant mass flow rate to remove the decay heat from the reactor core [16]. Despite this fundamental difference between the general features of the simplified control systems of a BWR and a SCWR, a **BWR has numerous safety system requirements, which can be directly applied to a SCWR concept without significant modifications.**

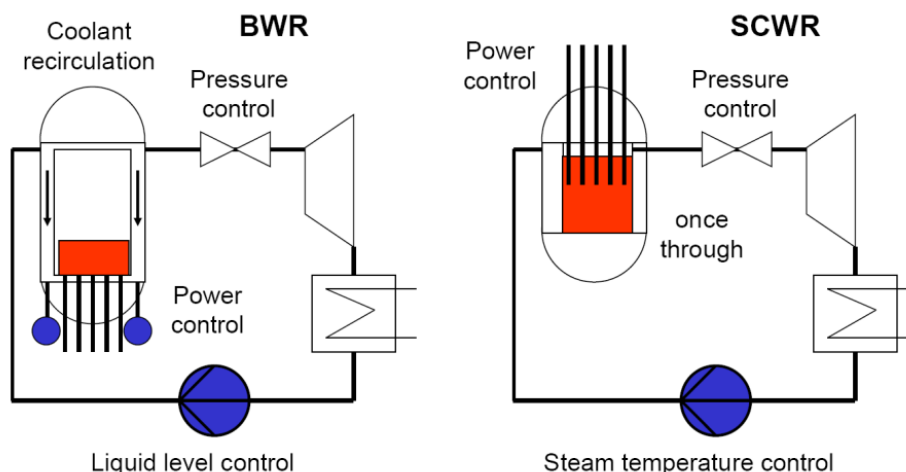


Figure 10. Comparison on general features of simplified control systems of a BWR and a SCWR [16]

BWR safety system requirements, which can be directly applied to a SCWR concept without significant modifications are [16]:

- reactor shutdown system consisting of either control rods or a boron injection system as a second and diverse shutdown system,

- containment isolation by active and passive containment isolation valves in each line penetrating the containment to close the third barrier in case of an accident,
- steam pressure limitation by pressure relief valves,
- automatic depressurization of the steam lines into a pool inside the containment through spargers to close the coolant loop inside the containment in case of a need for containment isolation,
- coolant injection system to refill coolant into the pressure vessel after intended or accidental coolant release into the containment,
- pressure suppression pool to limit the pressure inside the containment in case of steam release inside the containment, and
- residual heat removal system for long term cooling of the containment.

3.2.5.1 Overview of the Safety Assessments of Canadian SCWR:

A preliminary analysis has been performed including three ISAM tools (QSR, PIRT, and PSA) [15] during the concept development phase of the Canadian SCWR.

In the first step, Qualitative Safety Features Review (QSR) was performed. The QSR is an important tool due to it is the only tool of the ISAM which has been specifically developed for Generation IV reactor systems [16]. Based on the QSR analysis, many additional required information and understanding of a number of relevant but currently unknown phenomena (e.g., heat transfer under supercritical condition and high temperature materials) were identified due to the Canadian SCWR concept, which is essentially an innovative nuclear system with new phenomena that are not present in the current fleet of reactors [16].

In the second step PIRT was applied in order to identify, recognize, and qualify the relative importance of selected relevant phenomena (associated with the Canadian SCWR concept) with the associated rationales and their level of knowledge. The PIRT process consisted of the nine standard steps and allowed the evaluation of the reactor concept and design by following how each phenomenon influences the key measurable parameter(s), called the Figure-of-Merit (FOM), chosen consensually by a panel of experts [16].

PSA was not required at the conceptual development phase. Nevertheless, a simplified PSA has been performed in the third step to quantify the core damage frequency (CDF) for the safety systems of the Canadian SCWR concept [16].

As a summary of the preliminary application of the ISAM it can be pointed out that the method described in ISAM report has been strictly followed during the QSR, the PIRT, and the PSA (see above and in [16]). Thanks to these above mentioned three tools key areas were identified where additional knowledge is needed in order to maintain the adequacy of safety-related system design.

As a summary it can be stated that the Canadian SCWR concept offers the following advantages in mechanical design and during operations compared to present (Generation III) pressure-tube type CANDU reactors [16]:

- eliminates inlet feeders,
- eliminates channel closure plugs (two for each fuel channel),
- eliminates channel closure seal (two for each fuel channel),
- allows batch fuelling with simultaneous multi-channel fuelling and convenient access through a hollow inlet plenum,

- easier pressure tube replacement with convenient access to fuel channels by removing the inlet plenum head,
- enables a compelling safety case,
- passive core cooling is made possible through natural convection of heavy water moderator,
- small LOCA of an outlet has a small impact due to the common coolant inlet,
- provides 40% higher efficiency, and
- presents a compact footprint and layout.

3.2.5.2 Overview of the Safety Assessments of High-Performance Light Water Reactor:

The so-called High-Performance Light Water Reactor (HPLWR) is essentially a pressure vessel type SCWR design developed by the European SCWR community since the beginning of 2000'.

Figure 11 shows the minimum necessary set of safety systems for the HPLWR concept which fulfil the ISAM requirements. Control rods and shut down rods serve as the reactivity control system. They can be inserted into the reactor from the top like a pressurized water reactor (PWR) unlike for a BWR the HPLWR does not has steam separators and steam dryers [16].

The boron injection system in the form of a vessel containing boron acid must be placed inside or outside the containment (connected to the primary system) in order to enable an additional shut down option under accidental conditions. The boron injection system cannot be used to compensate excess reactivity during normal operation (different from the reactivity control system of a PWR) due to fuel rods doped with Gadolinium as a burnable poison planned to be used instead for this purpose [16].

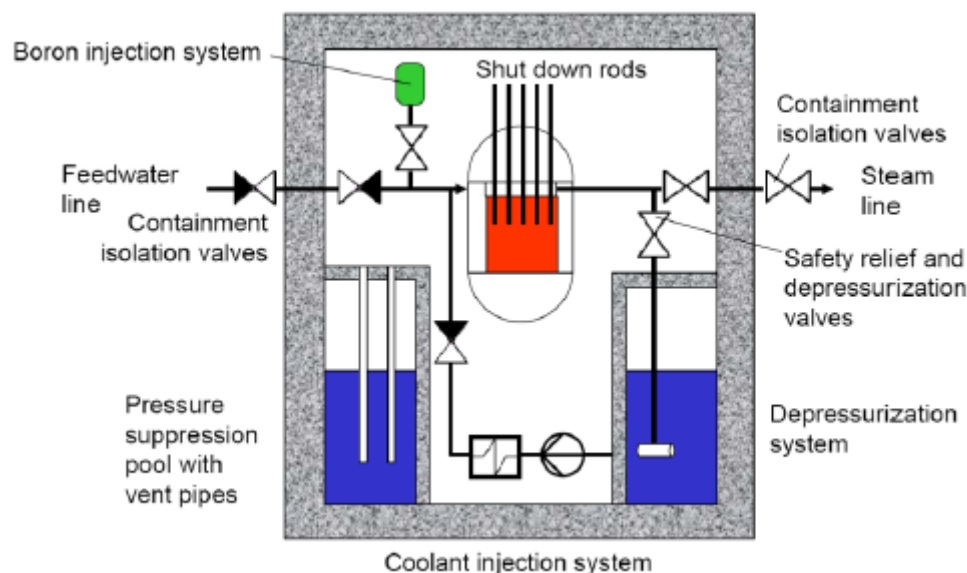


Figure 11. The minimum necessary set of safety systems for the HPLWR concept [16]

The containment isolation valves are planned to be check valves in the feedwater lines (these valves need to be damped in order to avoid the water hammer phenomena) and steam isolation valves with hydraulic and medium controlled actuators in other lines [16].

There is an essential need for a pressure suppression pool with vent pipes in the HPLWR to maintain the pressure of the containment below the design limit. This system can also play an important role as a heat sink for the automatic depressurization system [16].

Figure 11 shows a low-pressure coolant injection system with a heat exchanger to a secondary emergency coolant system under the reactor itself. It means that this system must be installed somewhere inside or outside of the containment but for sure it must be placed in a sheltered location [16].

In case of a simultaneous trip of all feedwater pumps caused by a station blackout, as a consequence, the primary system of the HPLWR must be depressurized immediately. It is the only option to maintain a coolant mass flow rate through the core in order to keep it coolable. In this situation it is not advised to close the turbine governor valve in order to keep a high system pressure (like in case of a BWR) because such an action would stop the steam flow simultaneously (which would overheat the reactor core itself) [16].

Based on the simulation results of the pressure and coolant temperature history in case of containment isolation of all feedwater and steam lines, the following conclusion can be drawn. A low-pressure coolant injection system with a certain pressure head (which is at least 6 MPa), enabling ramp up of coolant injection pump and emergency power supply within 20s, allows for long-term cooling of the reactor core. Having a closed cooling circuit within the containment prevents a significant and fast loss of coolant. Additionally, the residual heat would be removed to the secondary coolant system. While this time has been proven as feasible in conventional BWRs, it may appear to be short for SCC-SMR and it would be advantageous to have a longer grace period [16].

The safety assessments of Canadian SCWR and the European HPLWR have been concluded with the following lessons learnt. **There are no major impediments to further develop SCWR design concepts. But there are significant gaps in understanding how the pre-designed reactor cores would behave under SCWR pressures and temperature conditions** [16], [17].

3.2.5.3 *In conclusion:*

- The application of the ISAM methodology or at least selected portions of the five analytical tools should be performed in the framework of the next EU financed research project, providing the pre-conceptual design of the concept will become available. Preparatory steps to perform this task should be carried out already during the on-going ECC-SMART project.
- Despite significant reduction in the probability of severe accidents of SCWR conceptual designs, mitigation measures should be incorporated into the designs in order to avoid short- or long-term post-accident escalation.
- Ongoing R&D on the various technical aspects of the SCWR (e.g. materials, thermal hydraulics, etc.) should continue to close knowledge gaps.

3.2.6 Supercritical-water-cooled reactor (SCWR) System Safety Assessment document [17]

This document (SSA document in the following) summarizes the lessons concluded from system safety assessments of SCWR relevant reactor developments in the past and the current ongoing

SCWR R&D activities (up to 2018). The Canadian SCWR concept represents the pressure tube design while the European HPLWR represents a pressure vessel type SCWR concept among the six existing SCWR concepts.

As explained in the SSA document, from a safety point of view, the SCWR concepts have adopted passive safety systems to complement active safety systems to enhance the safety performance of the SCWR concepts in comparison to the current fleet of existing LWRs. Basically, the design of the safety systems adheres to the DiD safety principle [17].

3.2.6.1 *Main lessons learnt:*

Multiple fields where progress is needed have been identified in SSA document:

- Novel manufacturing processes: An example is the new fuel assembly concept like the Canadian fuel assembly containing 64 fuel rods. It may introduce new phenomena imminent to the new design itself (e.g. re-entrant flow).
- SCW is a harsher environment for any alloy compared to typical BWR or PWR water environments. Thus, extensive corrosion and environmentally-assisted cracking testing and demonstration on the suitability of alloys in SCW environments is required.
- Similarly, due to the peculiar characteristics of SCW with regards to thermal-hydraulics and neutronics and the strong coupling between the two, corresponding experimental testing and demonstration is required.
- Further validation of methods for SCWR concepts and of existing data is required, as significant simplifying assumptions and extensions of existing data and methods have been applied in the development of the various SCWR concepts.
- A significant knowledge gap of the SCWR is the change in material properties of fuel cladding materials as a function of irradiation damage. Experiments exposing fuel cladding materials to irradiation and corrosion at the same time have been rarely performed yet.
- It was explained that additional R&D data would be required in the field of “in-reactor” data despite there is reasonable confidence that in-reactor performance should be satisfactory during the relevant operational modes of the reactor.
- Additional material issues beside the ones above, e.g. the maximum so-called “diametral strain” estimated for the pressure tubes of the Canadian SCWR concept after 75 years of full power operation would require validation.
- Tests to provide data on thermal conductivity, fuel qualification, and performance of thorium-plutonium dioxide (Th, PuO₂) fuel to enable the implementation of this kind of fuel in different SCWR concepts.

3.2.7 Overall recommendations from reviewed IAEA, OECD-NEA and GIF documents

Based on the review of the above-mentioned documents, the most **important recommendations** are the following:

- The IAEA SSR-2/1 [2] should be used during the SCW-SMR development as a starting point for the design criteria development.
- [2]The available SMR regulation standards and guidelines should be used as a basis in order to consider the specific safety features of SMRs.
- After the development of the Design Criteria document of the SCWRs/SCW-SMRs, a safety design guidelines (SDG) on safety approach and design conditions for Generation IV SCWR/SCW-SMR should be issued.

- An exercise to apply the ISAM should be performed on the SCW-SMR concept as soon as a sufficiently detailed SCW-SMR concept becomes available.

In the meantime R&D on the SCWR and its different concepts to fill the knowledge gaps listed in Section 3.2.6. should be continued.

3.3 European Nuclear Safety Directive

Council Directive 2014/87/Euratom [24] amending Council Directive 2009/71/Euratom [25] *establishing a Community framework for the nuclear safety of nuclear installations*, applies to all nuclear stations. Therefore, it is classified as 1P (see Section 3 at page 19).

It is per nature generic and thus does not provide wide and concrete safety requirements. The few requirements of the directive are all well covered by existing IAEA safety guidance. For example, article 8a of the directive requires avoiding large and / or early releases, and this condition is already included in IAEA SSR-2/1 [2], under the so-called “practically eliminated conditions”. The main difference is that the EU directive is legally binding and thus mandatory for all EU member states, which is not the case for the IAEA set of safety principles, requirements and recommendations.

Council Directive 2013/59/Euratom [26] of 5 December 2013 only contains important information related to Annex I concerning reference levels for public exposure, which are important in order to establish arrangements for emergency preparedness. Since all the content applies to all nuclear stations, it is classified as 1P.

3.4 Other European Guidance

WENRA objectives and key positions gathered in [3] apply to all existing and new plants, and hence are classified as 1P.

This collection of safety elements fully aligns with EU dispositions and IAEA’s, in particular looking at eliminating the set of events leading to large and / or early release scenarios and limiting offsite countermeasures – even though qualitatively, cf. Table 3 in [3].

European Utility Requirements (EUR) collection of safety objectives and requirements [4] also align with international organization dispositions, specifying in quantitative fashion the limiting distances, where offsite countermeasures must apply, ranging from between 0.8 and 5 km, cf. [4]. FORATOM-ENISS, which is in charge of EUR, has recently issued a brief report on key positions specifically applicable to LW-SMRs. Seven key positions point at safety aspects. The elements contained here are therefore classified as 2P, and they sometimes match with safety requirements for new reactors, sometimes are a bit more demanding, e.g. smaller emergency planning zones (EPZs) for LW-SMRs or radiological safety objectives for severe accidents.

The Sustainable Nuclear Energy Technology Platform (SNETP) has also issued a position on SMRs in Europe [5], stating that *safety levels are equal or higher to 3rd Gen. PWRs*. It also states that *no radioactive release outside the limits of the plant, in order to (i) avoid the consequences of an accident requesting long term evacuation, and (ii) to avoid any consequences of an accident on a neighbouring country should occur*.

No radioactive releases outside the site boundary is in line with the second GIF safety goal. This statement however seems not to be consistent with (i) and (ii). Furthermore, the same report recommends applying *reduced emergency planning zones*. Therefore, it is unclear whether

SNETP recommends SMRs to avoid any offsite release like GIF, or only large or early releases, like EU nuclear safety directive, IAEA safety standards and WENRA objectives. SNETP dispositions are classified as 1P, within which 4th Gen. reactors are also implicitly included, hence classified as 3P as well.

4 National legal frameworks for nuclear safety and security

So far, no country that is seriously looking into SMRs has changed its national nuclear regulation to accommodate SMRs. But the regulators in some countries are in the process of changing their regulation accordingly or have announced that they will do so, e.g. STUK in Finland. However, nuclear regulators in some countries may have given opinions on SMR deployment in their countries, what they expect in terms of safety. In Canada and the U.S. some SMR designs (even non-water-cooled) are in the licensing or generic design assessment process. In the U.S. the licensing process for one LW-SMR, the PW-SMR of NuScale, is even completed. Although none of these SMR designs is a SCW-SMR design, some conclusions could be drawn from these processes for the SCW-SMR.

4.1 Canada

The information presented herein are excerpts taken from the Canadian Nuclear Safety Commission (CNSC) website (<https://nuclearsafety.gc.ca/eng/> [50]). As the licensing process is dynamic, it is recommended that the contents of this document be verified with information from the official CNSC website. Should discrepancies arise between this document and the official website of the CNSC, the information published in the CNSC would take precedence.

In Canada the legal regulator for the Canadian nuclear industry is the CNSC. It regulates the use of nuclear energy and materials to protect health, safety, security and the environment. It implements Canada's international commitments on the peaceful use of nuclear energy, and disseminate objective scientific, technical and regulatory information to the public.

Under the Nuclear Safety and Control Act (NSCA) [51], the CNSC has implemented regulations and by-laws with the approval of the Governor in Council. The foundation for the CNSC's regulatory framework is the Nuclear Safety and Control Act. The NSCA empowers the CNSC to make regulations and to develop other regulatory tools to establish requirements for and provide guidance related to the use of nuclear energy and materials in Canada. The NSCA came into force on May 31, 2000 when it replaced the Atomic Energy Control Act. It established the CNSC and set out the CNSC's mandate, responsibilities and powers. This Act provided the CNSC with the authority to regulate the development, production and use of nuclear energy and the production, possession and use of nuclear substances, prescribed equipment and prescribed information in Canada.

The CNSC's regulatory framework consists of laws passed by Parliament that govern the regulation of Canada's nuclear industry, and regulations, licences and documents that the CNSC uses to regulate the industry. These documents fall into two broad categories: those that set out requirements, and those that provide guidance on requirements. The CNSC is committed to providing regulatory tools that achieve clarity of its requirements.

The CNSC's regulatory framework elements fall into two categories: requirements and guidance.

Requirements are mandatory. Licensees or applicants must meet these requirements to obtain or retain a licence or certificate to use nuclear materials or operate a nuclear facility. Regulatory instruments under requirements include:

- NSCA – The Act to establish the CNSC, with the authority to regulate the nuclear industry.
- Regulations – Sets out statutory requirements.
- Regulatory documents – Provide greater detail than regulations, as to what the licensees and applicants must achieve in order to meet the CNSC's regulatory requirements.

Guidance provides direction to licensees and applicants on meeting requirements. Regulatory tools under guidance include:

- Guidance documents – Provide practical guidance to licensees and applicants on how to meet the regulatory requirements of the CNSC.
- Staff review procedures – Internal working documents used by CNSC staff to conduct regulatory reviews. Staff review procedures ensure consistent regulatory reviews.
- INFO-documents – Plain-language publications describing nuclear-related issues and regulatory requirements and processes, for the general public and other stakeholders. INFO documents also provide support and further information on other elements of the regulatory framework.

The CNSC's regulatory framework ensures clear direction to industry, stakeholders and CNSC staff. The CNSC's primary focus is to create a framework that supports the timely development and publication of regulations, regulatory documents and associated guidance.

With regard to the design of reactor design and licensing, the *Design of Reactor Facilities: Nuclear Power Plants (REGDOC-2.5.2)*²⁰ [52] document, sets out requirements and guidance for new licence applications for water-cooled nuclear power plants (NPPs or plants). It establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted international codes and practices. REGDOC-2.5.2 [52] provides criteria pertaining to the safe design of new water-cooled NPPs. All aspects of the design are taken into account, and multiple levels of defence are promoted in design considerations. To the extent practicable, the requirements and guidance provided herein are technology-neutral with respect to water-cooled reactors. An applicant or licensee may put forward a case to demonstrate that the intent of a requirement is addressed by other means and demonstrated with supportable evidence.

To a large degree, the REGDOC 2.5.2 [52] regulatory document represents the CNSC's adoption of the principles set forth by the International Atomic Energy Agency in SSR-2/1, *Safety of Nuclear Power Plants: Design* [2] as adapted to align with Canadian requirements.

This regulatory document considers all licensing phases, as information from the design process feeds into the processes for reviewing an application for a licence to construct an NPP, and other licence applications.

The REGDOC 2.5.2 [52] document is intended to form part of the licensing basis for a regulated facility or activity within the stated scope of the document. It is intended for inclusion in licences as either part of the conditions and safety and control measures in a licence, or as part of the safety and control measures to be described in a licence application and the documents needed to support that application.

For proposed new facilities: The REGDOC 2.5.2 [52] document will be used to assess new licence applications for reactor facilities. For existing facilities²¹: The requirements contained in the

²⁰ *Design of Reactor Facilities: Nuclear Power Plants* supersedes RD-337, which was published in 2008. In addition, it implements recommendations from the CNSC Fukushima Task Force Report. Currently the CNSC website states that REGDOC-2.5.2 Version 2 is under development (as of December 2021) and will supersede REGDOC-2.5.2

²¹ Existing facilities in this document are effectively those first licensed before 2014.

document do not apply unless they have been included, in whole or in part, in the licensing basis. The REGDOC 2.5.2 [52] regulatory document is part of the CNSC's Physical Design series of regulatory documents, which also covers: design of uranium mines and mills; design of fixed radiography installations; design of nuclear substance laboratories and nuclear medicine rooms; and exposure devices. The full list of regulatory document series can be found on the CNSC's website [50].

The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and establishes the basis for the CNSC's compliance program for that regulated facility or activity.

This regulatory document deals with a wide variety of topics related to the design of new NPPs. To the extent practicable, REGDOC 2.5.2 [52] is **technology-neutral** with respect to water-cooled reactors, and includes requirements and guidance for:

- establishing the safety goals and objectives for the design,
- utilizing safety principles in the design,
- applying safety management principles,
- designing structures, systems and components (SSCs),
- interfacing engineering aspects, plant features and facility layout,
- integrating safety assessments into the design process.

It is recognized that specific technologies may use alternative approaches. ***If a design other than a water-cooled reactor is to be considered for licensing in Canada, the design is subject to the safety objectives, high-level safety concepts and safety management requirements associated with this regulatory document. However, the CNSC's review of such a design will be undertaken on a case-by-case basis.***

In addition to the REGDOC 2.5.2 [52] regulatory document, the REGDOC-1.1.5 entitled ***“Supplemental Information for Small Modular Reactor Proponents”*** [53] provides information about CNSC safety and control areas as they apply to a licence application for an SMR facility. REGDOC-1.1.5 [53] provides information that is additional to the licensing process documented in REGDOC-3.5.1 [58], Licensing Process for Class I Nuclear Facilities and Uranium Mines and Mills, version 2, which provides an overview of the licensing process for Class I nuclear facilities.

REGDOC-1.1.5 [53] is a new regulatory document which set out requirements and guidance for an applicant to consider prior to submitting a licence application to the CNSC for a SMR. REGDOC-1.1.5 [53] also identifies the CNSC's considerations in assessing the adequacy of a licence application.

In accordance with CNSC regulations, the CNSC has developed licence application guides that set out requirements and guidance for submitting a formal application to the CNSC to obtain a licence for reactor facilities, including SMRs, in Canada. The licence application guides also help identify information that should be included in an application. REGDOC-1.1.5 [53] is intended to be used in conjunction with the following three licence application guides:

- REGDOC-1.1.1, Site Preparation and Site Evaluation for New Reactor Facilities [54],
- REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Nuclear Power Plant [55],

- REGDOC-1.1.3, Licence Application Guide: Licence to Operate a Nuclear Power Plant [56].

For the purposes of the REGDOC-1.1.5 document [53], the term SMR includes:

- water-cooled reactors (smaller than traditional reactors),
- advanced reactors with alternative coolant technologies (i.e. non-water-cooled).

SMRs may produce energy in the range of a few megawatts to a few hundred megawatts, and may be used for purposes beyond the generation of electricity. In some cases, an SMR facility could have multiple reactor units with a combined power output equivalent to that of a traditional nuclear power plant.

Protecting the environment is part of the CNSC's mandate. The CNSC requires the environmental effects of all licensed activities to be evaluated and considered when licensing decisions are made. This is a review of information used to support the Commission's determination on whether the licensee will make adequate provision for the protection of the environment and the health and safety of persons while carrying out a licensed activity. In addition to the information on environmental protection in the three licence application guides listed above, further information on this subject can also be found in REGDOC-2.9.1, *Environmental Protection: Environmental Principles, Assessments and Protection Measures*, version 1.1 [57], and REGDOC 3.5.1, *Licensing Process for Class I Nuclear Facilities and Uranium Mines and Mills*, version 2 [58].

The REGDOC-1.1.5 [53] also provides specific guidance for licence applications for SMRs, contains guidance on the application of the graded approach and the use of alternative approaches in the development of the licensing basis for SMR facilities, and outlines the role of pre-licensing engagement²².

REGDOC-1.1.1, Site Evaluation and Site Preparation for New Reactor Facilities [54]

This regulatory document sets out requirements and guidance for site evaluation and site preparation for new reactor facilities. It also includes a licence application guide for a licence to prepare a site for new reactor facilities. This document refers to both NPPs and small reactor facilities as "*reactor facilities*." Its content also addresses how site evaluation information obtained during site preparation activities is used and revisited in subsequent lifecycle phases of construction and operation.

REGDOC-1.1.2, Licence Application Guide: Licence to Construct a Nuclear Power Plant [55]

This regulatory document sets out requirements and guidance on submitting a formal application to the CNSC to obtain licence to construct a NPP.

REGDOC-1.1.3: Licence Application Guide: Licence to Operate a Nuclear Power Plant [56]

This regulatory document sets out requirements and guidance on submitting a formal application to the CNSC to obtain a licence to operate a NPP in Canada, and identifies the information that should be included in the application. This document will be used to assess licence applications for proposed new NPPs and for licence renewals for existing NPPs. Once the CNSC has granted a licence, the safety and control measures described in the licence application and the documents needed to support the application will form part of the licensing basis.

²² Pre-licensing engagement is optional

4.2 China

The information presented herein are taken from the document “Safety Review Principle for Small Pressurized Water Reactor Nuclear Power Plants (Trial)” drafted by China’s National Nuclear Safety Administration (NNSA). China’s NNSA draws up this review principle for onshore small PWR NPPs at the sites that may be adjacent to the users of power generation, heat supply, steam supply, etc. to clarify its position on a number of important safety issues. The formulation of the review principle as well as future practices in the safety review of small PWR NPPs will provide a reference and experience for laying down nuclear safety regulations and standards.

The review principle is set by referencing the latest findings of domestic and international research on the safety requirements of small PWR NPPs. It should be fully recognized that the establishment of safety requirements for small PWR NPPs must go through an iterative process of practices - cognition - more practices - further cognition. Such an attitude should also be held for the application of the review principle.

In view of current research status and nuclear safety review needs, the review principle is only applicable to demonstration projects with single reactor thermal power of several hundred MW or less and a few units at a single site. The review principles do not address the problems arising from numerous modular units arranged at a site, such as overall site risk, sharing of a control room or auxiliary control room, and an instrument or control panel among multiple units, and associated operator requirements. The requirements for these aspects will be gradually worked out according to research progresses and safety review needs.

Safety Goals

The overall nuclear safety goal of a small PWR NPP is to establish and maintain effective defenses against radiological hazards that threaten the safety of people, society, and environment.

The goal is supported by the objectives of radiation protection and technological safety. Radiation protection objective: ensuring that the radiation exposures within all operating small PWR NPPs or due to any planned discharge of radioactive materials from these plants remain below the prescribed limit and as low as reasonably achievable in a bid to mitigate the radiological consequences of any accident.

Technological safety objective: taking all reasonably practicable measures to prevent accidents at small PWR NPPs, and mitigate the consequences of any accident; ensuring with a high degree of confidence that radiological consequences are least severe and below the prescribed limit for all potential accidents considered in the design of a small PWR NPP, including the accidents of very low probability; virtually eliminating the possibility of discharging a significant amount of radioactive materials. Small PWR NPPs without off-site interventions should provide a higher level of protection to the public than large PWR NPPs with off-site interventions.

The virtual elimination of the possibility for discharging radioactive materials in large quantities means the virtual elimination of accident sequences that can result in high radiation doses or substantial radioactive discharges. It can be regarded as virtual elimination if certain conditions are in fact unlikely to occur, or highly unlikely to occur due to a high degree of confidence. This must be judged in conjunction with the phenomenological evaluation of important sequences of beyond design basis accidents, and the assessment of the engineering or design measures

associated with these phenomena, and the manner in which the accident source term is determined.

The above safety objectives allow small PWR NPPs to achieve a design goal: although the regulatory body may still require the setup of external interventions, the requirement for external interventions can nevertheless be technologically limited, or even exempted, as stated by IAEA in No. SSR-2/1 SAFETY OF NUCLEAR POWER PLANTS: DESIGN.

Concept of DiD

China's nuclear safety regulation namely the Design Requirements for Safety of Nuclear Power Plants (HAF102) has defined the DiD concept, that is, all safety related activities, including the organization, personnel behavior, or design, are safeguarded by multiple barriers, and even if a failure occurs, they will be detected, compensated, or corrected with appropriate measures in order to provide multi-layered protection against various transients, anticipated operational occurrences, and accidents resulting from on-site equipment failures or personnel activities and off-site events.

The concept of DiD is applied to design small PWR nuclear power plants, and form multi-layered defenses (inherent characteristics, equipment, and regulations) to prevent accidents, and offer adequate protection in case of failure in accident prevention.

While the overall five levels of DiD should be maintained for small PWR NPPs, the setting focus of the in-depth defense levels differs from that of conventional large LWRs, given the characteristics of the reactor type. Small PWR NPPs, for instance, should focus on the first three, or at most, the first four levels of DiD so that "the requirement for external interventions can be technologically limited, or even exempted."

The reasonableness in setting each defense level in depth should be justified by complete safety assessment.

Overall Design Basis

The state of small PWR NPPs is divided into four categories, including anticipated operational occurrences, design basis accidents, and beyond design basis accidents, in addition to normal operating conditions. The classification of these NPP states is determined by referencing the occurrence range of events, and the experience in existing and other reactor types. The frequency range of anticipated operational occurrences and design basis accidents is divided by referencing the occurring frequency of postulated initiating events (PIE); beyond design basis accidents are determined by the combination of the probability theory, determinism, and engineering judgment.

Small PWR NPPs comply with the national standards promulgated and applicable in China. We also recommend internationally recognized, practical, and authoritative standards and specifications be used as much as possible, which are generally approved or recognized by the nuclear safety regulatory authorities of the countries where they are developed.

Special attention should be paid to demonstrating the appropriateness and applicability of the standards referenced, and obtaining the consent from the NNSA. Consistency should be noted when using standards and specifications of different systems.

External Events Protection

Small PWR NPPs must be provided with reliable protection against external natural hazards and contrived events, which can be achieved through the design of the containment vessels or combined design of nuclear island powerhouses and containment vessels.

Regarding external natural hazards, the protection against the external natural hazards of design basis shall also be provided, and reserved with appropriate safety margins.

The protection against external contrived events shall meet the requirements of China's current nuclear safety regulations and standards, and properly mirror the latest international practices and requirements of associated regulations and standards.

Accident Source Item

In terms of conventional large LWR nuclear power plants, the early U.S. federal regulations "Reactor Site Criteria (10CFR100)" and the recent NRC regulatory guidelines "Radioactive Source Items for Evaluating Nuclear Power Plant Design Basis Accidents (RG 1.183)" have identified hypothetical accident source terms, but there are no domestic or international regulations or standards for small nuclear power plants.

As mentioned earlier, for the design concept of small PWR NPPs, a higher safety goal namely "virtually eliminating the possibility of discharging considerable amounts of radioactive materials" so that "the requirement for external interventions can be technologically limited, or even exempted" has been set. In other words, these plants should provide a higher level of protection to the public without off-site interventions than existing large PWR NPPs with off-site interventions, and the accident source term must be determined in a way that is compatible with this goal.

The important event sequences of design basis accidents and beyond design basis accidents at small PWR NPPs must be analyzed to identify the discharge of radioactive materials, from which conservative and enveloping source terms can be selected as source terms for site selection and contingency planning to evaluate whether the safety objectives are being met. The reasonableness of the model should be carefully analyzed, and when the radioactive material discharging mechanism is uncertain to a significant extent, or the required data is inadequate, moderate conservatism must be considered.

4.3 Czech Republic

Nuclear legislation

Czech nuclear legislation is relatively new - the new Atomic Act came into force on 1 January 2017, followed by 16 implementing decrees issued in 2017-2018, which are part of the Collection of Laws of the Czech Republic and are therefore legally binding. This new nuclear legislation is general and practically technology-neutral. As such, it applies to all types of nuclear installations, primarily VVER and PWR NPPs, radioactive material handling and storage facilities and, when applying a graded approach, also to nuclear research facilities. It therefore also applies to potential SMRs, irrespective of their technology system. Due to their specificity, in real licensing practice, the national regulator SÚJB would probably impose ad-hoc licensing requirements and procedures (graded approach) as it is applied in the case of the research reactors.

At the top of the pyramid of Czech nuclear legislation is the Atomic Act, which defines conditions for the peaceful utilization of nuclear energy and ionising radiation, including the activities

requiring a SÚJB licence. It gives the responsibility for governmental administration and management to the SÚJB and defines its power and competency.

Atomic Act - Act No. 263/2016 Coll., Atomic Act - was approved by the Parliament of the Czech Republic on July 14, 2016 and has entered into force on January 1, 2017. Its preparation has not been primarily driven by the need to set up completely new legal relationships, but rather to supplement and particularly refine existing legislation on the basis of experience gained over nearly two decades of application of previous legislation - Act No. 18/1997 Coll., on Peaceful Utilisation of Nuclear Energy and Ionising Radiation (the Atomic Act) and on Amendments and Additions to Related Acts, and the new recommendations of international institutions and other new findings.

The Atomic Act respects all changes presented both at the international level and within the Euratom. In addition to individual IAEA recommendations, these are, for example, Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations and Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.

The Atomic Act establishes activities for which a licence issued by the SÚJB is required. Besides the main licences for the siting, construction, operation and decommissioning of a nuclear installation, many more licences are concerned - see Chapter 7.2.2.

The Atomic Act authorizes the SÚJB, and in strictly defined cases other bodies of the state administration, to issue a set of related implementing regulations. They include but are not limited to the following regulations, which are mentioned in the text below:

- Decree No. 358/2016 Coll., on requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment;
- Decree No. 359/2016 Coll., on details of ensuring radiation extraordinary event management;
- Decree No. 360/2016 Coll., on radiation situation monitoring;
- Decree No. 361/2016 Coll., on security of nuclear installation and nuclear material;
- Decree No. 378/2016 Coll., on siting of a nuclear installation;
- Decree No. 408/2016 Coll., on management system requirements;
- Decree No. 409/2016 Coll., on activities especially important from nuclear safety and radiation protection viewpoint, special professional qualification and training of persons ensuring radiation protection of the registrant;
- Decree No. 422/2016 Coll., on radiation protection and security of a radioactive source;
- Decree No. 21/2017 Coll., on ensuring nuclear safety of a nuclear installation;
- Decree No. 162/2017 Coll., on requirements for safety assessment pursuant to the Atomic Act;
- Decree No. 329/2017 Coll., on the requirements for nuclear installation design.

A complete text of the Atomic Act and its implementing legal regulations is available at the SÚJB website.

In addition to the Atomic Act, there are other important acts, which defines special requirements for the construction of nuclear facilities, particularly:

- Act No. 100/2001 Coll., on Environmental Impact Assessment and on Amendment to Certain Related Acts (the Act on Environmental Impact Assessment);
- Act No. 183/2006 Coll., on Spatial Planning and Building Rules (Building Act).

System of licensing

The Atomic Act regulates and defines conditions for the performance of activities related to the use of nuclear energy, activities in exposure situations, activities in the field of radioactive waste management, carriage of radioactive or fissile materials, activities in the area of non-proliferation of nuclear weapons and other activities important for ensuring nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management, and security. A precondition for the performance of such activities is the licence issued by the SÚJB within an administrative procedure independent of the other administrative procedures including the procedure required under the Building Act.

Approvals of the state authorities include, in addition to a licence for the siting, construction and operation, a number of other separate licences issued by the SÚJB in accordance with Section 9 of the Atomic Act in various stages of the life cycle of a nuclear installation, specifically the following:

Pursuant to Section 9 of the Atomic Act, a licence shall be required for performing the following activities:

- a) the siting of a nuclear installation;
- b) the construction of a nuclear installation;
- c) the first physical start-up of a nuclear installation with a nuclear reactor;
- d) the first power-generation start-up of a nuclear installation with a nuclear reactor;
- e) the commissioning of a nuclear installation without a nuclear reactor;
- f) the operation of a nuclear installation;
- g) the individual phases of decommissioning of a nuclear installation, and;
- h) the carrying out of modifications affecting nuclear safety, technical safety and physical protection of a nuclear installation.

In the case of construction of a nuclear installation, the Building Act sets out a three-stage procedure for its licensing (site decision, construction permit and final inspection approval). According to the Building Act, the site decision on the siting is issued by a local department of planning and building management; the construction permit and operating permit (but also the permit or order to remove the structure) are then issued by special authority of planning and building management, which is the Ministry of Industry and Trade of the Czech Republic. Section 86) and Section 110) of the Building Act directly imposes liability upon the applicant and operator to present binding opinions or decisions of the authorities concerned according to special regulations, i.e. the Atomic Act.

Provided the related procedure involves interests protected by special regulations, the department of planning and building management shall decide in cooperation with or based on approval from the competent state administrative bodies protecting such interests. A competent state administrative body shall condition its approval upon fulfilment of conditions specified in its resolution issued in agreement with the special act entitling the body to do so. The bodies include in particular:

- technical inspection bodies dealing with conventional safety, including the safety of pressure components and electric systems;
- regional and municipal authorities in respect to fire safety, waste management, water consumption and effluents discharge;
- Czech Environmental Inspection – in respect to air pollution;

- Local body in charge of public health protection in respect to industrial safety in agreement with Act No. 258/2000 Coll., on Public Health Protection.

Act No. 100/2001 Coll., on Environmental Impact Assessment and on Amendment to Certain Related Acts (the Act on Environmental Impact Assessment) impose the obligation to assess installations from the viewpoint of their impact on the environment (the so-called “Environmental Impact Assessment”), within a separate procedure preceding the licensing procedure. This procedure involves the affected municipalities, authorities and the public represented by individuals as well as societies. The Ministry of the Environment is the relevant authority responsible for the issuance of a binding opinion concerning the environmental impact of the nuclear facility.

System of regulatory inspection and assessment

The SÚJB performs inspections of compliance with the requirements of the Atomic Act, regulations issued for its implementation and fulfilment of the decisions issued on the basis thereof, as well as the commitments arising from international treaties by which the Czech Republic is bound if they relate to the peaceful use of nuclear energy and ionising radiation, and fulfilment of the obligations set out by the Act on Metrology in the case of meters intended or used for measurement of ionising radiation and radioactive substances. The inspection activities of the SÚJB are regulated in detail by Section 200 and the following paragraphs of the Atomic Act and Act No. 255/2012 Coll., on Inspection (Inspection Code), which sets out the general rules for inspection authorities in the implementation of inspection activities. The two acts above provide the SÚJB with corresponding power and competency for the execution of the state supervision. For a detailed description of the above-mentioned requirements and approval procedures see Chapters 14, 15, 16, 17, 18 and 19 hereof.

Evaluation of site related factors

Site selection and evaluation of site-related characteristics are the crucial steps for the siting of a nuclear installation. Site suitability for the siting of a nuclear is one of the fundamentals of the “defence in depth” principle (IAEA Fundamental Safety Principles SF-1, 2006) and is assessed from the aspect of potential external natural or anthropogenic influences on a nuclear installation and from the aspect of site characteristics, which could affect the migration of radioactive substances in the environment.

The legal framework governing the issue of a siting licence which covers the nuclear safety and radiation protection aspects is established by the Atomic Act and its implementing regulations such as:

- Decree No. 378/2016 Coll., on siting of a nuclear installation;
- Decree No. 408/2016 Coll., on management system requirements;
- Decree No. 422/2016 Coll., on radiation protection and security of a radioactive source;
- Decree No. 359/2016 Coll., on details of ensuring radiation extraordinary event management;
- Decree No. 329/2017 Coll., on the requirements for nuclear installation design.

A licence for siting of a nuclear installation is issued by the SÚJB in accordance with the provisions of Section 9 of the Atomic Act. Application for the nuclear installation siting must be, in accordance with Annex 1 of the Atomic Act, documented by the following documentation:

1. management system program;

2. initial safety analysis report;
3. analysis of physical protection assurance needs and possibilities;
4. intention to ensure the monitoring of discharges from the nuclear installation;
5. monitoring program;
6. intention to ensure radiation extraordinary event management;
7. proposal of the safe decommissioning concept;
8. description of the method of quality assurance for the preparation of construction project implementation and
9. principles of quality assurance for the subsequent phases of the nuclear installation's life cycle.

Pursuant to the implementing regulations of the Atomic Act, in particular Decree No. 378/2016 Coll., on siting of a nuclear installation and in agreement with the IAEA (in particular NS-R-3 rev. 1, 2015; SSG-9, 2010; SR-85, 2015; NS-G-3.4, 2003, NS-G-3.6, 2005; SSG-18, 2011) and WENRA (Safety Reference Levels for Existing Reactors: update in relation to lessons learned from Tepco Fukushima Daiichi accident, 2014, and Guidance Document Issue T: Natural Hazards Head Document, 2015) recommendations, a nuclear installation should be designed while taking into account the historically most significant phenomena registered in the particular site for a nuclear installation and its vicinity, as well as a combination of natural phenomena, phenomena resulting from human activity and accident conditions due to these phenomena.

The Atomic Act further establishes the following requirements:

- Pursuant to Section 5(1) (c) – act in a way ensuring that the risk to natural persons and the environment is kept as low as can reasonably be achieved taking into account the current state of technical knowledge and economic and societal aspects.
- Pursuant to Section 5(2) (a) – as a matter of priority, ensure nuclear safety, the safety of nuclear items and radiation protection, while respecting the present level of science and technology and good practice.
- Pursuant to Section 46(2) (e) – ensure resilience and protection of the nuclear installation against the hazards resulting from the site characteristics of the site for a nuclear installation and from external influences.
- Pursuant to Section 47(1) – The site for a nuclear installation shall be evaluated in terms of:
 - a) its characteristics that can affect nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management and security during the life cycle of the nuclear installation, and
 - b) the impact of the nuclear installation on individuals, the general public, society and the environment.

And furthermore, pursuant to paragraph 2 or 3:

Siting of nuclear installations is prohibited in the area, whose characteristics pursuant to paragraph 1 (a) reduce the required level of nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management and security during the life cycle of the nuclear installation so that in terms of the current level of science and technology is not possible to remedy in the form of technical or administrative measures.

Prior to siting a nuclear installation, a baseline survey of the site for a nuclear installation shall be conducted with regard to radiation situation monitoring by measuring and

evaluating the baseline radionuclide content in the environment and food chain. The results of the baseline survey shall be retained for the needs of the complete decommissioning.

- Pursuant to Section 50 – ensure that already reached the level of nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management and security of another nuclear installation located in the site for a nuclear installation under construction is sited does not degrade.
- Pursuant to Section 54 (a) – continuously ensure, verify and document that the nuclear installation is capable of stable and safe operation.

Detailed requirements for the assessment of site characteristics and phenomena for a nuclear installation are laid down in Decree No. 378/2016 Coll., on siting of a nuclear installation, which also provides a list of assessed characteristics and phenomena in Section 3 as follows:

1) Site characteristics, assessed in site for a nuclear installation in terms of their capability to affect the nuclear safety, radiation protection, technical safety, radiation situation monitoring, radiation extraordinary event management and security during the life cycle of a nuclear installation, are the occurrences of:

- a. natural characteristics and phenomena, specifically
 1. seismicity;
 2. crack of the site for a nuclear installation as a result of a fault in the Earth's crust (hereinafter referred to as the "fault");
 3. floods;
 4. groundwater circulation;
 5. other geodynamic phenomena and geotechnical parameters of the foundation soils;
 6. climatic and meteorological phenomena;
 7. biological phenomena, and;
 8. natural fires;
- b. phenomena originated in human activity, specifically
 1. aircraft crash and crash of any other flying object;
 2. explosions and fires originated in human activity, and their products;
 3. collision with the protection or safety zone;
 4. impact of the nuclear installation already located in the area;
 5. strong vibrations;
 6. electromagnetic interference;
 7. eddy current;
 8. adverse impacts of air, road, rail and waterway transport;
 9. effects of the pipelines and power transmission lines;
 10. pollution of the air, rock environment, surface and groundwater, and
 11. operation of an installation where readily flammable, explosive, toxic, suffocating, corrosive or radioactive material are located or are released therefrom, and
- c. any other phenomenon that may adversely affect the nuclear safety, radiation protection, radiation situation monitoring, radiation extraordinary event management and security of a nuclear installation.

2) Characteristics of the site for a nuclear installation, assessed in terms of the impact of a nuclear installation on individuals, the public, society and the environment are the occurrences of phenomena capable of affecting the effect of a nuclear installation on the surrounding environment, specifically

- a. spread of radioactive substance via air, ground and surface water, and food chain, and
- b. population distribution and density, and its development.

Decree No. 378/2016 Coll., also regulates the general principles of assessment when pursuant to Section 4:

1. Site evaluation for a nuclear installation shall assess the extent to which the characteristics pursuant to Section 3 are capable of affecting nuclear safety, radiation protection, radiation situation monitoring, radiation extraordinary event management and security.
2. The results of the assessment of the site for a nuclear installation shall be compared with the features of the characteristics of the area, the achievement of which causes the siting of a nuclear installation to be prohibited.
3. The evaluation of the site for a nuclear installation shall include the assessment of:
 - a) the simultaneous effect and the interaction of characteristics pursuant to Section 3, their intensity and duration;
 - b) the future development of characteristics pursuant to Section 3 during the life cycle of a nuclear installation, and
 - c) the impact of installed capacity of a nuclear installation on the site for a nuclear installation in case of a nuclear installation with a nuclear reactor.
4. The site evaluation for a nuclear installation shall take place for the site area of a nuclear installation and up to such a distance thereof, which allows assessing the impact of characteristics pursuant to Section 3 on nuclear safety, radiation protection, radiation situation monitoring, radiation extraordinary event management and security, however at least up to the distance determined for that characteristic by this Decree.

It also specifies the characteristics, the achievement of which causes the siting of a nuclear installation to be prohibited.

The Atomic Act and Decree No. 378/2016 Coll., are followed by detailed SÚJB Safety Guides to assess compliance with the criteria relating to the above phenomena:

- SÚJB Safety Guide BN-JB-4.1, Siting of Nuclear Installation – Assessment of Natural Phenomena.
- SÚJB Safety Guide BN-JB-4.2, Siting of Nuclear Installation – Assessment of Phenomena Caused by Human Activity.

Approval process for design and construction of nuclear installation

Construction of a nuclear installation is one of the activities to which the SÚJB issues approval in accordance with the provision of Section 9 of the Atomic Act, from the nuclear safety and radiation protection point of view. Pursuant to Section 24 of the Atomic Act, the application for the construction of a nuclear installation shall be provided with the documentation as set out in Annex 1, Part 1 b) of the Atomic Act:

1. management system program;

2. limits and conditions;
3. programme of inspections for the construction stage;
4. preliminary safety report;
5. list of selected equipment, including classification of selected equipment into safety classes;
6. list of activities important to nuclear safety and description of the system of education, training and exercises for the personnel, including a description of the qualifications of the personnel;
7. description of the system of training for selected workers;
8. programme of construction of the nuclear installation, including the timetable;
9. preliminary commissioning plan for the nuclear installation;
10. preliminary probabilistic safety assessment;
11. preliminary physical protection assurance plan;
12. concept for the safe permanent shutdown of the installation to be licensed, including the method of the radioactive waste management produced;
13. monitoring program;
14. radiation extraordinary event analysis and assessment for the period from the commencement of construction of the nuclear installation to the commencement of decommissioning;
15. on-site emergency plan;
16. establishment of the emergency planning zone;
17. preliminary ageing management program;
18. proof that the financing of the radioactive waste management has been secured, if such waste is generated;
19. evaluation of quality assurance for the preparation of the construction of the nuclear installation;
20. description of the method of quality assurance for construction project implementation;
21. principles of quality assurance for the phases of the nuclear installation's life cycle following construction.

The documentation referred to in points 2, 3, 5, 11, 13, 15 and 16 shall be subject to approval by the SÚJB.

Pursuant to Section 5 of the Atomic Act, anyone who uses nuclear energy shall, among others, as a matter of priority, ensure nuclear safety and radiation protection; precede radiation extraordinary events and, if they occur, minimise their consequences, and act in a way ensuring that the risk to natural persons and the environment is kept as low as can reasonably be achieved taking into account the current state of technical knowledge and all economic and societal aspects.

Section 4(2) a) of the Atomic Act defines nuclear safety as: “the state and capability of nuclear installations and natural persons operating the installation to prevent uncontrolled self-sustaining fission chain reaction or release of radioactive substances or ionising radiation into the environment and to mitigate the consequences of accidents”.

In practice, compliance with the IAEA recommendations, in particular, SF-1 and SSR-2/1 (Rev. 1) and WENRA (“Safety of New NPP Designs” (2013) and “Safety Reference Levels for Existing Reactors” (2014)) is required for the project of a NPP in the Czech Republic by the SÚJB. These

requirements were also applied in the framework of the administrative procedure for a licence for siting of Temelín NPP Units 3 and 4, which took place between 2013 and 2014.

The new nuclear legislation of the Czech Republic introduces a concept of “practical elimination”. According to the Dec. No 329, §2, the “practical elimination” means a condition, state or event, the occurrence of which are considered physically impossible, or which are, with a high degree of confidence very unlikely.

The §4 of the Dec. 329 requires as design safety objectives that design of the nuclear installation, shall ensure practical elimination of following events:

- 1) a radiation accident for which there is not sufficient time for the implementation of urgent protective measures for the population (early radiation emergency); and
- 2) a radiation emergency requiring urgent protective measures for the public that cannot be limited in location or time (large radiation emergency).

The §7 of the Dec. 329 requires that the implementation of the concept of practical elimination is demonstrated using the conservative approach. The PSA used to prove a very low probability (low frequency of occurrence) of major radiation accidents must meet the requirements described in Dec. No. 162/2017 Coll.

Application of the concept of practical elimination may result in simplified siting of nuclear installations, in particular in respect of emergency planning zone. According to Annex No. 2 letter (a) Dec. 359 for radiation accidents with a frequency less than 1×10^7 per year there is no need to determine the emergency planning zone.

Application of DiD and improvement of nuclear safety in design extension conditions

The requirement for the application of DiD principle in the context of ensuring nuclear safety using multiple physical safety barriers and applying the safety functions to protect the integrity and functionality of these barriers at the various levels of DiD is set out in Section 6 of Decree No. 329/2017 Coll., on the requirements for nuclear installation design. In the context of ensuring compliance with requirements for the application of DiD, there is an obligation to ensure prevention and management of abnormal operating conditions, design basis accidents and design extension conditions including severe accidents.

Section 5 of the Decree sets out that nuclear installation design shall ensure resistance and protection of a nuclear installation against the risk arising from the occurrence of internal events during operation of a nuclear installation as a result of accidental failure of its SSCs, external events caused by the characteristics of the location for siting of a nuclear installation, human activities, and combination thereof.

Section 7 of the Decree sets out the obligation to ensure, in order to manage design extension conditions, such resistance of a nuclear installation which provides that a severe accident, which could lead to an early radiation accident or a large radiation accident, is a practically eliminated event, i.e. its occurrence is possible to consider as physically impossible or as very unlikely with a high degree of confidence.

Sections 12 and 29 of Decree No. 329/2017 Coll. on the requirements for nuclear installation design set out the requirements for nuclear installation design to ensure fulfilment of the principles

of the safe use of nuclear energy and reliable performance of the safety functions of structures, systems and components important to nuclear safety, including but not limited to:

- by using passive functions;
- by creating the ability to put them automatically into a state in which they contribute to the management of abnormal operation conditions or accident conditions in a nuclear installation;
- by automatic intervention or intervention on the basis of interventions by operators in accordance with internal regulations;
- by their physical separation and functional isolation, and
- by independence and backing up of systems and by using diversification.

Incorporation of proven technologies

The obligation to use proven methods, processes and technologies in the nuclear installation design is directly mentioned in Section 46 of the Atomic Act and the obligation to verify the required characteristics of SSCs important to nuclear safety is then set out in Sections 8 and 9 of Decree No. 329/2017 Coll., on the requirements for nuclear installation design.

Design for reliable, stable and manageable operation

In addition to the Atomic Act, the conditions for ensuring reliable, stable and manageable operation are elaborated in particular in Decree No. 21/2017 Coll., on ensuring nuclear safety of a nuclear installation. Among others, this Decree sets out the requirements for limits and conditions for maintaining the safety-significant physical and technological parameters of a nuclear installation within its operating state (Section 7), and for the processes and activities on a nuclear installation designed to prevent the development of accident conditions and to mitigate their consequences (Sections 26 to 30).

The requirements for the workplaces and systems used to control the nuclear installation, i.e. the requirements for functions and equipment of the main control room and the backup workplace (emergency control room) and means for managing a nuclear installation, are set out in Section 41 of Decree No. 329/2017 Coll., on the requirements for nuclear installation design.

Commissioning approval process

The requirements for the commissioning of nuclear installations in the Czech Republic as well as for all stages of their operation are laid down so as to avoid an accident with radiological consequences and in the case of its occurrence, to mitigate the consequences of such accident, which is in compliance with the main objective and principles of the Vienna Declaration on Nuclear Safety (VDNS).

The description of the approval process, in general for siting, designing and construction, operation and decommissioning of a nuclear installation are given in Chapter 7.

The legal framework for approval of the operation of a nuclear installation from the nuclear safety and radiation protection point of view is established by the Atomic Act and its implementing decrees, in particular:

- Decree No. 361/2016 Coll., on security of nuclear installation and nuclear material;
- Decree No. 21/2017 Coll., on ensuring nuclear safety of a nuclear installation;
- Decree No. 422/2016 Coll., on radiation protection and security of a radioactive source;

- Decree No. 359/2016 Coll., on details of ensuring radiation extraordinary event management;
- Decree No. 377/2016 Coll., on the requirements for the safe management of radioactive waste and on the decommissioning of nuclear installations or category III or IV workplaces;
- Decree No. 358/2016 Coll., on requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment;
- Decree No. 408/2016 Coll., on management system requirements;
- Decree No. 162/2017 Coll., on requirements for safety assessment pursuant to the Atomic Act;
- Decree No. 329/2017 Coll., on the requirements for nuclear installation design.

As further mentioned in Chapter 3.1.2, commissioning and operation of a nuclear installation are activities for which the SÚJB authorization is required under the provision of Section 9 of the Atomic Act as to nuclear safety, radiation protection, technical safety, emergency preparedness and security.

An application for the issue of authorization for the individual stages of nuclear installation commissioning must be, in accordance with Sections B, C, D and E of Annex to the Atomic Act, accompanied with the following documentation:

a) For construction stages of a nuclear installation:

1. management system programme;
2. limits and conditions;
3. programme of inspections for the construction stage;
4. preliminary safety report;
5. list of selected equipment, including classification of selected equipment into safety classes;
6. list of activities important to nuclear safety and description of the system of education, training and exercises for the personnel, including a description of the qualifications of the personnel;
7. description of the system of training for selected workers;
8. programme of construction of the nuclear installation, including the timetable;
9. preliminary commissioning plan for the nuclear installation;
10. preliminary probabilistic safety assessment;
11. preliminary physical protection assurance plan;
12. concept for the safe permanent shutdown of the installation to be licensed, including the method of the radioactive waste management produced;
13. monitoring program;
14. radiation extraordinary event analysis and assessment for the period from the commencement of construction of the nuclear installation to the commencement of decommissioning;
15. on-site emergency plan;
16. establishment of the emergency planning zone;
17. preliminary ageing management program;

18. proof that the financing of the radioactive waste management has been secured, if such waste is generated;
19. evaluation of quality assurance for the preparation of the construction of the nuclear installation;
20. description of the method of quality assurance for construction project implementation and
21. principles of quality assurance for the phases of the nuclear installation's life cycle following construction.

b) For the first loading of nuclear fuel into a reactor (physical start-up):

1. management system programme;
2. limits and conditions;
3. programme of in-service inspection;
4. final safety analysis report for the first physical start-up of a nuclear installation with a nuclear reactor;
5. list of selected equipment, including classification of selected equipment into safety classes;
6. list of activities important to nuclear safety and description of the system of education, training and exercises for the personnel, including a description of the qualifications of the personnel;
7. description of the system of training for selected workers;
8. neutron-physical characteristics of the nuclear reactor core;
9. certificate of successful completion of construction and certificate of readiness of the installation, personnel and internal regulations for the next phase of the nuclear installation's life cycle;
10. physical start-up program, including the timetable;
11. probabilistic safety assessment;
12. physical protection assurance plan;
13. statement that all checks of radiation extraordinary event response preparedness in the emergency planning zone under Section 156 (2)(c) through (g) of the Atomic Act have been conducted;
14. pre-operational ageing management program;
15. document demonstrating that safe radioactive waste management has been ensured; including the financing thereof, if radioactive waste is generated;
16. assessment of the quality of selected equipment;
17. emergency operating procedures and
18. severe accident management guidelines.

c) For stages following the first nuclear fuel loading into the reactor: (power start-up)

1. management system programme;
2. limits and conditions;
3. programme of operational controls checks;
4. final safety analysis report;
5. list of selected equipment, including classification of selected equipment into safety classes;

6. list of activities important to nuclear safety and description of the system of education, training and exercises for the personnel, including a description of the qualifications of the personnel;
7. description of the system of training for selected workers;
8. neutron-physical characteristics of the nuclear reactor;
9. certificate of successful completion of the first physical start-up and certificate of readiness of the installation, personnel and internal regulations for the operation of the nuclear installation;
10. program for the first power-generation start-up of a nuclear installation with a nuclear reactor and trial operation, including the timetable;
11. probabilistic safety assessment;
12. physical protection assurance plan;
13. operational ageing management program for the first power-generation start-up and trial operation of a nuclear installation with a nuclear reactor;
14. document demonstrating that safe radioactive waste management has been ensured, including the financing thereof, if radioactive waste is generated;
15. statement on verification of the expiry date of iodine prophylaxis antidotes distributed in accordance with Section 156(2) d);
16. emergency operating procedures and
17. severe accident management guidelines.

After a positive evaluation of the above-mentioned documentation, the SÚJB issues the approvals for the individual phases of the reactor commissioning, whilst the programme of the phases, proposed security method, changes in security assurance, proposed decommissioning method, on-site emergency plan, in-service inspection programme, as well as the Limits and Conditions for the safe operation of a nuclear installation, are subject to separate approval by the SÚJB.

Operation

Application for issuing the authorization for the nuclear installation operation must be, in accordance with Appendix F to the Atomic Act, accompanied with the following documentation:

1. management system programme;
2. limits and conditions;
3. programme of in-service inspection;
4. final safety analysis report;
5. list of selected equipment, including classification of selected equipment into safety classes;
6. neutron-physical characteristics of the nuclear reactor core;
7. list of activities important to nuclear safety and description of the system of education, training and exercises for the personnel, including a description of the qualifications of the personnel;
8. description of the system of training for selected workers;
9. certificate of readiness of the installation, personnel and internal regulations for the operation of the nuclear installation;
10. evaluation of the results of the first power-generation start-up of a nuclear installation with a nuclear reactor;

11. evaluation of the results of trial operation if this is the first licence for the operation of a nuclear installation;
12. operational programme, including the timetable;
13. probabilistic safety assessment;
14. physical protection assurance plan;
15. decommissioning plan;
16. estimation of decommissioning costs;
17. operational ageing management programme;
18. document demonstrating that safe radioactive waste management has been ensured, including the financing thereof, if radioactive waste is generated;
19. emergency operating procedures and
20. severe accident management guidelines.

After a positive evaluation of the above-mentioned documentation, the SÚJB issues the authorization for nuclear installation, whilst changes in the documentation, approved in previous stages, are subject to separate approval by the SÚJB. The licence for operation is not time-limited by law.

The above list of national nuclear legislation and licensing procedures represents only the basis that would govern the licensing process for the implementation of SMRs in the Czech Republic, in particular site evaluation, design process and commissioning followed by commercial operation. In addition to the above requirements, other requirements would apply in the case of the construction of such a facility, particularly in the areas of radiation protection, emergency preparedness and emergency response, including physical protection requirements. Some licensing procedures may also involve environmental non-governmental organisations, which may complicate the licensing process for a nuclear installation.

4.4 Finland

The legal basis for the use of nuclear energy in Finland is laid down in the Nuclear Energy Act (990/1987) and the Nuclear Energy Decree (161/1988). The Radiation and Nuclear Safety Authority of Finland (STUK) has published guidelines to ensure safe usage of nuclear energy and that the objectives of legislation for the safety of nuclear energy are achieved [28], [29]. STUK has also published Regulatory Guides on Nuclear Safety and Security (the YVL guides), which contain instructions and rules for operators of nuclear facilities and suppliers of nuclear SSCs. The guides are divided into five groups [30]:

- Group A: Safety management of a nuclear facility (12 separate guides: YVL A.1-YVL A.12);
- Group B: Plant and system design (8 guides: YVL B.1-YVL B.8);
- Group C: Radiation safety of a nuclear facility and environment (7 guides: YVL C.1-YVL C.7);
- Group D: Nuclear materials and waste (7 guides: YVL D.1-YVL D.7);
- Group E: Structures and equipment of a nuclear facility (13 guides: YVL E1-YVL E.13).

The French AFCEN codes (i.e. RCC-M, RCC-E, ...), German KTA and ASME Boiler and Pressure Vessel code are accepted as supporting standards for the YVL guides.

The different phases of the current Finnish nuclear facility licensing process are shown in Figure 12. The phases are identical regardless of reactor size. The licensing procedure was

prepared for large power reactors and is not necessarily fully applicable to SMR licensing. Construction of SMRs may deviate from the construction of large power reactors in many ways [31]:

- A plant may include several separate reactor units that are constructed at different times (the construction may be timed so that one or several units are already in operation when the construction of the next units is started);
- The new-build project is not necessarily backed by a traditional NPP operator and electricity (energy) supplier, but an industrial company (e.g. chemical industry, steel manufacturer) that wants to use SMRs to provide energy directly and exclusively for its manufacturing process or the SMRs primarily serve a different purpose other than electricity generation (e.g. seawater desalination);
- The construction time of a SMR unit is significantly shorter compared to large power reactors.

Current Finnish nuclear legislation and regulation does not exclude applying for a decision-in-principle for several reactors or facility sites at the same time. However, the common practice is to provide a license for only one reactor at a time.

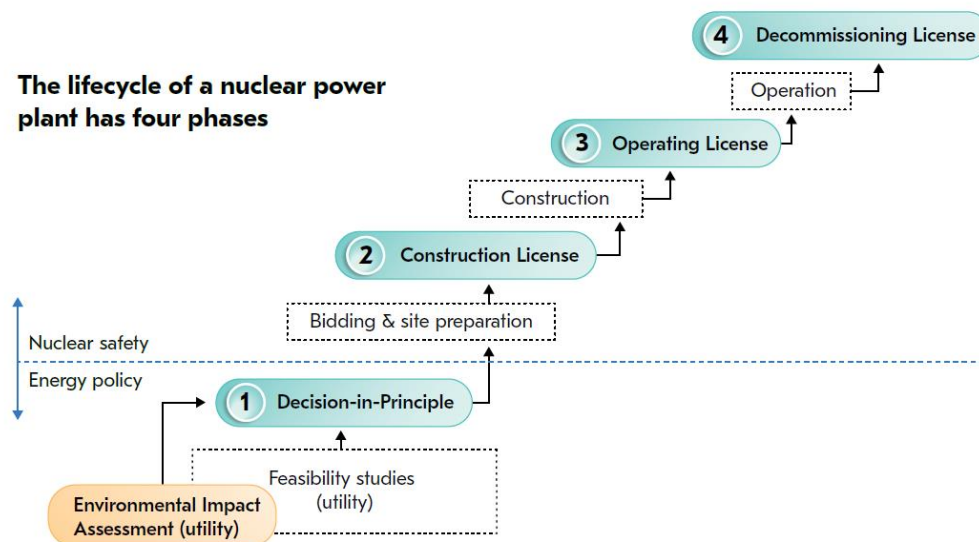


Figure 12. Licensing phases of Finnish nuclear facilities. Each phase is assessed by STUK before they are accepted [31].

In June 2017 the Ministry of Economic Affairs and Employment (MEAE) established a working group to explore the objectives, development measures and alternative solutions for safe and efficient nuclear energy and radioactive waste management well into the future. It was recognized that SMRs may require changes in waste management principles in Finland [32]. In 2019, Finnish National Cooperation Group on Nuclear Safety (appointed by MEAE in 2016, operates under STUK) reviewed the applicability of Finnish nuclear regulation with respect to SMRs and made an initiative in October 2019 to MEAE to reform the legislation and regulation. According to the cooperation group, MEAE should tackle the following issues [33]:

- Approval for nuclear facility locations and design certification should be included in the updated Nuclear Energy Act;
- Waste management cooperation improvement (between the licensees);
- Improvement of decision-making rights concerning waste disposal facilities so that governmental decisions-in-principle are not needed for the extension of their operation;

- National expertise should be ensured in view of possible deployment of SMRs.

MEAE established a specific working group in October 2019 to assess the need to amend the Nuclear Energy Act. It was asked to assess the current state and development outlook of nuclear energy production, the life cycle of nuclear facilities and the related fuel cycle as well as the needs to develop the legislation on the final disposal of spent fuel. The final report of the working group was submitted to Minister of Economic Affairs on 27 August 2020 [34], [35].

Hearings with energy companies and organizations (especially concerning SMRs), environmental actors, municipalities, energy intensive industries, and labour market organizations were organised. The working group concluded that an overall reform of nuclear energy legislation in Finland should be initiated because of the following reasons [35]:

- The existing legislative framework is complex and difficult to understand.
- Significant changes have taken place in the operating environment of nuclear facilities and they are expected to continue following the involvement of new players and the introduction of new operating models and technologies.
- Emission reductions and actions to combat climate change have become the key topics in energy and climate policy.

The working group outlined the key principles for the reform. Compliance with international agreements, commitments and best practices in nuclear energy usage will be ensured. Keeping in place a licensing system that covers the entire life cycle of nuclear facilities and transparently and effectively upholds democratic decision-making is necessary. Some improvements are required to the present situation, e.g. license processing and its predictability and reducing overlaps in the decision making. The used concepts should be understandable. The working group has made proposals for the definition of nuclear facilities, nuclear material, spent nuclear fuel, and nuclear and other radioactive waste [35].

STUK coordinates and implements the Finnish Support Programme (FINSP) to the “IAEA safeguards” in which “Safeguards by Design for Small Modular Reactors” task was assigned to FINSP. The SMR task was offered to FINSP as there is a lot of interest in SMRs at Finnish universities, companies and research organizations (e.g., Lappeenranta University of Technology and VTT Technical Research Centre of Finland are both developing their own SMR designs for district heating and Fortum Ltd. has had its own interest in SMRs at least since 2013) [36], [37].

4.5 United Kingdom

The legal framework for the nuclear industry is based around the Health and Safety at Work etc. Act 1974 (HSWA) [38], the Energy Act 2013 [39] and the Nuclear Installations Act 1965 [40]. HSWA [38] places duties on all employers, including those in the nuclear industry, to look after the health and safety of both their employees and the public. However, because of the particular hazards associated with the nuclear industry, including the potential for accidents to cause widespread harm and social disruption, further legislation is also in place, specifically the Nuclear Installations Act 1965 [40]. Additionally, there are provisions for nuclear regulations to be made under the Energy Act 2013 [39], as well as specific regulations under HSWA such as the Ionising Radiations Regulations 2017 (IRR17) [41] and Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPPiR) [42]. In the following, the above-mentioned acts and regulations are briefly described (for more details see Licensing nuclear installations document from November 2021 [43]).

Operators of nuclear installations in Britain are required to comply with the HSWA [38] and its relevant statutory provisions ('Part 1 of the Act [sections 1-54]; regulations made under section 15 and the existing statutory provisions, principally provisions of the Acts in schedule 1). The HSWA places a duty on employers to ensure, so far as is reasonably practicable, the health, safety and welfare at work of all their employees. It also imposes a duty on employers to ensure, so far as is reasonably practicable, that persons not in their employment are not exposed to risks to their health or safety as a result of the activities undertaken.

Relevant parts of the nuclear industry must comply with Nuclear Installations Act 1965 [40] which according to [43] has three key purposes:

- a) It requires the licensing of sites which are to be used for the installation or operation of nuclear reactors (except reactors forming part of a means of transport), and certain other classes of nuclear installations which have been prescribed. Currently the latter are prescribed by the Nuclear Installations Regulations 1971 (Statutory Instrument 1971/381) [44].
- b) It provides for control, via permit, of processes for the enrichment of uranium and the extraction of plutonium or uranium from irradiated matter and the application of associated security measures.
- c) It provides a special legal regime to govern the liability of nuclear site licensees and owners towards third parties for certain kinds of damage caused by nuclear matter on, or coming from, their sites.

The UK employs Safety Assessment Principles (SAPs) [45] to establish their expectations for safety for nuclear facilities. The IAEA safety standards also include fundamental principles [6], but these cover a wider scope than safety assessment. ONR's principles have therefore been drawn from the aspects of IAEA's principles that are relevant to the remit of the SAPs:

- Responsibility for safety (FP.1) (note: drawn from IAEA SF-1 Principle 1: Responsibility for safety);
- Leadership and management for safety (FP.2) (note: drawn from IAEA SF-1 Principle 3: Leadership and management for safety);
- Optimisation of protection (FP.3) (note: drawn from IAEA SF-1 Principle 5: Optimization of protection);
- Safety assessment (FP.4) (note: drawn from IAEA SF-1 Principle 3²³: Leadership and management for safety);
- Limitation of risks to individuals (FP.5) (note: drawn from IAEA SF-1 Principle 6: Limitation of risks to individuals);
- Prevention of accidents (FP.6) (note: drawn from IAEA SF-1 Principle 8: Prevention of accidents);
- Emergency preparedness and response (FP.7) (note: drawn from IAEA SF-1 Principle 9: Emergency preparedness and response);
- Protection of present and future generations (FP.8) (note: drawn from IAEA SF-1 Principle 7: Protection of present and future generations).

Licensees are expected to use three forms of analysis to establish adequate safety for fault and accident conditions, namely design basis analysis, probabilistic safety assessment and severe accident analysis. They are intended to check that the necessary high level of safety has been achieved and provide important input to the design, operation and emergency preparedness of

²³ Principle 3: Leadership and management for safety includes also safety assessment

the facility. The fundamental requirement in relation to health and safety in the UK is to do whatever is reasonably practicable to control and reduce risks to employees and the public. The latest Rev. 1 of SAP 2014 version [45] in addition to the lessons from Fukushima also take account of recent work by the International Atomic Energy Agency (IAEA), in particular the development of IAEA's design standard on the safety of nuclear power plants (SSR 2/1). As with the previous version of the SAPs, ONR believe that they are fully in line with IAEA guidance and standards. ONR acknowledge that these SAPs cannot reflect the breadth and depth of the entire suite of IAEA publications and so ONR explicitly identify those documents as relevant good practices within ONR Technical Assessment Guides (TAGs). The SAP document [45] in the introduction among other explains the purpose of SAP, gives regulatory background and discuss application of SAP. Later in the document besides fundamental principles also leadership and management for safety, regulatory assessment of safety cases, siting aspects, engineering principles, radiation protection, fault analysis, numerical targets, accident management and emergency preparedness, radioactive waste management, decommissioning and land quality management are presented. The SAPs contain principles and guidance. New facilities are also considered. One of the aims of the SAPs is to support the regulatory safety assessment of new (proposed) nuclear facilities. They represent ONR's view of good practice and ONR would expect modern facilities to satisfy their overall intent. Multi-facility sites are also considered.

A key principle of the UK's approach is that nuclear licensees are required to build, operate and decommission nuclear sites in a way that ensures that risks are kept as low as reasonably practicable. The requirement for risks to be reduced so far as is reasonably practicable or as low as reasonably practicable (ALARP) [46] is fundamental and applies to all activities within the scope of the HSWA [38] and NIA 1965 [40]. Put simply, it is a requirement to take all measures to reduce risk where doing so is reasonable. In most cases this is not done through an explicit comparison of costs and benefits, but rather by applying established relevant good practice and standards. The development of relevant good practice and standards includes ALARP considerations, so in many cases meeting them is sufficient. In other cases, either where standards and relevant good practice are less evident or not fully applicable, the onus is on the licensee to implement measures to the point where the costs of any additional measures (in terms of money, time or trouble – the sacrifice) would be grossly disproportionate to the further risk reduction that would be achieved (the safety benefit). Fundamental to demonstrating that nuclear safety risks are ALARP for a nuclear installation, is the requirement to produce a 'safety case', which demonstrates that nuclear related facilities and activities can be operated within safe limits and conditions.

Part 3 of The Energy Act 2013 (TEA 13) [39] established ONR as a statutory corporation. This makes ONR responsible for the enforcement of statutory provisions which are 'relevant statutory provisions' for the purposes of that Act. These provisions include sections 1; 3-6; 22 and 24A of NIA 1965 [40] as well as the Nuclear Industries Security Regulations 2003 (NISR) [47], the Nuclear Safeguards Regulations 2019 (NSR19) [48] and the Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations (CDG 2009) [49] as they apply to the carriage of radioactive material for civil purposes.

NISR 2003 [47] provides for the regulation of the civil nuclear industry for security purposes and provides definitions of nuclear material (NM), other radioactive material (ORM) and sensitive nuclear information (SNI) for security purposes. It contains provisions for the security of nuclear premises, the transport of nuclear material and the security of SNI. NISR 2003 [47] is enforced by ONR's Civil Nuclear Security and Safeguards (CNSS) Division, which has produced Security

Assessment Principles (SyAPs) and guidance to assist inspectors make consistent regulatory judgements. For licensing, the applicant must produce a security plan that meets specified outcomes. ONR will expect the security plan to be developed as security risks change during construction and operation. Security assessments will draw from the SyAPs and guidance to meet regulatory expectations.

NSR19 [48] deals with nuclear safeguards regulations. Nuclear safeguards are measures to verify that states comply with their international obligations not to use nuclear materials (plutonium, uranium and thorium) for nuclear explosives purposes. Global recognition of the need for such verification is reflected in the requirements of the Treaty on the Non-Proliferation of Nuclear Weapons, which relates to the application of safeguards by the International Atomic Energy Agency (IAEA). NSR19 regulations require operators of qualifying civil nuclear facilities to maintain standards and arrangements for nuclear material accountancy in accordance with an accountancy and control plan (ACP). NSR19 [48] is a relevant statutory provision of TEA 13 [39].

The Ionising Radiations Regulations 2017 (IRR17) [41] and their associated approved code of practice cover the protection of workers and the public from work involving ionising radiations. They include a general duty to keep exposures ALARP and, among other requirements, set limits on such exposure. They implement, in part, the latest revision of the European Atomic Energy Community's (Euratom) Basic Safety Standards Directive.

The Radiation (Emergency Preparedness and Public Information) Regulations 2019 (REPP19) [42] provides a framework for the protection of the public and workers from and in the event of radiation emergencies that originate from sites. REPP19 [42] requires the licensee to undertake a hazard evaluation and consequence assessment and present the conclusions in a consequences report that is used to inform off-site emergency planning. The licensee must also produce an operator's on-site emergency plan where the hazard evaluation shows a radiation emergency may arise. The regulations also place duties on the local authority, to prepare (and if necessary, implement) an off-site emergency plan for dealing with the consequences of any reasonably foreseeable radiation emergency in an area surrounding the site that is determined by the local authority. The local authority is also required to ensure that relevant information is supplied to the affected population in the event that a radiation emergency should occur.

Finally, Annex 5 of Licensing nuclear installations document [43] for SMRs states that SMRs are usually based on proven water-cooled reactor designs, but on a reduced scale with electrical power outputs less than 300 MW. SMRs are called modular reactors because they can be manufactured in factories and transported to site for installation. There is currently no legal definition of an SMR and NIA 1965 [40] does not contain any specific provisions for them. The UK regulatory approach to SMRs has not been decided at the time of writing Licensing nuclear installations document [43]. The UK Government has committed to opening the generic design assessment (GDA) process to SMR technologies in 2021. The government has also created an advanced nuclear fund, aiming to develop an SMR design by 2030.

5 Summary of specific safety elements for SCW-SMR

The review and discussion documented in Sections 2, 3, and 4 are summarized in this section. First, the guiding principles are summarized. This is followed by the outline of generic and specific safety elements for ECC-SMART.

No safety elements developed specifically for the SCW-SMR have been found. Also, with the exception of Canada and China, no specific regulations for SMRs appear to be in force in the countries reviewed within this project. Specific requests for SCW-SMR are also not yet available in the Chinese and Canadian SMR regulations.

This is why safety criteria covering particular aspects of the ECC-SMART conceptual designs will have to be developed in parallel with the specific design features featured by the designs under ECC-SMART. This section is therefore closed with the recommendations for the future re-evaluations, refinements and potential enhancements of the safety elements.

5.1 Guiding principles for review and selection of the relevant safety elements

The guiding principles are developed and substantiated in Section 2. They are summarized for convenience in Table 4.

Table 4. Guiding principles for review and selection of the relevant safety elements

Guiding Principle	Section
All compatible ²⁴ safety elements of new reactors (3rd Gen. NPPs) apply to ECC-SMART.	2.1
Safety elements specific of SMR designs apply to ECC-SMART depending on the design, hence on a case-by-case basis. Safety demonstration dealing with passive safety systems performance and plant accident response need conservative safety margins.	2.2
All the top-level safety requirements identified by GIF apply to ECC-SMART, in particular, the elimination of DEC-B scenarios.	2.3
No new challenging safety requirement specific of ECC-SMART in the accident domain is foreseen. Quantitatively more demanding phenomena affecting safety need robust safety demonstration and conservative safety margins.	2.4

5.2 Safety Elements for ECC-SMART

ECC-SMART comprises a set of nuclear reactor designs representing an intersection between 3rd Gen. NPPs (new reactors), SMRs and 4th Gen. NPPs. This is why safety principles, requirements and objectives of ECC-SMART should come from all these domains.

The majority of the IAEA high level safety requirements apply to the ECC-SMART. The most important IAEA safety standards and reports with their scope of application (e.g., full, partial) to ECC-SMART are listed in Table 3. IAEA SSR-2/1 [2] should be used during the SCW-SMR development as a starting point for the design criteria development.

²⁴ Within the current context, 'compatible' is intended as applicable to ECC-SMART in terms of design and performance.

The SCWR is among the selected GIF designs. As such, ECC-SMART should meet the relevant GIF safety goals. In summary (for discussion see Section 3.2.1):

- The frequency of DEC-A scenarios shall be very low and its damage extension limited.
- DEC-B scenarios shall be eliminated.

The SDC document [13] was developed for Sodium Cooled fast reactors and cannot be directly applied to the SCWR. It can however be used as a good example how the IAEA standards (e.g., SSR-2/1 [2]) and guidelines can be applied to a Generation IV system through suitable adaptation of IAEA documents (for discussion see Section 3.2.2). A preliminary screening of the applicability of criteria developed in the SDC document [13] is given in Annex 1 and in Section 3.2.3. For multiple units, the revised document [20] introduces the term “multi-module unit” for SMRs emphasizing the differences between the small and large multi-unit NPPs. This is the reason why the safety specific aspects of so-called “multi-module unit” SMRs should be more deeply evaluated in the frame of the ECC-SMART project or possible follow-up project(s). For more discussion about this topic see Sub-section 3.2.2.1 and 3.2.2.2.

5.3 Recommendations

Some relevant aspects to be considered in the future refined conceptual design and in the parallel development of safety criteria, respective R&D programmes and PIRT, are already identified in Sections 3.2.6 and 3.2.7.

An exhaustive identification of the specific design and performance features should be carried out in the future. Within the scope of this project, the necessary preparations to develop the PIRT are being planned to support the implementation of the ISAM methodology (Section 3.2.4.1) in possible future projects.

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Criterion	39												
Criterion	40												
Criterion	41	5.69	5.70										
Criterion	42												
Criterion	42bis	5.71	5.72	5.73	5.74	5.75	5.76						
Criterion	43												
Criterion	44	6.1	6.2	6.3									
Criterion	45	6.3bis	6.3ter	6.3quater									
Criterion	46	6.4	6.5	6.6	6.6bis								
Criterion	47	6.7	6.8	6.9	6.10	6.11	6.12						
Criterion	48	6.13	6.14	6.14bis	6.14ter	6.15	6.15bis	6.15ter	6.16	6.16bis	6.16ter	6.16quater	6.16quinques
Criterion	49												
Criterion	50												
Criterion	51	6.17	6.17bis										
Criterion	52	6.18	6.19	6.19bis									
Criterion	53												
Criterion	54	6.19A	6.19B										
Criterion	55	6.19C											
Criterion	56	6.20	6.21										
Criterion	57	6.22	6.23	6.24									
Criterion	58	6.25	6.26										
Criterion	59	6.27	6.28	6.28A	6.28B	6.29	6.30						
Criterion	60	6.31	6.31bis										
Criterion	61												
Criterion	62	6.32	6.33	6.33bis									
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Criterion	64	6.37											
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Criterion	66	6.39	6.40	6.40A									
Criterion	67	6.41											
Criterion	68	6.42											
Criterion	69	6.43	6.44	6.44A	6.44B	6.44C	6.44D	6.45	6.45A				
Criterion	70												
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Criterion	72	6.47											
Criterion	73												
Criterion	74	6.48	6.49										
Criterion	75	6.50	6.51	6.52	6.53	6.54	6.54bis	6.54ter					
Criterion	76												
Criterion	76bis	6.55											
Criterion	76ter												
Criterion	77	6.56	6.57	6.58	6.58A								
Criterion	78	6.59	6.60										
Criterion	79	6.61	6.62	6.63									
Criterion	80	6.64	6.65	6.66	6.67	6.68	6.68A	6.68bis					
Criterion	81	6.69	6.70	6.71	6.72	6.73	6.74	6.75	6.76				
Criterion	82	6.77	6.78	6.79	6.80	6.81	6.82	6.83	6.84				

Annex 2: The comments of the review on SDC document [13]

The review the SDC document [13] performed by the safety experts of BME NTI is documented below. The relevant pages from the document SDC document [13] are depicted on the left with the comments on the right.

3. MANAGEMENT OF SAFETY IN DESIGN

Criterion 1: Responsibilities in the management of safety in plant design

An applicant for a licence to construct and/or operate a nuclear power plant shall be responsible for ensuring that the design submitted to the regulatory body meets all applicable safety requirements.

3.1 All organizations, including research and design organizations, engaged in activities important to the safety of the design of a nuclear power plant shall be responsible for ensuring that safety matters are given the highest priority.

Criterion 2: Management system for the plant design¹⁴

The design organization shall establish and implement a management system for ensuring that all safety requirements established for the design of the plant are considered and implemented in all phases of the design process and that they are met in the final design.

3.2 The management system shall include provision for ensuring the quality of the design of each structure, system and component, as well as of the overall design of the nuclear power plant, at all times. This includes the means for identifying and correcting design deficiencies, for checking the adequacy of the design and for controlling design changes through a corrective action program.

3.3 The design of the plant, including subsequent changes, modifications or safety improvements, shall be in accordance with established procedures that call on appropriate engineering codes, standards and related supporting research results, and shall incorporate relevant requirements and design bases. Interfaces shall be identified and controlled.

3.4 The adequacy of the plant design, including design tools and design inputs and outputs, shall be verified and validated by individuals or groups separate from those who originally performed the design work. Verification, validation and approval of the plant design shall be completed as soon as is practicable in the design and construction processes, and in any case before operation of the plant is commenced. Any research activity performed to support the safety justification of the plant design shall be subject to quality assurance. Clear links to experimental records and results shall be established and maintained. Design choices made during the design process shall be recorded with adequate tracking.

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Summary of Comments on pages with comments
Ref.12_GIF.SDC.report.rev1_commented.by.BME.NTI.pdf

Page: 1

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Modification not accepted. Modification not necessary.

conditions for the full duration of its design life and can be safely decommissioned, and that the restoration of the facility and the environment is minimized.

4.5 The design for a nuclear power plant shall be such as to ensure that the safety requirements of the operating organization, the requirements of the regulatory body and the requirements of relevant legislation, as well as applicable national and international codes and standards, are all met, and that due account is taken of human capabilities and limitations and of factors that could influence human performance. Adequate information on the design shall be provided for ensuring the safe operation and maintenance of the plant, and to allow subsequent plant modifications to be made. Recommended practices shall be provided for incorporation into the administrative and operational procedures for the plant (i.e. the operational limits and conditions).

4.6 The design shall take due account of relevant available experience that has been gained in the design, construction and operation of other nuclear power plants, and of the results of relevant research programmes.

4.7 The design shall take due account of the results of deterministic safety analyses and probabilistic safety analyses, to ensure that due consideration is given to the prevention of accidents and to mitigation of the consequences of any accident conditions.

4.8 The design shall be such as to ensure that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume, by means of appropriate design measures and operational and decommissioning practices.

Criterion 7: Application of defence in depth

The design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable.

The design of a nuclear power plant shall be such that level 4 of the defence in depth and the associated safety design for prevention and/or mitigation of severe accident conditions shall be incorporated, in order that significant radioactive release can be considered as belonging to the residual risk.

4.9 The defence in depth concept shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment and ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails.

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Page: 2

Number: 1 Author: BME.NTI Subject: Highlight Date: 15.07.2021 10:05:34 +02:00
Modification not accepted. The original text in SSR 271 (Impact on environment) has a broader meaning.

Number: 2 Author: BME.NTI Subject: Highlight Date: 15.07.2021 10:05:47 +02:00
Modification not accepted. There is no reason for emphasizing only the D10 level 4.

4.10 The design shall take due account of the fact that the existence of multiple levels of defence is not a basis for continued operation in the absence of one level of defence. All levels of defence in depth shall be kept available at all times and any relaxations shall be justified for specific modes of operation.

4.11 The design:

- (a) Shall provide for multiple physical barriers to the release of radioactive material to the environment;
- (b) Shall be conservative, and the construction shall be of high quality, so as to provide assurance that failures and deviations from normal operation are minimized, that accidents are prevented as far as is practicable and that a small deviation in a plant parameter does not lead to a cliff-edge effect;
- (c) Shall provide for the control of plant behaviour by means of inherent and engineered features, such that failures and deviations from normal operation requiring actuation of safety systems are minimized or excluded by design to the extent possible;
- (d) Shall provide for supplementing the control of the plant by means of automatic actuation of safety systems, such that failures and deviations from normal operation that exceed the capability of control systems can be controlled with a high level of confidence, and the need for operator actions in the early phase of these failures or deviations from normal operation is minimized;
- (e) Shall provide for systems, structures and components and procedures to control the course of and as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems;
- (f) Shall provide multiple means for ensuring that each of the fundamental safety functions is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation.

It shall consider the benefit of implementing passive safety features for shutdown and cooling.

4.12 To ensure that the concept of defence in depth is maintained, the design shall prevent as far as is practicable:

- (a) Challenges to the integrity of physical barriers;
- (b) Failure of one or more barriers;

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- (c) Failure of a barrier as a consequence of the failure of another barrier;
- (d) The possibility of harmful consequences of errors in operation and maintenance.

4.13 The design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant.

4.13A. The levels of defence in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.

Criterion 8: Interfaces of safety with security and safeguards

Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another.

It shall take into account the potential for adverse effects on safety or security when designing, and before implementing changes to, the plant configurations, facility conditions, engineering and administrative controls.

Criterion 9: Proven engineering practices

Items important to safety for a nuclear power plant shall be designed in accordance with the relevant national and international codes and standards.

4.14 Items important to safety for a nuclear power plant shall preferably be of a design that has previously been proven in equivalent applications, and if not shall be items of high quality and of a technology that has been qualified and tested.

4.15 National and international codes and standards that are used as design rules for items important to safety shall be identified and evaluated to determine their applicability, adequacy and sufficiency, and shall be supplemented or modified as necessary to ensure that the quality of the design is commensurate with the associated safety function.

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4.16. Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, safety shall be demonstrated by means of appropriate supporting research programmes, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications. The new design or feature or new practice shall also be adequately tested to the extent practicable before being brought into service, and shall be monitored in service to verify that the behaviour of the plant is as expected.

Criterion 10: Safety assessment ¹⁰⁹

Comprehensive deterministic safety assessments and probabilistic safety assessments shall be carried out throughout the design process for a nuclear power plant to ensure that all safety requirements on the design of the plant are met throughout all stages of the lifetime of the plant, and to confirm that the design as delivered meets requirements for manufacture and for construction, and as built, as operated and as modified.

4.17. The safety assessments shall be commenced at an early point in the design process, with iterations between design activities and confirmatory analytical activities, and shall increase in scope and level of detail as the design programme progresses.

4.18. The safety assessments shall be documented in a form that facilitates independent **substantiation**.

Criterion 11: Provision for construction

Items important to safety for a nuclear power plant shall be designed so that they can be manufactured, constructed, assembled, installed, erected, **inspected and tested** in accordance with established processes that ensure the achievement of the design specifications and the required level of safety.

4.19. In the provision for construction and operation, due account shall be taken of relevant experience that has been gained in the construction of other similar plants and their associated structures, systems and components. Where best practices from other relevant industries are adopted, such practices shall be shown to be appropriate to the specific nuclear application.

Criterion 12: Features to facilitate waste management and decommissioning

Special consideration shall be given at the design stage of a nuclear power plant to the incorporation of features to facilitate radioactive **and chemical** waste management and the future decommissioning and dismantling of the plant.

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 100650 +02'00'
4.18s For deterministic and safety assessment, the special features and parameters of the SCW reactors shall be considered. The applied methods and tools (e.g. computer codes) have to be verified and validated for these special conditions.			
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Number: 3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 100639 +02'00'
Not relevant for SCWR			

4.20. In particular, the design shall take due account of

- (a) The choice of materials, so that amounts of radioactive waste will be minimized to the extent practicable and decontamination will be facilitated,
- (b) The access capabilities and the means of handling that might be necessary,
- (c) The facilities necessary for the treatment and storage of radioactive **and chemical** waste generated in operation and provision for managing the radioactive waste that will be generated in the decommissioning of the plant.

31) The disposal and/or reuse of the sodium after the reactor final shutdown shall be investigated.

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Not relevant for SCWR			
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Criterion 15: Design limits

A set of design limits consistent with the key physical parameters for each item important to safety for the nuclear power plant shall be specified for all operational states and for accident conditions.

5.4. The design limits shall be specified and shall be consistent with relevant national and international standards and codes, as well as with relevant regulatory requirements

Criterion 16: Postulated initiating events

The design for the nuclear power plant shall apply a systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design.

5.5. The postulated initiating events shall be identified on the basis of engineering judgement, **operating experience** and a combination of deterministic assessment and probabilistic assessment. A justification of the extent of usage of deterministic safety analysis and probabilistic safety analysis shall be provided, to show that all foreseeable events have been considered.

5.6. The postulated initiating events shall include all foreseeable failures of structures, systems and components of the plant, as well as operating errors and possible failures arising from internal and external hazards, whether in full power, low power or shutdown states.

5.7. An analysis of the postulated initiating events for the plant shall be made to establish the preventive measures and protective measures that are necessary to ensure that the required safety functions will be performed.

5.8. The expected behaviour of the plant in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority:

- (1) A postulated initiating event would produce no safety significant effects or would produce only a change towards safe plant conditions by means of inherent characteristics of the plant.
- (2) Following a postulated initiating event, the plant would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event.

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 1006:16 -02'00'
Important criterion. The inclusion of the very different design limits of SCW reactors should be considered (rather in Reg. 54 than in the criterion itself)			
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Acceptable modification (in spite of lacking the operating experience with SCWR). Operating and maintenance experience with SCW cooled coal plants could serve as an appropriate basis.			

- (3) Following a postulated initiating event, the plant would be rendered safe by the activation of safety systems that need to be brought into operation in response to the postulated initiating event.
- (4) Following a postulated initiating event, the plant would be rendered safe by following specified procedures.

5.9. The postulated initiating events used for developing the performance requirements for the items important to safety in the overall safety assessment and detailed analysis of the plant shall be grouped into a number of representative event sequences that identify bounding cases and that provide the basis for the design and the operational limits for items important to safety.

5.10. A technically supported justification shall be provided for exclusion of any initiating event **from the design analysis** that is identified in accordance with the comprehensive set of postulated initiating events.

5.11. Where prompt and reliable action would be necessary in response to a postulated initiating event, provision shall be made in the design for automatic safety actions for the necessary actuation of safety systems, to prevent progression to more severe plant conditions.

5.12. Where prompt action in response to a postulated initiating event would not be necessary, it is permissible for reliance to be placed on the manual initiation of systems or on other operator actions. For such cases, the time interval between detection of the abnormal event or accident and the required action shall be sufficiently long, and adequate procedures (such as administrative, operational and emergency procedures) shall be specified to ensure the performance of such actions. An assessment shall be made of the potential for an operator to worsen an event sequence through erroneous operation of equipment or incorrect diagnosis of the necessary recovery process.

5.13. The operator actions that would be necessary to diagnose the state of the plant following a postulated initiating event and to put it into a stable long term shutdown condition in a timely manner shall be facilitated by the provision of adequate instrumentation to monitor the status of the plant, and adequate controls for the manual operation of equipment.

5.14. The design shall specify the necessary provision of equipment and the procedures necessary to provide the means for keeping control over the plant and for mitigating any harmful consequences of a loss of control.

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Unnecessary modification, the PIE is excluded from the design (or design basis), not from the analysis. The original IAEA requirement is acceptable.			

5.15 Any equipment that is necessary for actions to be taken in minimal response and recovery processes shall be placed at the most suitable location to ensure its availability at the time of need and to allow safe access to it under the environmental conditions anticipated.

Criterion 17: Internal and external hazards

All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the nuclear power plant, shall be identified and their effects shall be evaluated. Hazards shall be considered in designing the layout of the plant and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the plant.

17A Items important to safety shall be designed and located, with due consideration of other implications for safety, to withstand the effects of hazards or to be protected, in accordance with their importance to safety, against hazards and against common cause failure mechanisms generated by hazards.

17B For multiple unit plant sites, the design shall take due account of the potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously.

Internal hazards

5.16 The design shall take due account of internal hazards such as fire, explosion, flooding, missile generation, collapse of structures and falling objects, pipe whip, jet impact, release of fluid from failed systems or from other installations on the site. **Hot sodium chemical reaction with air, water and other materials, including associated pressure waves, temperature increase and product releases, e.g. hydrogen.** Appropriate features for prevention and mitigation shall be provided to ensure that safety is not compromised.

External hazards, 171

5.17 The design shall include due consideration of those natural and human induced events of origin external to the plant that have been identified in the site evaluation process. Causation and likelihood shall be considered in postulating potential hazards. In the short term, the safety of the plant shall not be dependent on the availability of off-site services such as electricity supply and fire fighting services. **The design shall take into account site specific conditions to determine the delay after which off-site services need to be available.**

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Page: 9

Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 100646 -02'00'
The text is identical to the IAEA requirement, no modification.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 102654 -02'00'
Important requirement for SMRs where up to 5-10 reactors can operate at the same site. AC (some SMR types (e.g. NuScale) the reactors are in the same containment building, which creates a much tighter connection between the units, including some interconnections of the safety systems. This special feature could be put into the requirement.			
Number: 3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 102905 -02'00'
Not relevant for SCWR.			
Number: 4	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 102914 -02'00'
Unnecessary modification.			

17C For all the postulated initiating events that threaten the supply of power or the heat sinks, due consideration shall be taken of the capability of the plant to reach and maintain a safe state, without external intervention, for a long period after an event. For this purpose, the period of time during which a safety function is ensured in an event without the need of action by personnel should be maximised.

5.19 Features shall be provided to minimize any interactions between structures containing items important to safety (including power cabling and control cabling) and any other plant structure as a result of external events considered in the design.

5.20 This paragraph was deleted and its content, with a broader scope, has been transferred to the new paragraph 5.15A.

5.21 The design shall include due consideration of extreme external hazards and their consequences. In addition specific equipments qualified to withstand these hazards should be provided (e.g. dedicated AC power, instrumentation...) and the design of the plant shall provide for an adequate margin to protect items important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site, and to avoid cliff edge effects. 5.21A. The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent **significant radioactive release** in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site.

5.22 This paragraph was deleted and its content, with a broader scope, has been transferred to the new paragraph 5.15B.

Criterion 18: Engineering design rules

The engineering design rules for items important to safety at a nuclear power plant shall be specified and shall comply with the relevant national or international codes and standards, with proven engineering practices and with relevant research, with due account taken of their relevance to nuclear power technology.

5.23 Methods to ensure a robust design shall be applied and proven engineering practices shall be adhered to in the design of a nuclear power plant to ensure that the fundamental safety functions are achieved in all operational states and for all accident conditions.

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Page: 10

Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 102924 -02'00'
Acceptable modification (however, it should be located in the chapter considering the power supply issues).			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 102934 -02'00'
Not acceptable. The original term "prevent an early radioactive release or a large radioactive release" refers to 2 different phenomena. With the modification the term "early" is missing.			

Criterion 19: Design basis accidents

A set of accidents that are to be considered in the design shall be derived from postulated initiating events for the purpose of establishing the boundary conditions for the nuclear power plant to withstand, without acceptable limits for radiation protection being exceeded.

5.24 Design basis accidents shall be used to define the design bases, including performance criteria, for safety systems and for other items important to safety that are necessary to control design basis accident conditions, with the objective of returning the plant to a safe state and mitigating the consequences of any accidents.

5.25 The design shall be such that for design basis accident conditions, key plant parameters do not exceed the specified design limits. A primary objective shall be to manage all design basis accidents so that they have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective actions.

5.26 The design basis accidents are preferably analyzed in a conservative manner. This approach involves postulating certain failures in safety systems, specifying design criteria and using conservative assumptions, models and input parameters in the analysis. **The design basis accidents could also be analysed in a best estimate manner, together with adequately analysed and evaluated uncertainties.**

Criterion 20: Design extension conditions

A set of design extension conditions shall be derived on the basis of engineering judgement, operating experience, 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.

The design of a nuclear power plant shall be such that the level 4 of the defence in depth and the associated safety design for prevention and/or mitigation of severe core degradation and of serious fuel failures during fuel handling and storage shall be incorporated, in order that significant radioactive release can be considered as belonging to the residual risk.

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Page: 11

Number: 1	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 100945 -02'00"
Modification not accepted. The detailed evaluation of different deterministic approaches is necessary. The application of BEPU (best estimate plus uncertainty) approach is based on the development of physical and mathematical models of the considered phenomena. For GEN3 (WR), this approach is used in the design process and for DEC calculations, but for the analysis of DEC accidents usually the conservative or the combined method - best estimate computer codes with conservative parameters - is applied, see IAEA SSG 2). However, in case of supercritical water cooled reactors, the deep understanding of accident phenomena are still missing.			
Number: 2	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 1031016 -02'00"
Not accepted. There is no reason for selecting and highlighting level 4.			

5.27 An analysis of design extension conditions for the plant shall be performed. The main technical objective of considering **probabilistic** design extension conditions is to provide assurance that the design of the plant is such as to prevent accident conditions that are not considered as design basis accident conditions, or to mitigate their consequences. This might require additional safety features for design extension conditions, or extension of the capability of safety systems to prevent, or to mitigate the consequence of a severe accident, or to maintain the containment function. These additional safety features for design extension conditions, or this extension of the capability of safety systems, shall ensure the capability of managing accident conditions in which there is a significant amount of **released and** radioactive material in the containment (including radioactive material resulting from severe degradation of the reactor core). The plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that the possibility of plant states arising that could lead to a significant radioactive release is "practically eliminated". The effectiveness of provisions to ensure the functionality of the containment could be analysed on the basis of the best estimate approach.

5.28 The design extension conditions shall be used to define the design specifications for safety features and for the design of all other items important to safety that are necessary for preventing such conditions from arising, or, if they do arise, for controlling them and mitigating their consequences.

5.29 The analysis undertaken shall include identification of the features that are designed for use in, or that are capable¹ of preventing or mitigating, events considered in the design extension conditions. These features:

- (a) Shall be independent, to the extent practicable, of those used in more frequent accidents;
- (b) Shall be capable of performing in the environmental conditions pertaining to these design extension conditions, including design extension conditions in severe accidents, where appropriate²;
- (c) Shall have reliability commensurate with the safety function that they are required to fulfil.

5.30 In particular, the containment and its safety features shall be able to withstand extreme scenarios that include, among other things, melting of the reactor core. These scenarios shall be selected by using engineering judgement and input from probabilistic safety assessments.

¹ For returning the plant to a safe state or for mitigating the consequences of an accident, consideration could be given to the full design capabilities of the plant and to the temporary use of additional systems. (From IAEA SSG 2) Footnote 2)

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Number: 1	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 1010330 -02'00"
Not accepted. The term 'probabilistic' is not right for DEC. (Initial events are postulated)			
Number: 2	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 1010339 -02'00"
Not relevant for SCWR.			
Number: 3	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 1010347 -02'00"
"b)" Shall be capable of withstanding extreme external conditions taken into account in DEC conditions (e.g. effects of external hazards such as earthquake, flooding etc), taken into consideration the possible combination of hazards."			

5.31. The design shall be such that design extension conditions that could lead to significant radioactive releases are practically eliminated. **Where a fast reactor core is not in its most reactive configuration under normal operating conditions, the following design features for prevention and mitigation of severe accidents in postulated design extension conditions shall be considered:**

- (a) **Additional reactor shutdown measures against failure of active reactor shutdown systems;**
- (b) **Mitigation provision to avoid recritically loading large mechanical energy release during a core degradation progression;**
- (c) **Means for deep head removal of a degraded core, and**
- (d) **Containment capability of enduring thermal and mechanical loads under severe accident conditions.**

Combinations of events and failures

5.32. Where the results of engineering judgement, operating experience, deterministic safety assessments and probabilistic safety assessments indicate that combinations of events could lead to anticipated operational occurrences or to accident conditions, such combinations of events shall be considered to be design basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence. Certain events might be consequences of other events, such as a flood following an earthquake. Such consequential effects shall be considered to be part of the original postulated initiating event.

Criterion 21: Physical separation and independence of safety systems

Interference between safety systems or between redundant elements of a system shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication (data transfer), as appropriate

5.33. Safety system equipment (including cables and raceways) shall be readily identifiable in the plant for each redundant element of a safety system.

Criterion 22: Safety classification

All items important to safety shall be identified and shall be classified on the basis of their function and their safety significance.

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Criterion 31: Aging management

The design life of items important to safety at a nuclear power plant shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, such as embrittlement and wear-out, and of the potential for age related degradation, due to high operating temperature, the sodium coolant, and the fast neutron irradiation, to ensure the capability of items important to safety to perform their necessary safety functions throughout their design life.

5.51. The design for a nuclear power plant shall take due account of ageing and wear-out effects in all operational states for which a component is created, including testing, maintenance, maintenance outages, plant states during a postulated initiating event and plant states following a postulated initiating event.

5.52. Provision shall be made for monitoring, testing, sampling and inspection to assess ageing mechanisms predicted at the design stage and to help identify unanticipated behaviour of the plant or degradation that might occur in service.

5.3 Human Factors

Criterion 32: Design for optimal operator performance

Systematic consideration of human factors, including the human-machine interface, shall be included at an early stage in the design process for a nuclear power plant and shall be continued throughout the entire design process.

5.53. The design for a nuclear power plant shall specify the minimum number of operating personnel required to perform all the simultaneous operations necessary to bring the plant into a safe state.

5.54. Operating personnel who have gained operating experience in similar plants shall as far as is practicable be actively involved in the design process conducted by the design organization in order to ensure that consideration is given as early as possible in the process to the future operation and maintenance of equipment.

5.55. The design shall support operating personnel in the fulfilment of their responsibilities and in the performance of their tasks, and shall limit the likelihood and the effects of operating errors on safety. The design process shall give due consideration to plant layout and equipment layout, and to procedures, including procedures for maintenance and inspection, to facilitate interaction between the operating personnel and the plant, in all plant states.

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5.4 Other Design Considerations

Criterion 33: Safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant

Each of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions.

63. To further enhance safety, means allowing interconnections between units of a multiple unit nuclear power plant shall be considered in the design.

Criterion 34: Systems containing fissile material or radioactive material

All systems in a nuclear power plant that could contain fissile material or radioactive material shall be so designed as to prevent the occurrence of events that could lead to an uncontrolled radioactive release to the environment; to prevent accidental criticality and overheating; to ensure that radioactive releases are kept below authorized limits on discharges in normal operation and below acceptable limits in accident conditions, and are kept as low as reasonably achievable; and to facilitate mitigation of radiological consequences of accidents.

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

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

Criterion 36: Escape routes from the plant

A nuclear power plant shall be provided with a sufficient number of escape routes, clearly and durably marked, with reliable emergency lighting, ventilation and other services essential to the safe use of these escape routes.

5.64. Escape routes from the nuclear power plant shall meet the relevant national and international requirements for radiation zoning and fire protection, and the relevant national requirements for industrial safety and plant security.

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:11:17 +02:00
The criterion shall be re-evaluated considering the SMR safety approaches. The term 'multiple-unit' are applied by IAEA documents for SMRs instead of multiple units. However, the specific features of such in-dome-unit type SMRs (heavy new type of interconnections between the reactors compared to the traditional LWRs (e.g. common containment building or decay heat removal water pools).			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:11:35 +02:00
See Crite. 33. In case of SMRs, the term 'multiple-unit' is not applicable.			
Number: 3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:11:44 +02:00
Should be replaced with 'district or process heating'			

5.65. At least one escape route shall be available from workplaces and other occupied areas following an internal event or an external event or following combinations of events considered in the design.

Criterion 37: Communication systems at the plant

Effective means of communication shall be provided throughout the nuclear power plant to facilitate safe operation in all modes of normal operation and to be available for use following all postulated initiating events and in accident conditions. [Also accounting for the interface of safety with security.

5.66. Suitable alarm systems and means of communication shall be provided so that all persons present at the nuclear power plant and on the site can be given warnings and instructions, in operational states and in accident conditions.

5.67. Suitable and diverse means of communication necessary for safety within the nuclear power plant and in the immediate vicinity, and for communication with relevant off-site agencies, shall be provided.

Criterion 38: Control of access to the plant

The nuclear power plant shall be isolated from its surroundings with a suitable layout of the various structural elements so that access to it can be controlled.

5.68. Provision shall be made in the design of the buildings and the layout of the site for the control of access to the nuclear power plant by operating personnel and/or for equipment, including emergency response personnel and vehicles, with particular consideration given to guarding against the unauthorized entry of persons and goods to the plant. [detecting, assessing, and delaying the entry.

Criterion 39: Prevention of unauthorized access to or interference with items important to safety

Unauthorized access to, or interference with, items important to safety, including computer hardware and software, shall be prevented.

Criterion 40: Prevention of harmful interactions of systems important to safety

The potential for harmful interactions of systems important to safety at the nuclear power plant that might be required to operate simultaneously shall be evaluated, and effects of any harmful interactions shall be prevented.

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:11:54 +02:00
Not accepted. Modification is not necessary.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:02 +02:00
Not accepted. The requirement is not exactly dealing with the physical protection of the NPP.			

6. DESIGN OF SPECIFIC PLANT SYSTEMS

6.1 Overall Plant System

Criterion 42b: Plant system performance of a sodium-cooled fast reactor

The overall plant system shall be designed considering the specific characteristics of a sodium-cooled fast reactor as described below.

- (a) The reactor core is not in its most reactive configuration under normal operating conditions. This could lead to a positive reactivity insertion due to an unfavourable change in reactor core geometry.
- (b) The sodium void reactivity may be positive in the central region of the reactor core. This could lead to a positive reactivity insertion due to sodium boiling or gas entrainment.
- (c) The high boiling temperature of sodium at standard atmospheric pressure enables the reactor coolant system to operate at low pressure with a large margin to boiling.
- (d) The high thermal conductivity and heat transfer coefficient of sodium, the large temperature gradient in the reactor core, and the decrease of sodium density with increasing temperature enable decay heat removal by natural circulation of the coolant.
- (e) Sodium is chemically active and opaque, and it is solid below 98 °C.
- (f) Some mist and vapour of sodium can deposit on the components.

6.2 Reactor Core and Associated Features

Criterion 43: Performance of fuel elements and assemblies

Fuel elements and assemblies for the nuclear power plant shall be designed to maintain their structural integrity, and to withstand satisfactorily the anticipated radiation levels and other conditions in the reactor core, **including fast neutron fluence**, in combination with all the processes of deterioration that could occur in operational states.

6.1. The processes of deterioration to be considered shall include those arising from: differential expansion and deformation, internal pressure increase due to temperature, fission products and the build-up of helium, irradiation of fuel and other materials in the fuel assembly, variations in temperature resulting from variations in power demand, chemical effects, static and dynamic loading, including flow induced vibrations and mechanical vibrations, and variations in temperature in relation

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:11 +02:00
Not relevant for SCWR. However, the insertion of similar criterion concerning the specific features of SCWRs can be considered.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:24 +02:00
Not relevant for SCWR.			

to heat transfer that could result from distortions or chemical effects. Allowance shall be made for uncertainties in data, in calculations and in manufacture.

6.2. Fuel design limits shall include limits on the permissible leakage of fission products from the fuel in anticipated operational occurrences so that the fuel remains available for continued use.

6.3. Fuel elements and fuel assemblies shall be capable of withstanding the loads and stresses associated with fuel handling.

Criterion 44: Structural capability of the reactor core

The fuel elements and fuel assemblies and their supporting structures for the nuclear power plant shall be designed so that, in operational states and in accident conditions **due to both internal and external events** other than severe accidents, a geometry that allows for adequate cooling is maintained, **core geometry is preserved to prevent excessive reactivity changes**, and the insertion of control devices is not impeded.

For the design extension conditions, provisions shall be included to avoid re-criticality resulting in potentially large mechanical energy release during a core disruptive accident.

3Ms: The supporting structures shall be designed with due account taken of the creep properties, thermal stripping, fast neutron induced changes, other ageing effects, and the material compatibility with sodium and its compounds.

6.3ter. The fuel assemblies and associated core support structure shall be designed to prevent mis-loading of fuel assemblies and any coolant channel blockages.

3quater: The assemblies and associated core support structure shall be designed so that the core geometry can be preserved to prevent excessive reactivity effects.

Criterion 45: Control of the reactor core

Distributions of neutron flux that can arise in any state of the reactor core in the nuclear power plant, including states arising after shutdown and during or after refuelling, and states arising from anticipated operational occurrences and from accident conditions not involving degradation of the reactor core, shall be inherently stable. The demands made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized.

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:33 +02:00
Not necessary.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:42 +02:00
Not relevant for SCWR.			
Number: 3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:12:50 +02:00
Not relevant for SCWR.			

6.4. Adequate means of detecting the neutron flux in the reactor core and its change shall be provided for the purpose of ensuring that there are no regions of the core in which the design limits could be exceeded.

6.5. In the design of reactivity control devices, due account shall be taken of wear-out and of the effects of irradiation, such as burn-up, changes in physical properties and dimensions, and production of gas during normal operation, anticipated operational occurrences and accident conditions.

6.6. The maximum degree of positive reactivity and its rate of increase by insertion in operational states and accident conditions not involving degradation of the reactor core shall be limited or compensated for to prevent any resultant failure of the boundary of the reactor coolant systems, to maintain the capability for cooling and to prevent any significant degradation of the reactor core.

Notes: To avoid significant mechanical energy release during a core disruptive accident, the reactor core shall be designed to have favourable neutronic, thermal, and structural characteristics, considering all reactivity feedbacks, including sodium void worth, to mitigate the consequences of such design extension conditions.

Criterion 46: Reactor shutdown

Means shall be provided to ensure to shut down the reactor of the nuclear power plant in operational states and in accident conditions, and that the shutdown condition can be maintained even for the most reactive conditions of the reactor core.

6.7. The effectiveness, speed of action and shutdown margin of the means of shutdown of the reactor shall be such that the specified design limits for fuel are not exceeded.

6.8. In judging the adequacy of the means of shutdown of the reactor, consideration shall be given to failures arising anywhere in the plant that could render part of the means of shutdown inoperative (such as failure of a control rod to insert) or that could result in a common cause failure.

6.9. The means for shutting down the reactor shall consist of at least two diverse and independent systems. **For design extension conditions, inherent power reduction with complementary shutdown method and/or passive shutdown capabilities shall be provided to prevent severe core degradation and to avoid re-criticality in the long run.**

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:15:58 +02:00
Not relevant for SCWR.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:13:07 +02:00
The modification requires re-evaluation for DDC conditions, usually no additional shutdown systems are required in case of GEN IV reactors. Apart from ATW, diverse shutdown systems are capable of stopping the chain reaction.			

6.10. At least one of the two different shutdown systems shall be capable, on its own, of maintaining the reactor subcritical by an adequate margin and with high reliability, even for the most reactive conditions of the reactor core.

6.11. The means of shutdown shall be adequate to prevent any fireseenable increase in reactivity leading to unintentional criticality during the shutdown or during refuelling operations or other routine or non-routine operations in the shutdown state.

6.12. Instrumentation shall be provided and tests shall be specified for ensuring that the means of shutdown are always in the state stipulated for a given plant state.

4.3 Reactor Coolant Systems

Criterion 47: Design of reactor coolant systems

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

6.13. Pipework connected to the **reactor coolant** boundary for the nuclear power plant shall be equipped with adequate isolation devices to limit any loss of radioactive fluid (primary coolant) and to prevent the loss of coolant through interfacing systems so that cooling of the reactor core can be maintained.

6.14. The design of the reactor coolant boundary shall be such that flaws are very unlikely to be initiated, and any flaws that are initiated and propagate result in leaks long before the flaws would grow to an unstable size, thereby permitting the timely detection of coolant leakage.

Notes: inert gas shall be used as a cover gas in sodium-filled components to prevent chemical reaction at the free surface of sodium, and the boundary of the cover gas shall be designed to be leak tight with isolation valves, except when the lines are equipped with pressure relief valves to protect the reactor vessel from excessive pressure load (over or under pressure). The reactor coolant boundary shall be designed as a barrier against radioactive materials release and be closed by the reactor cover gas boundary.

Notes: Provisions shall be made to detect sodium leaks and to mitigate the consequence of sodium chemical reaction in case of postulated sodium leaks from the reactor coolant systems. The

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Number: 1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:13:17 +02:00
Not accepted. The original term 'pressure boundary of reactor coolant system' is more appropriate.			
Number: 2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:13:27 +02:00
Not relevant for SCWR.			
Number: 3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:15:11 +02:00
Not relevant for SCWR.			

fundamental safety functions shall be maintained under severe sodium leak events considered in the design extension conditions.

6.15. The design of the reactor coolant systems shall be such as to ensure that plant states in which components of the reactor coolant boundary could exhibit embrittlement are avoided.

1501s. *The components of the reactor coolant systems shall be designed with due account taken of creep properties, thermal stripping, fast neutron fluence, and other ageing effects, as well as its compatibility with sodium, and with thermal stress and dynamic load on thin-walled structures used under low pressure and high temperature conditions.*

1501r. *The design shall consider the potential for flow and thermal disturbances, such as flow induced vibrations or thermal stripping, and shall reduce or eliminate such effects to maintain the structural integrity of the components of the reactor coolant systems.*

6.16. The design of the components contained inside the reactor coolant boundary, such as pump impellers and valve parts, shall be such as to minimize the likelihood of failure and consequential damage to other components of the reactor coolant system that are important to safety, in all operational states and in design basis accident conditions, with due allowance made for deterioration that might occur in service.

1601s. *Components, which constitute the reactor coolant boundary, shall be designed to maintain the boundary function and to maintain a sufficient sodium inventory in the reactor coolant system in case of anticipated transients without scram.*

1601r. *Chemical reactions between sodium and water/steam or other working fluids shall be considered in the design of the secondary coolant system. Provisions to prevent and/or mitigate such chemical reactions shall be incorporated in the design:*

(a) *Provisions shall be made to detect leaks of working fluid, to control any leak propagation, and to automatically mitigate any leak accident to prevent further damages, such that isolation and relief valves in working fluid system, when a heat exchange system between the sodium and the working fluid is used.*

(b) *A pressure relief system shall be employed in the secondary coolant system to protect the secondary coolant system from consequences resulting from sodium interactions with water/steam in the steam generator or with other working fluids in the heat exchanger.*

(c) *The fundamental safety functions shall be maintained under postulated design extension conditions for severe chemical reactions between the sodium and the working fluid.*

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Number:1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:15:42 +02:00
Not relevant for SCWR.			
Number:2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:15:50 +02:00
Not relevant for SCWR.			
Number:3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:00 +02:00
Not relevant for SCWR.			
Number:4	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:08 +02:00
Not relevant for SCWR.			

1701r. *Lines that penetrate the reactor coolant and cover gas boundaries shall be designed in order to prevent air and water ingresses.*

1701q. *The design of the reactor coolant system shall be such as to consider implementation of in service inspection of structures and components, important to safety contained inside the reactor coolant boundary with dedicated equipments that overcome the sodium opaqueness.*

Criterion 48: Excessive pressure load/protection of the reactor coolant boundary

Provision shall be made to ensure that the operation of pressure relief devices will protect the reactor coolant boundary against excessive pressure load.

1801r. *Level of reactor coolant*

Provision shall be made for controlling the level of the reactor coolant to ensure that specified design limits are not exceeded in operational states and that the cooling of fuel is maintained in accident conditions, with taking due account of volumetric changes to ensure that the core remains covered.

Guard vessels and guard pipes shall be designed so as to maintain the sodium surface of the reactor coolant system at a level necessary for decay heat removal in the case of a sodium leak accident in the reactor coolant system. Due considerations shall be taken of a dependent failure and a common cause failure between the reactor vessel and the guard vessel, as well as between main coolant pipes and guard pipes. Provisions shall be made to reduce the amount of sodium that leaks from the reactor coolant system in case of a failure of the reactor coolant boundary.

Criterion 50: Cleanup of reactor coolant

Adequate facilities shall be provided for the removal of radioactive and chemical substances from the reactor coolant, including activated corrosion products and fission products deriving from the fuel, and non-radioactive substances.

6.17. The capabilities of the necessary plant cleanup systems shall be based on the specified design limit on permissible leakage of the fuel, with a conservative margin to ensure that the plant can be operated with a level of circuit activity that is as low as reasonably practicable, and to ensure that the requirements are met for radioactive releases to be as low as reasonably achievable and below the authorized limits on discharges.

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Number:1	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:17 +02:00
Not relevant for SCWR.			
Number:2	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:26 +02:00
Not relevant for SCWR.			
Number:3	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:36 +02:00
Not relevant for SCWR.			
Number:4	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:43 +02:00
Modifications are not relevant for SCWR.			
Number:5	Author: BME NTI	Subject: Highlight	Date: 15.07.2021 10:16:51 +02:00
Modifications are not relevant for SCWR.			

1701s: Concentration of impurities in the sodium shall be controlled within a limit value in order to prevent excessive corrosion, coolant channel blockage, or other effects resulting from dissolved or particulate impurities in the coolant. A cover gas cleanup system shall be included to ensure purity of the cover gas and to recover any reaction products or contamination, radioactive and chemical.

Criterion 51: Decay heat removal system

Means shall be provided for the removal of decay heat from the reactor core to an ultimate heat sink after shutdown of the nuclear power plant in operational states and in accident conditions.

18. The decay heat removal systems for cooling of the reactor core shall be such as to ensure that

- (a) The design limits for fuel, the reactor coolant boundary and structures important to safety are not exceeded in the shutdown state of the nuclear power plant,
- (b) The cooling of the fuel is restored and maintained under accident conditions even if the integrity of the reactor coolant boundary is not maintained, and
- (c) The function to transfer decay heat from items important to safety at the nuclear power plant to an ultimate heat sink shall be carried out with very high levels of reliability for all plant states.

19. The decay heat removal system shall be designed as follows:

- (a) To provide diversity to the extent practicable and redundancy for reducing common cause failures, including external events,
- (b) To prevent freezing of the sodium coolant to avoid blockage of coolant circulation, and
- (c) To provide detection and mitigation measures against postulated decay heat fluid leaks.

1901s: In design extension conditions, means for decay heat transfer shall be provided, in addition to a decay heat removal system for anticipated operational occurrence and design-basis accidents, with the conditions listed below. Means shall be provided for the capability of core cooling under postulated plant conditions with core degradation.

- (a) The cooling of the reactor core is possible even under extreme external hazards and their consequences, such as long-term loss of all AC power supplies,
- (b) Passive mechanisms are used to the extent practicable, and
- (c) Decay heat removal system has diversity to the extent practicable.

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Included: Criterion 52: Emergency Cooling of the reactor core
(Included in Criterion 51)

Criterion 53: Heat transfer to an ultimate heat sink

The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states.

6.19A. Systems for transferring heat shall have adequate reliability for the plant states in which they have to fulfil the heat transfer function. This may require the use of a different ultimate heat sink or different access to the ultimate heat sink.

6.19B. The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site.

6.4 Containment Structure and Containment System

Criterion 54: Containment system for the reactor

A containment system shall be provided to ensure or to contribute to the fulfilment of the following safety functions at the nuclear power plant: (i) confinement of radioactive substances in operational states and in accident conditions, (ii) protection of the reactor against natural external events and human induced events and (iii) radiation shielding in operational states and **accident conditions.**

Criterion 55: Control of radioactive releases from the containment

The design of the containment shall be such as to ensure that any radioactive material release from the nuclear power plant to the environment is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions.

6.20. The containment structure and the systems and components affecting the leaktightness of the containment system shall be designed and constructed so that the leak rate can be tested after all penetrations through the containment have been installed and, if necessary during the operating lifetime of the plant. **The design basis for the containment shall consider pressure increase and thermal loads due to sodium fire and severe accident.**

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Inclusion of additional requirement should be considered. 6.19C. The design of containment shall ensure the proper arrangement of systems, structures and component, making possible to perform their safety functions. (Especially for passive safety systems where the proper elevations are key factors for natural convection development).			
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6.21 The number of penetrations through the containment shall be kept to a practical minimum and all penetrations shall meet the same design requirements as the containment structure itself. The penetrations shall be protected against reaction forces caused by pipe movement or accidental loads such as those due to missiles caused by external or internal events.

Criterion 56: Isolation of the containment

Each line that penetrates the containment at a nuclear power plant as part of the reactor coolant boundary ~~but the reactor cover gas boundary~~ or that is connected directly to the containment atmosphere shall be automatically and reliably sealable in the event of an accident in which the leak-tightness of the containment is essential to preventing radioactive releases to the environment that exceed acceptable limits.

6.22 Lines that penetrate the containment, as part of the reactor coolant boundary and ~~the reactor cover gas boundary~~, and lines that are connected directly to the containment atmosphere shall be fitted with at least two adequate containment isolation valves arranged in series³, and shall be provided with suitable leak detection systems *for preventing the containment bypass of radioactive materials*. Containment isolation valves shall be located as close to the containment as is practicable, and each valve shall be capable of reliable and independent actuation and of being periodically tested.

6.23 Each line that penetrates the containment and is neither part of the reactor coolant boundary nor ~~the reactor cover gas boundary and is not~~ connected directly to the containment atmosphere shall have at least one adequate containment isolation valve. The containment isolation valves shall be located outside the containment and as close to the containment as is practicable.

6.24 Exceptions to the requirements for containment isolation, stated in paragraphs 6.22, 6.23, shall be permissible for specific classes of lines such as instrumentation lines, or in cases in which application of the methods of containment isolation, specified in paragraphs 6.22, 6.23, would reduce the reliability of a safety system that includes a penetration of the containment.

³ In most cases, one containment isolation valve or check valve is outside the containment and the other is inside the containment. Other arrangements might be acceptable, however, depending on the design. (From IAEA SSR 21 Footnote 11)

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Criterion 57: Access to the containment

Access by operating personnel to the containment at a nuclear power plant shall be through airlocks equipped with doors that are interlocked to ensure that at least one of the doors is closed during reactor power operation and in accident conditions.

6.25 Where provision is made for entry of operating personnel for surveillance purposes, provision for ensuring protection and safety for operating personnel shall be specified in the design. Where equipment airlocks are provided, provision for ensuring protection and safety for operating personnel shall be specified in the design.

6.26 Containment openings for the movement of equipment or material through the containment shall be designed to be closed quickly and reliably in the event that isolation of the containment is required.

Criterion 58: Control of containment conditions

Provision shall be made to control the pressure and temperature in the containment at a nuclear power plant and to control any build-up of fission products or other gaseous, liquid or solid substances that might be released inside the containment and that could affect the operation of systems important to safety.

6.27 *If present*, the design shall provide for sufficient flow routes between separate compartments inside the containment. The cross-sections of openings between compartments shall be of such dimensions as to ensure that the pressure differentials occurring during pressure equalization in accident conditions do not result in unacceptable damage to the pressure bearing structure or to systems that are important in mitigating the effects of accident conditions.

6.28 The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain it at acceptably low levels. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.

6.28A Design provision shall be made to prevent the loss of the structural integrity, e.g. due to temperature and/or pressure increases, of the containment in all plant states. The use of this provision shall not lead to a significant radioactive release.

38B omitted

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6.29 Design features to control fission products **[to be removed]**, hydrogen and other substances that might be released into the containment shall be provided as necessary.

(a) To reduce the amounts of fission products that could be released to the environment in accident conditions;

(b) **[to be removed]** prevent or mitigate sodium combustion, sodium-concrete reaction, and debris-concrete interaction and to control the concentration of fission products in the containment atmosphere in accident conditions so as to prevent thermal, deflagration or detonation loads that could challenge the integrity of the containment.

[3.0. Omitted]

6.5 Instrumentation and Control Systems

Criterion 59: Provision of instrumentation

Instrumentation shall be provided for determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant, for obtaining essential information on the plant that is necessary for its safe and reliable operation, for determining the status of the plant in accident conditions, and for making decisions for the purposes of accident management.

6.31. Instrumentation and recording equipment shall be provided to ensure that essential information is available for monitoring the status of essential equipment and the course of accidents, for predicting the locations of releases and amounts of radioactive material that could be released from the locations that are so intended in the design, and for post-accident analysis.

[3.0. This instrumentation, which penetrates or is connected to the boundary of the reactor coolant systems, shall be designed so that sodium leaks and combustions caused by their failure are prevented and/or mitigated]

Criterion 60: Control systems

Appropriate and reliable control systems shall be provided at the nuclear power plant to maintain and limit the relevant process variables within the specified operational ranges.

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6.34 Design techniques such as testability, including a self-checking capability where necessary, fail-safe characteristics, functional diversity, and diversity in component design and in concepts of operation shall be used to the extent practicable to prevent the loss of a safety function.

6.35 Safety systems shall be designed to permit periodic testing of their functionality when the plant is in operation, including the possibility of testing channels independently for the detection of failures and losses of redundancy. The design shall permit all aspects of functionality testing for the sensor, the input signal, the final actuator and the display.

6.36 When a safety system, or part of a safety system, has to be taken out of service for testing, adequate provision shall be made for the clear indication of any protection system bypasses that are necessary for the duration of the testing or maintenance activities.

Criterion 63: Use of computer based equipment in systems important to safety

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

6.37. For computer based equipment in safety systems or safety **[to be removed]** systems:

(a) A high quality of, and test practices for, hardware and software shall be used, in accordance with the importance of the system to safety.

(b) The entire development process, including control, testing and commissioning of design changes, shall be systematically documented and shall be reviewable.

(c) An assessment of the equipment shall be undertaken by experts, who are independent of the design team and the supplier team to provide assurance of its high reliability, and who are qualified with respect to the environment that the equipment may be subjected to during normal operation, anticipated operational occurrences and accident conditions;

(d) Where safety functions are essential for achieving and maintaining safe conditions, and the necessary high reliability of the equipment cannot be demonstrated with a high level of confidence, diverse means of ensuring the fulfilment of the safety functions shall be provided;

(e) Common cause failures deriving from software shall be taken into consideration;

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Not accepted, "reliable" is the proper term.			

types that are consistent with all the requirements of the safety systems to be supplied with power, and their functional capability shall be testable.

6.44A omitted

6.44B omitted

6.44C. The alternate power source shall be independent of and physically separated from the emergency power supply. The connection time of the alternate power source shall be consistent with the depletion time of the battery.

6.44D. Continuity of power for the monitoring of the key plant parameters and for the completion of short term actions necessary for safety shall be maintained in the event of loss of the AC (alternating current) power sources.

6.45. The design basis for any diesel engine or other prime mover⁴ that provides an emergency power supply to items important to safety shall include:

- (a) the capability of the associated fuel oil storage and supply systems to satisfy the demand within the specified time period;
- (b) the capability of the prime mover to start and to function successfully under all specified conditions and at the required time;
- (c) auxiliary systems of the prime mover such as coolant systems.

6.45A. The design shall also include features to enable the safe use of non-permanent equipment to restore the necessary electrical power supply.²⁵

6.7 Supporting Systems and Auxiliary Systems

Criterion 69: Performance of supporting systems and auxiliary systems.

The design of supporting systems and auxiliary systems shall be such as to ensure that the performance of these systems is consistent with the safety significance of the system or

⁴ A prime mover is a component (such as a motor, solenoid operator or pneumatic operator) that converts energy into action when commanded by an actuation device. (From IAEA SSR 2/1 Prime-mover 22)

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component that they serve at the nuclear power plant with due consideration of the principle of independence of levels of defence in depth.

Criterion 70: Heat transport systems

Auxiliary systems shall be provided as appropriate to remove heat from systems and components at the nuclear power plant that are required to function in operational states and in accident conditions.

6.46. The design of heat transport systems shall be such as to ensure that non-essential parts of the systems can be isolated.

Criterion 71: Process sampling systems and post-accident sampling systems

Process sampling systems and post-accident sampling systems shall be provided for determining, in a timely manner, the concentration of specified radionuclides in fluid process systems, and in gas and liquid samples taken from systems or from the environment, in all operational states and in accident conditions at the nuclear power plant.

6.47. Appropriate means shall be provided at the nuclear power plant for the monitoring of activity in fluid systems that have the potential for significant contamination, and for the collection of process samples.

Criterion 72: Compressed air and gas systems

The design basis for any compressed air or gas system that serves an item important to safety at the nuclear power plant shall specify the quality, flow rate and cleanliness of the air or gas to be provided.

Criterion 73: Air conditioning systems and ventilation systems

Systems for air conditioning, air heating, air cooling and ventilation shall be provided as appropriate in auxiliary rooms or other areas at the nuclear power plant to maintain the required environmental conditions for systems and components important to safety in all plant states.

6.48. Systems shall be provided for the ventilation of buildings at the nuclear power plant with appropriate capability for the cleaning of air **and gas**.

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(a) To prevent unacceptable dispersion of airborne radioactive substances within the plant,
 (b) To reduce the concentration of airborne radioactive substances to levels compatible with the need for access by personnel to the area,
 (c) To keep the levels of airborne radioactive substances in the plant below authorized limits and as low as reasonably achievable,
 (d) To ventilate rooms containing inert gases or noxious gases without impairing the capability to control radioactive effluents,
 (e) To control gaseous radioactive releases to the environment below the authorized limits on discharges and to keep them as low as reasonably achievable.

6.49 Areas of higher contamination at the plant shall be maintained at a negative pressure differential (partial vacuum) with respect to areas of lower contamination and other accessible areas.

Criterion 74: Fire protection systems

Fire protection systems, including fire detection systems and fire extinguishing systems, fire containment barriers and smoke control systems, shall be provided throughout the nuclear power plant, with due account taken of the results of the fire hazard analysis. **Water systems used for firefighting shall not be located in the same compartment as sodium circuits and tanks**

6.50 The fire protection systems installed at the nuclear power plant shall be capable of dealing safely with fire events of the various types **including sodium fires**, that are postulated.

6.51 Fire extinguishing systems shall be capable of automatic actuation where appropriate. Fire extinguishing systems shall be designed and located to ensure that their rupture or spurious or inadvertent operation would not significantly impair the capability of items important to safety.

6.52 Fire detection systems shall be designed to provide operating personnel promptly with information on the location and spread of any fires that start.

6.53 Fire detection systems and fire extinguishing systems that are necessary to protect against a possible fire following a postulated initiating event shall be appropriately qualified to resist the effects of the postulated initiating event.

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

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6.50bis. Adequate means of protecting the human body from sodium compounds generated by sodium fires shall be provided.

6.51ter. Compartments with sodium components shall be protected from the impacts induced by sodium fire to prevent the fire spread and from water ingress to prevent sodium-water chemical reactions, especially from water used in case of fire fighting in an adjacent compartment.

Criterion 75: Lighting systems

Adequate lighting shall be provided in all operational areas of the nuclear power plant in operational states and in accident conditions.

Criterion 76: Overhead lifting equipment

Overhead lifting equipment shall be provided for lifting and lowering items important to safety at the nuclear power plant, and for lifting and lowering other items in the proximity of items important to safety.

6.55 The overhead lifting equipment shall be designed so that

- (a) Measures are taken to prevent the lifting of excessive loads;
- (b) Conservative design measures are applied to prevent any unintentional dropping of loads that could affect items important to safety;
- (c) The plant layout permits safe movement of the overhead lifting equipment and of items being transported;
- (d) Such equipment can be used only in specified plant states (by means of safety interlocks on the crane);
- (e) Such equipment for use in areas where items important to safety are located is seismically qualified.

Criterion 76bis: Sodium Heating Systems

Heating systems shall be provided for components as necessary to prevent loss of fundamental safety functions by sodium freezing. These heating systems and their controls shall be appropriately

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designed to assure that the temperature distribution and rate of change of temperature are maintained within the limits.

Criterion 76a: Sodium Chemical Reaction Prevention and Mitigation

Due to chemical risk of sodium which burns in air and reacts with water, impact of such chemical reactions to items important to safety must be prevented. Water systems shall be avoided in compartment containing or likely to contain sodium unless justified with the demonstration that risk of the sodium-water reactions are properly managed.

6.8 Other Power Conversion Systems

Criterion 77: Power conversion systems, including potential steam supply systems, feedwater systems and turbine generators

The design of the power conversion systems, including potential steam supply systems, feedwater systems and turbine generators, for the nuclear power plant shall be such as to ensure that the appropriate design limits of the boundary of the reactor coolant systems are not exceeded in operational states and in accident conditions.

6.56 The design of the power conversion systems shall provide for appropriately rated and qualified working fluid isolation valves capable of closing under the specified conditions in operational states and in accident conditions.

6.57 The working fluid supply system shall be of sufficient capacity and shall be designed to prevent anticipated operational occurrences from escalating to accident conditions.

6.58 The turbine generators shall be provided with appropriate protection such as overspeed protection and vibration protection, and measures shall be taken to minimize the possible effects of turbine generated missiles on items important to safety.

6.9 Treatment of Radioactive Effluents and Radioactive Waste

Criterion 78: Systems for treatment and control of waste

Systems shall be provided for treating solid radioactive waste and liquid radioactive waste at the nuclear power plant to keep the amounts and concentrations of radioactive releases below the

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Number: 4	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 10:21:04 +02:00
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Number: 5	Author: BME/NTI	Subject: Highlight	Date: 15.07.2021 10:21:13 +02:00
The insertion of a new requirement concerning the radioactive contamination of steam supply system could be considered. IAEA. The possible radioactive contamination of the turbine generators shall be taken into consideration during the design of the system and for the planning of radiation protection of plant personnel.			

authorized limits on discharges and as low as reasonably achievable in normal operation and below acceptable limits in accident conditions.

6.59 Systems and facilities shall be provided for the management and storage of radioactive waste on the nuclear power plant site for a period of time consistent with the availability of the relevant disposal option.

6.60 The design of the plant shall incorporate appropriate features to facilitate the movement, transport and handling of radioactive waste. Consideration shall be given to the provision of access to facilities and to capabilities for lifting and for packaging.

Criterion 79: Systems for treatment and control of effluents

Systems shall be provided at the nuclear power plant for treating liquid and gaseous radioactive effluents to keep their amounts below the authorized limits on discharges and as low as reasonably achievable in normal operation and below acceptable limits in accident conditions.

6.61 Liquid and gaseous radioactive effluents shall be treated at the plant so that exposure of members of the public due to discharges to the environment is as low as reasonably achievable.

6.62 The design of the plant shall incorporate suitable means to keep liquid radioactive effluents to the environment as low as reasonably achievable and to ensure that radioactive releases remain below the authorized limits on discharges.

6.63 The cleanup equipment for the gaseous radioactive substances shall provide the necessary retention factor to keep radioactive releases below the authorized limits on discharges. Filter systems shall be designed so that their efficiency can be tested, their performance and function can be regularly monitored over their service life, and filter cartridges can be replaced while maintaining the throughput of air.

6.10 Fuel Handling and Storage Systems

Criterion 80: Fuel handling and storage systems

Fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage including internal and external events.

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6.64 The design of the plant shall incorporate appropriate features to facilitate the lifting, movement and handling of fresh fuel and spent fuel.

6.65 The design of the plant shall be such as to prevent any significant damage to items important to safety during the transfer of fuel or casks, or in the event of fuel or casks being dropped.

6.66 The fuel handling and storage systems for irradiated and non-irradiated fuel shall be designed

- (a) To prevent criticality by a specified margin, by physical means or by means of physical processes, and preferably by the use of geometrically safe configurations, even under conditions of optimum moderation;
- (b) To permit inspection of the fuel;
- (c) To permit maintenance, periodic inspection and testing of components important to safety;
- (d) To prevent damage to the fuel;
- (e) To prevent the dropping of fuel in transit *and the interruption of the transit*;
- (f) To provide for the identification of individual fuel assemblies;
- (g) To prevent mis-loading;**
- (h) To provide proper means for meeting the relevant requirements for radiation protection;
- (i) To ensure that adequate operating procedures and a system of accounting for, and control of, nuclear fuel can be implemented to prevent any loss of, or loss of control over, nuclear fuel.

6.67 In addition, the fuel handling and storage systems for irradiated fuel **and in-core actinide bearing fuel** shall be designed

- (a) To permit adequate removal of heat from the fuel *and monitoring its status in operational states and in accident conditions, including during long-term loss of all AC power supplies*;
- (b) To prevent the dropping of spent fuel in transit *and the interruption of the transit*;
- (c) To avoid causing unacceptable handling stresses on fuel elements or fuel assemblies;
- (d) To prevent the potential of damaging the fuel by dropping of heavy objects, such as spent fuel casks, cranes or other objects *onto the fuel*;
- (e) To permit safe keeping of suspect or damaged fuel elements or fuel assemblies;
- (f) To control levels of soluble absorber if this is used for criticality safety.

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- (g) To facilitate maintenance and future decommissioning of fuel handling and storage facilities;
- (h) To facilitate decontamination of fuel handling and storage areas and equipment when necessary;
- (i) To accommodate, with adequate margins, all the fuel removed from the reactor in accordance with the strategy for core management that is foreseen *and including the entire inventory of fuel in the reactor core*;
- (j) To facilitate the removal of fuel from storage and its preparation for off-site transport.

6.68 For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to **significant radioactive release is "practically eliminated" and so as to avoid high radiation fields on the site. The design of the plant:**

- (a) Shall provide the necessary fuel cooling capabilities;
- (b) Shall provide features to prevent the uncovering of fuel assemblies in the event of a leak or a pipe break;
- (c) Shall provide a capability to restore the water inventory.

The design shall also include features to enable the safe use of non-permanent equipment to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation.

6.68A For reactors using a water pool system for fuel storage, the design of the plant shall include the following

- (a) Means for monitoring and controlling the water temperature for operational states and for accident states that are of relevance for the spent fuel pool;
- (b) Means for monitoring and controlling the water level for operational states and for accident conditions that are of relevance for the spent fuel pool;
- (c) Means for monitoring and controlling the activity in water and in air for operational states and means for monitoring the activity in water and in air for accident conditions that are of relevance for the spent fuel pool;
- (d) Means for monitoring and controlling the water chemistry for operational states

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Not accepted. The original term concerning early or large release is correct.

(j) Means for removal and inactivation of sodium adhered to the fuel during the transport from a sodium environment to a water pool, in order to prevent fuel damage and for keeping water quality of the water pool.

(k) Means for providing adequate heat removal from the fuel and for monitoring its status in operational states and in accident conditions, including during long term loss of all AC power supplies.

(l) For reactors using a sodium tank system for fuel storage, the design shall include the following:

- (a) Means for monitoring and controlling the sodium temperature for operational states and for accident states that are of relevance for the fuel storage tank;**
- (b) Means for monitoring and controlling the sodium level in the fuel storage tank and for detecting leakages for operational states and for accident conditions that are of relevance for the fuel storage tank;**
- (c) Means for monitoring and controlling the activity in sodium and in cover gas for operational states and means for monitoring the activity in sodium and in cover gas for accident conditions that are of relevance for the fuel storage tank;**
- (d) Means for monitoring and controlling the water chemistry for operational states;**
- (e) Means for preventing the uncovering of fuel assemblies in the tank in the event of a leakage;**
- (f) Means for providing adequate heat removal from the fuel and for monitoring its status in operational states and in accident conditions, including during long-term loss of all AC power supplies;**
- (g) Means for preventing sodium freezing to avoid blockages of coolant circulation.**

6.11 Radiation Protection

Criterion S1: Design for radiation protection

Provision shall be made for ensuring that doses to operating personnel at the nuclear power plant will be maintained below the dose limits and will be kept as low as reasonably achievable, and that the relevant dose constraints will be taken into consideration.

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6.69 Radiation sources throughout the plant, **including radioactive sodium coolant** shall be comprehensively identified and exposures and radiation risks associated with them shall be kept as low as reasonably achievable ^{B4}, the integrity of the first cladding shall be maintained, and the generation and transport of corrosion products and activation products shall be controlled.

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

6.71 For the purposes of radiation protection, provision shall be made for preventing the release or the dispersion of radioactive substances, radioactive waste and contamination at the plant.

6.72 The plant layout shall be such as to ensure that access of operating personnel to areas with radiation hazards and areas of possible contamination is adequately controlled, and that exposures and contamination are prevented or reduced by this means and by means of ventilation systems.

6.73 The plant shall be divided into radiation zones that are related to their expected occupancy and to radiation levels and contamination levels in operational states (including refuelling, maintenance and inspection) and to potential radiation levels and contamination levels in accident conditions. Shielding shall be provided so that radiation exposure is prevented or reduced.

6.74 The plant layout shall be such that the doses received by operating personnel during normal operation, refuelling, maintenance and inspection can be kept as low as reasonably achievable, and due account shall be taken of the necessity for any special equipment to be provided to meet these requirements.

6.75 Plant equipment subject to frequent maintenance or manual operation shall be located in areas of low dose rate to reduce the exposure of workers.

6.76 Facilities shall be provided for the decontamination of operating personnel and plant equipment.

Criterion S2: Means of radiation monitoring

Equipment shall be provided at the nuclear power plant to ensure that there is adequate radiation monitoring in operational states and accident conditions.

6.77 Stationary dose rate meters shall be provided for monitoring local radiation dose rates at plant locations that are routinely accessible by operating personnel and where the changes in radiation levels in operational states could be such that access is allowed only for certain specified periods of time.

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Number: 1 Author: BME NTI Subject: Highlight Date: 15.07.2021 10:22:11 -02'00'
Modifications are not relevant for SCWR.

Annex 3: The comments of the review on SDG document [14]

The review the SDC document [14] performed by the safety experts of BME NTI is documented below. The relevant pages from the document SDC document [14] are depicted on the left with the comments on the right.

1. INTRODUCTION

1.1. Background and Objectives

Safety Design Criteria (SDC) for Generation-IV (Gen-IV) Sodium-cooled Fast Reactor (SFR) systems [1] have been developed by an SDC Task Force (TF), under the auspices of the GIF Policy Group (PG). Following approval by the PG, the SDC Report was distributed to international organisations and national regulatory bodies for review.

The following needs were identified by the SDC TF members:

- ▶ Development of detailed guidelines to support practical applications of SDC.
- ▶ Expansion of selected topics, specifically on:
 - ✓ **Practical elimination (PE) of severe accidents,**
 - ✓ design basis of specific components (e.g. containments).
- ▶ **Justification of technical issues such as:**
 - ✓ prevention and mitigation measures against sodium fire and sodium-water reaction accidents;
 - ✓ implications of the fact that the reactor core is not in its most reactive configuration.

In response, the PG and the GIF Senior Industrial Advisory Panel recommended the TF to:

- ▶ develop guidelines for application of the SFR SDC, and
- ▶ pursue justification of key criteria to demonstrate the safety advantages of Gen-IV SFRs over current LWR designs.

The objective of this report is to provide recommendations and guidance on how to comply with the GIF SFR Safety Design Criteria. It presents examples for the measures stated in criteria as the best practices to help the designers achieve high levels of safety. The guidelines are placed below the SDC in the hierarchy of safety standards as depicted in Figure 1 below. Initially, the guidelines will focus on specific safety concerns, such as reactivity characteristics of SFRs and heat removal issues. In the future, the guidelines will be extended by taking into account additional issues, such as fuel handling and storage. These guidelines deal only with safety issues, security considerations are dealt by the Proliferation and Resistance and Physical Protection Working Group (PRPPWG) of GIF or by national regulators.

The guidelines, except Chapter 4 "Guidelines for application of safety design criteria", provide recommendations and guidance on how to comply with the SFR SDC, which are generally applicable for Generation-IV SFR systems. Chapter 4 covers specific issues to be considered in the design of structures, systems and components (SSCs) with examples of design provisions.

6

Summary of Comments on Chapters 1 & 2 Contents

Page: 1

Number	Author/bois	Subject/Highlight	Date
1		As "practical elimination" plays a main role in the safety approach, it would be useful to describe, which regulation defines its probability value.	10.09.2021 165503 +02:00
2		Not relevant.	12.07.2021 235621 +02:00

1.2. Scope of the Safety Design Guidelines

The primary focus of this report is to provide Safety Design Guidelines (SDG) for Gen-IV SFR systems as a technical supplement to the SDC report. The enhancement of Gen-IV SFR safety is mainly focused on improving each level of Defence-in-Depth, including the 4th level, with particular attention on the robustness of safety demonstrations (practical elimination demonstrations, independence of lines of defence). The definition of Defence-in-Depth and plant state follows the definition in IAEA SSR 2/1 (2016), which consults INSA-G-12 for the Defence-in-Depth principle: i.e. the plant states shown in Figure 2 are operational states include normal operation and anticipated operational occurrences, accident conditions include design basis accidents and design extension conditions.

Defence-in-Depth Levels				
Level 1	Level 2	Level 3	Level 4	Level 5
Plant states (considered in design)				
Operational States		Accident conditions		
Normal operation	Anticipated operational occurrences	Design basis accidents	Design extension conditions	
			Without significant fuel degradation	With core melting
				Off-site emergency response (out of the design)

Figure 2 Defence-in-Depth level and Plant States (including Severe Accident) based on IAEA INSA-G-12 & SSR-2/1 (Rev. 1, 2016)

Like Generation-III Light Water Reactors (LWRs), Gen-IV SFR safety is primarily based on the use of multiple-redundant engineered safety features to lower the probability of accidents and to limit the consequences of anticipated operational occurrences and design basis accidents. These safety features include independent and diverse scram systems, multiple coolant pumps and heat transport loops, decay heat removal systems, and multiple barriers against release of radioactive materials. In addition to these features, passive/inherent features for cooling and shutdown / power reduction should also play a significant role in the safety performance of Gen-IV SFRs by improving the diversity of safety systems and reducing reliance on electrical power and external water sources during design extension conditions.

For very low probability Design Extension Conditions (DECs), the following two major groups of accidents merit special attention due to challenges in designing an SFR to successfully respond to these accidents without severe consequences:

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Page: 2

Number	Author/bois	Subject/Highlight	Date
1		The use of DID levels defined in WENRA reference level should be considered here (as it is already used in the EU). In that version Level 3 is divided into 2 parts: (DEC-4 and DEC-1) so level-4 means the management of severe accidents.	13.07.2021 074321 +02:00

4 MAIN CHARACTERISTICS OF GEN-IV SFR SYSTEMS

A Sodium-cooled Fast Reactor (SFR) uses liquid sodium as the reactor coolant, as it can maintain fast neutron spectrum and allows high power density with low coolant volume fraction, at low pressure. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air and water and requires a sealed coolant system. The current plant size options range from small modular reactors, 50 to 300 MWe, to larger plants up to 1,500 MWe. The coolant outlet temperature of primary system is 500 - 550°C, which allows the use of materials developed and proven in prior fast reactor programmes. An SFR closed fuel cycle enables generation of fissile fuel and facilitates the management of minor actinides. Important safety features of SFR systems include core reactivity characteristics such as shorter neutron lifetime, a large margin to coolant boiling during normal operating conditions, a primary system that operates at near atmospheric pressure, and an intermediate coolant loop between the radioactive sodium in the primary circuit and the power conversion system. Water/steam, nitrogen and supercritical carbon-dioxide are among the working fluids considered for the power conversion system to achieve high efficiency, safety and reliability.

Much of the basic technology of the SFR has been established in former fast reactor programmes, and is confirmed by the Phébus end-of-life tests in France, the initial start-up test of Moxju in Japan, the lifetime extension of BN-600 and the start-up of BN-800 in Russia, as well as the start-up of the China Experimental Fast Reactor.

A high level of safety can be achieved for an SFR using a combination of engineered active, passive and inherent features, which may also accommodate transients and bounding events with significant safety margins.

An SFR can be arranged in a pool layout or in a compact loop layout. The following **three options are considered in the GF SFR System Research Plan**, with examples provided in Figures 3-1 to 3-4 [2].

- ▶ A large size (600 to 1,500 MWe) loop-type reactor with mixed uranium-plutonium oxide fuel and potentially minor actinides, supported by a fuel cycle based on advanced aqueous processing at a central location serving a number of reactors, as shown in Figure 2-1.
- ▶ An intermediate-to-large size (300 to 1,500 MWe) pool-type reactor with oxide or metal fuel, as shown in Figures 2-2 and 2-3.

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Page: 3

Number: 1	Author: bots	Subject: Highlight	Date: 10.09.2021 165832 +02'00'
The topic is important, the content is not relevant for SCWRs.			
Number: 2	Author: bots	Subject: Highlight	Date: 10.09.2021 165816 +02'00'
Not relevant for SCWR.			

3. GENERAL APPROACH

3.1. Design Basis and Residual Risk

In the design of a nuclear power plant, compliance with the fundamental safety objectives¹ [1] should be demonstrated for all initiating events. Initiating events are identified and grouped into a limited number of plant states, primarily on the basis of their frequency of occurrence. Initiating events in the plant states [3], described below, provide the design basis² for the safety design of nuclear power plants, while the residual risk is not included in the plant states. Situations to be practically eliminated are considered to be part of the residual risk. An illustration of design basis and residual risk is given in Figure 4.

3.1.1. Plant States for Design Basis

Normal operation – The plant is operating as intended, with all plant parameters (temperature, pressure, etc.) within the design ranges for normal operation, and is considered for the development of design measures for Defence-in-Depth Level 1 [1].

Anticipated Operational Occurrence (AOO) – Includes events, which disrupt plant conditions from their normal state and are expected to occur during the lifetime of a plant, and which are typically caused by a failure or an inadvertent operation of a single SSC, accommodated by the safety systems. When one selects initiating events in each plant state, safety classification of SSCs are taken into account in the relation of its frequency of failure. A failure of an active system manufactured to normal industrial standards would be treated as an AOO. The expected frequency of occurrence of AOOs should be estimated according to component reliability, but it should not be lower than approximately 1×10^3 per reactor year because such events may occur during the lifetime of the plant. Plant conditions, expected as a result of an AOO, are considered for the development of design measures for Defence-in-Depth Level 2 [1]. Any significant consequences of an AOO should be prevented (no clad or fuel damage and no significant release of radioactive materials).

Design Basis Accident (DBA) – Includes events, which are typically caused by the failure of a single SSC with consequences of greater severity than those considered for AOOs and which

¹ The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits. (Criterion 5 in SDC) The dose limits are defined by each national regulatory authority.

² The range of conditions and events to be explicitly taken into account in the design of a facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems. The authorized limits are defined by each national regulatory authority.

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Page: 4

Number: 1	Author: bots	Subject: Highlight	Date: 3.09.2021 07:49:56 +02'00'
It is important to mention here that power changes, shutdowns, start, maintenance and refueling are part of NO.			

are not expected to occur during the lifetime of a plant. **The expected frequency of occurrence of DBAs is less than 1×10^{-7} per reactor year, and as low as approximately 1×10^{-8} per reactor year or less, consistent with global probabilistic objectives assigned to core damage frequencies.**

Plant conditions, especially as a result of a DBA, are considered for the development of safety systems for Defence-in-Depth Level 3 [1]. Reactor and plant systems should respond to prevent any significant core damage or radioactive release exceeding acceptable limits. Although a limited number of fuel pin failures may be acceptable, these should be prevented as far as reasonably practical.

Design Extension Condition (DEC) – Accident conditions that are typically of lower probability than design basis accidents and involve the failure of more than one SSC important to safety or part of a safety system. Design extension conditions should include accident sequences that could lead to severe accident conditions and are considered for design measures to prevent core damage and mitigate core damage as shown in Figure 4. Plant conditions, expected as a result of such events, are considered for the development of design measures for Defence-in-Depth Level 4 [1].

3.1.2. Situations to be practically eliminated situations and Residual Risk

Situations to be practically eliminated situations – Specific situations, whose consequences can lead to early or large radioactive release and which cannot be managed by the design at acceptable conditions, have to be practically eliminated by implemented design measures[4]. These situations have to be demonstrated either as physically impossible by design, or as extremely unlikely to arise with a high level of confidence. Situations to be practically eliminated situations are part of the residual risk.

Residual Risk – Accidents with sufficiently low frequency due to the implementation of defence in depth principle and not considered in the design basis.

³ DBAs retained in the safety demonstration are defined as “envelopes” of families of abnormal operating conditions that gather several initiating events of the same type.
⁴ DBAs definition is as follows – “Event sequences that would lead to an early radioactive release or a large radioactive release are required to be “practically eliminated. The possibility of certain conditions arising may be considered to have been “practically eliminated” if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.”

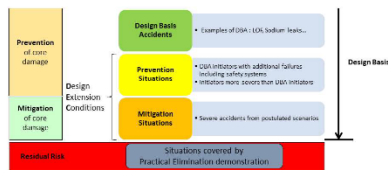


Figure 4 Illustration of design basis and residual risk

3.2. General Approach to Normal Operation, ACOs and DBAs

In the new SFR designs, strong emphasis should be given to the prevention, detection and control of accident sequences [5]. An SFR must be designed to allow for stable normal operation, which requires being able to control reactor temperatures, within a small range set by the designers, possibly also by varying the reactor coolant flow using the coolant pumps, and by keeping the reactor power in balance with the demand for power from the electric grid. Reactor power is regulated using control rods, which are moved in response to the changing demand for power production, such that the steam flow rate and temperature at the turbine can be maintained.

ACOs are expected to be managed to return to a normal operation condition by the plant control provision as that reactor shutdown and decay heat removal systems are not required to operate.

While the reactor shutdown and decay heat removal systems should respond to ACOs if needed, DBAs are managed by using safety systems to shut down the reactor and remove decay heat. These rapidly responding safety systems can be actuated by using a signal from the plant (i.e. the plant protection system) that detects an off-normal condition of sufficient magnitude (set by

¹ In general, ACO and DBA for the safety analysis are selected as envelopes and representatives of similar kind of events. Therefore most cases of ACOs require activation of safety systems such as automatic reactor shutdown and decay heat removal. Actually ACO domain includes minor troubles, which doesn't require the automatic reactor shutdown. For such cases, measures should be provided to detect and control deviations from normal operation in order to prevent ACOs from escalating to accident conditions. Plant equipment such as normal reactor control system and balance-of-plant is used for that purpose.

the plant designer). **The goal, in cases of AOCs or DBAs, is to ensure the reactor core and system temperatures remain within applicable design limits.**

3.3. General Approach to Design Extension Conditions

This document is focused on events that begin with a reactor **under normal power operation**, as the initial plant condition, excluding accident conditions associated with fuel handling and storage facilities. The following section describes the general approach for the consideration of DECs in the design and involves identifying postulated events to be included in DECs.

3.3.1. Application to Design

Design provisions for DECs are assigned within the fourth level of the Defence-In-Depth (DiD) as shown in Figure 5. This level includes measures for "prevention of core damage" in "prevention situations" and for "mitigation of core damage" in "mitigation situations".

The goal of design measures in "prevention of core damage" of DiD Level 4 is to provide lines of defence to prevent conditions leading to significant core damage. Design features for "prevention of core damage" in DECs deal with accident sequences that are typically caused by failure of one or more systems related to safety, such as the reactor coolant pumps, followed by failure of other safety systems needed to prevent excessive power and/or temperatures resulting from the off normal conditions of the plant. "Prevention situations" of DECs also include initiating events, more severe than those in DBAs.

Design measures in "mitigation situations" provide design features for mitigation of consequences of postulated accidents where significant core damage occurs, with the objective of maintaining the containment function to limit radioactive release within applicable design limits and practically eliminating a large or early release which would lead to unacceptable off-site consequences.

As described in the following section, design provisions to achieve In-Vessel Retention (IVR) is crucial for addressing "mitigation situations", since the reactor vessel (RV) can, in this case, serve as the boundary for retention and cooling of core material, limiting any threat to the containment.

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Page 7

Number:1	Author:bois	Subject:Highlight	Date:3.09.2021 11:18:53 +02:00
The general description is useful and could be used for SCWRs as well.			
Number:2	Author:bois	Subject:Highlight	Date:3.09.2021 11:18:48 +02:00
Should be replaced with "under normal operation" that includes the shutdown states as well.			

3.3.3. Exploiting SFR Characteristics to Enhance Safety

Inherent and Passive Safety

Inherent reactivity feedback effects are obtained by using intrinsic SFR features to reduce power as the core temperature rises in accident conditions. The large temperature margin to sodium boiling (e.g. from maximum coolant channel temperature in normal operation to about 900°C until boiling) of the reactor coolant provides sufficient room to use reactivity feedback due to thermal expansion of core components, as well as neutron leakage effects due to the change in coolant density. When the change in coolant density may contribute positively to the overall reactivity, it is important that the net reactivity (with contributions from all feedback effects) is negative to foster the potential for inherently safe behaviour in case of a ATWS.

Passive shutdown systems, such as the Self-Actuated Shutdown System (SASS) [6] are also applicable. In SASS, a Curie-point magnetic alloy is utilized for automatic de-latching of control rods under high coolant temperature accident conditions, higher than for normal operating conditions, but still below the coolant boiling point. A Hydraulically Suspended Rod (HSR) [7] system (also called "flow-levitated absorbers"), where the control rods are automatically dropped into the core when the hydraulic force is reduced under accident flow reduction conditions, could also be used. In fast reactors, due to the sensitivity of the core reactivity to neutron leakage, it is also possible to consider using concepts like the Gas-Expansion Module (GEM) [8], where a decrease in core inlet pressure is exploited to increase neutron leakage under a flow reduction condition.

These inherent and passive safety features should contribute to improve the diversity of safety systems and to reduce reliance on electrical power and external water sources.

Decay Heat Removal (DHR)

Since an SFR is operated at nearly atmospheric pressure and at temperatures far below the coolant boiling point, coolant leakage or a pipe break does not lead to the same type of loss-of-coolant accident as postulated in an LWR, which has the potential for depressurisation, coolant boiling and loss of cooling capability. The requirements for core cooling of an SFR comprise keeping the sodium coolant level above the reactor core and the circulation of the liquid coolant to an appropriate heat sink for decay heat removal. As long as these two requirements are satisfied, significant core damage can be prevented. Depending on the plant states, various measures such as the normal heat transport system, a separate sodium-to-air heat exchanger, or any other system that would allow cooling of the sodium can be provided. Natural

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Page 8

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Not relevant.			
Number:3	Author:bois	Subject:Highlight	Date:3.09.2021 11:33:42 +02:00
Not relevant.			
Number:4	Author:bois	Subject:Highlight	Date:3.09.2021 11:43:29 +02:00
Description is not relevant for SCVSR. Similar description about SCVWR could be useful.			

circulation of a single phase sodium coolant can be effectively utilised if an adequate difference in height is available between the core and the heat exchanger, due to the fact that sodium has a relatively large density variation with temperature. Such passive Decay Heat Removal Systems (DHRSs) can be placed in different locations, e.g. in the reactor vessel (RV) or in the primary/secondary-coolant circuits. Alternative emergency cooling can be made available via steam generators and guard vessels (GVs) for enhancing diversity.

6. In-Vessel Retention (IVR)

For an SFR, in-vessel retention is a safety design strategy aimed at ensuring long-term retention of core materials inside the RV for any accident situation, including those resulting in degradation or loss of core integrity, by providing coolability of the core materials under sub-critical conditions. This is typically accomplished by providing the means to keep the core submerged under the sodium coolant and the decay heat removal paths available. Such an approach can be a key design measure to address "mitigation situations" for DECs.

3.4. Design Considerations for Design Extension Conditions

3.4.1. Anticipated Transient Without Scram (ATWS)

Postulated ATWS events cause an imbalance between generated power (heat) in the reactor core and its removal from the system, either by the normal path through the power production part of the plant, or by specific heat removal systems. If adequate heat removal is not provided, substantial core degradation will occur, which may lead to severe consequences, such as large energy releases. Reliable means of maintaining the balance between heat generation and heat removal must be provided to avoid such consequences. This can be accomplished by ensuring alternate means of shutdown, including the use of inherent and/or passive reactor shutdown, as long as sufficient heat removal capability is also provided. Provisions for retention and cooling of degraded core materials are necessary to mitigate the consequences of a core damage.

Provisions for ATWS can be summarised as follows:

- ▶ **"Prevention of core damage"**
Means for maintaining an acceptable balance between reactor power and heat removal capabilities should be provided to avoid core damage, given an assumed failure of the active reactor shutdown function in ACOs. These capabilities should include inherent and/or passive means. In order to terminate the accident, means for reactor shutdown should be provided.

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Related Criteria in the SDC Report

Criterion 20: Design extension conditions
Criterion 46: Reactor shutdown⁶

- ▶ **"Mitigation of core damage"**
Provisions for prevention of a large energy release that could threaten the integrity of the containment and provisions for long-term cooling of a degraded core to avoid reactor coolant boundary failure, should be made available for achieving IVR against unprotected transients⁷ with core damage.

Related Criteria in the SDC Report

Criterion 20: Design extension conditions
Criterion 44: Structural capability of the reactor core⁸
Criterion 45: Control of the reactor core⁹
Criterion 47: Design of reactor coolant systems¹⁰
Criterion 51: Decay heat removal system¹¹

6.2. Loss of Safety Systems for Decay Heat Removal

In a situation of a failed DHRS after the reactor has been shut down and the heat generation has dropped to only a few percent of the nominal power shortly, the temperature of the reactor

⁶ *Criterion 46: Reactor shutdown*
6.9. The means for shutting down the reactor shall consist of at least two diverse and independent systems. For design extension conditions, passive or inherent reactor shutdown capabilities shall be provided to prevent severe core degradation and to avoid re-criticality in the long run.

⁷ "Unprotected transients" means accident sequences with failure of active reactor shutdown system, which include ATWS.

⁸ *SDC Criterion 44: Structural capability of the reactor core*
For the design extension conditions, provisions shall be included to avoid re-criticality resulting in potentially large mechanical energy release during a core disruptive accident.

⁹ *SDC Criterion 45: Control of the reactor core*
6.05s. To avoid significant mechanical energy release during a core disruptive accident, the reactor core shall be designed to have favourable neutronic, thermal, and physical characteristics, considering all reactivity feedbacks, including sodium void worth, to mitigate the consequences of such design extension conditions.

¹⁰ *SDC Criterion 47: Design of reactor coolant systems*
6.18s. Components, which constitute the reactor coolant boundary, shall be designed to maintain the boundary function and to maintain a sufficient sodium inventory in the primary coolant system in case of anticipated transients without scram.

¹¹ *SDC Criterion 51: Decay heat removal system*
6.18s. Means shall be provided for the capability of core cooling under postulated plant conditions with core degradation.

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coolant system, including the core, coolant and reactor coolant boundary, increase. The rate at which the temperature increases depends on the overall heat capacity of the system, and it may take a long time before reaching temperatures that would threaten the core or system integrity. It is therefore possible to consider recovery actions for failed DHRs and/or to implement back-up cooling measures, before reaching unacceptable temperatures for SSCs. However, if no heat sink is available the coolant boundary will eventually fail due to creep damage, leading to release of radioactive fission products and sodium vapours into the containment. Such situations should be practically eliminated by design measures for enhanced core cooling capabilities (described in Section 3.5.3). Provisions can be summarised as follows:

► **"Prevention of core damage"**
Extension of the DHRs (normally designed for DBAs) capability should be considered, and other alternative cooling provisions should be made available to prevent core damage and reactor coolant boundary failures due to overheating, given the assumed causes of DHRs failures as DECs.

Related Criterion in the SDC Report
Criterion 51: Decay heat removal system¹¹

3.4.3 Reactor Coolant Level Reduction

If the core is uncovered, following events causing a reduction of the reactor coolant level, it is impossible to avoid core melt. Depending on the course of the accident and under some circumstances, significant radioactive material would be released into the containment atmosphere. Therefore, an uncovered core configuration should be practically eliminated by design measures (described in Section 3.5.3). Provisions can be summarised as follows:

¹¹ SDC Criterion 51: Decay heat removal system

6.19 The decay heat removal system shall be designed as follows:

- To provide diversity to the extent practicable and redundancy for reducing common cause failures, including external events.
- To prevent freezing of the sodium coolant to avoid blockage of coolant circulation, and
- To provide detection and mitigation measures against postulated decay heat fluid leaks.

6.19a. In design reference conditions, means for decay heat transfer shall be provided, in addition to a decay heat removal system for anticipated operational occurrences and design-basis accidents, with the conditions listed below:

- The cooling of the reactor core is possible even under extreme external hazards, and their consequences, such as long-term loss of all AC power supplies.
- Passive mechanisms are used to the extent practicable, and
- Decay heat removal system has diversity to the extent practicable.

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Page: 11

Number	Author	bois	Subject	Highlight	Date	3	09	2021	13:54:40	+02:00
1										
Not necessarily true for SCWRs.										
2										
Instead of Level Reduction, LOCA should be applied here.										
3										
Do not agree. Core integrity is the predecessor of a core damage / melting, for which we have the CDF limit. Core melting is a typical BECC event (or maybe fuel melting with uncovered core is outside of the DEC2 conditions?).										
4										
Not relevant.										

► **"Prevention of core damage"**
Reactor Vessels (RVs) and Guard Vessels (GVs) should be designed, manufactured, installed, maintained and inspected to have the highest level of reliability in order to prevent double leakage from RVs and GV. For a loop-type reactor, measures for ensuring a minimal primary coolant level to prevent core damage should be provided against postulated leakage from the primary loop components and piping. If double leakage from RVs and GV, cannot be practically eliminated, the situation has to be considered for implementing design provisions.

Related Criterion in the SDC Report
Criterion 49: Level of reactor coolant¹²

3.4.4 Conditions Considered for DEC Design Provisions

When considering design provisions for the prevention of core damage and mitigation of the potential consequences of DECs, the following points need to be taken into account:

- Identification of expedited safety functions
- Identification of accident conditions for the expected safety functions
- Design to ensure performance under a postulated accident condition and, if necessary, design to ensure performance of any required supporting system, such as plant protection and control systems, and power supplies, under the postulated accident condition.
- Provision of emergency operation manuals for accident diagnosis and management actions, if necessary
- Performing safety evaluations with validated analytical codes. The evaluations will be based on best estimates to show that design limits are not exceeded.
- Performing reliability analyses to re-tweak risk reduction effects against the initiating event, to complement the deterministic analysis as needed.
- Consideration of the independence of the selected design provisions from other safety related SSCs.

¹² SDC Criterion 49: Level of reactor coolant

Guard vessels and guard pipes shall be designed so as to maintain the sodium surface of the primary coolant system at a level necessary for decay heat removal in the case of a sodium leak accident in the primary coolant system. Due considerations shall be taken of a dependent failure and a common cause failure between the reactor vessel and the guard vessel, as well as between main coolant pipes and guard pipes. Provisions shall be made to reduce the amount of sodium that leaks from the primary coolant system in case of a failure of the reactor coolant boundary.

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Page: 12

Number	Author	bois	Subject	Highlight	Date	3	09	2021	14:38:30	+02:00
1										
Not relevant for SCWR.										
2										
Not relevant.										

► Consideration of testability of the abovementioned provisions. Since simulation of severe plant conditions in DEC's are generally not possible during normal reactor operations, a best-estimate analysis based on related data, including those obtained during normal reactor operations and their extrapolation, those obtained in experimental installations to faulted conditions as far as possible, will be used to evaluate the effectiveness of the safety functions throughout the life-time of the plant.

Probabilistic Safety Assessments (PSAs) should be implemented at the design stage to support the estimated probability of failures for initiating events, and the adequacy of the design measures for addressing events in DEC's. PSA results will also provide the means to evaluate that no particular feature or initiating event makes a disproportionately large or uncertain contribution to the overall risks.

3.5. Practical Elimination of Accident Situations

This section describes the design approach for practical elimination of accident situations, examples of individual situations and considerations for design measures.

One of the high-level GF safety goals is "elimination of the need for offsite emergency response". In order to achieve this goal, all scenarios that could lead to early or large release of radioactivity to the environment need to be prevented, mitigated, or practically eliminated. GF SFR SDC and SDG approach places an emphasis on prevention, and then on mitigation.

Some scenarios for which the mitigation measures may not be sufficient to manage the core damage and subsequent release of radioactivity (or they are too costly to implement) will have to be practically eliminated. These scenarios are usually design-specific, but the selected list included in the SDC report is fairly common set of practically eliminated accidents considered for the Gen-IV SFR design tracks.

3.5.1. Application to Design

► An early or large radioactive release is subject to the practical elimination approach

► **The approach should be applied to any operation state of the plant, including fuel handling and spent fuel storages.**

► Situations, which may lead to early or large radioactive release and which cannot be mitigated under acceptable conditions, are identified to be practically eliminated by implementation of design provisions

► The approach is intended to demonstrate that the identified situation is physically impossible by design, or that the implemented provisions reduce the likelihood to a residual risk with a high degree of confidence.

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Number	Author	bois	Subject	Highlight	Date	4.09.2021	16:19:24	+02:00
1								

Earlier it was stated that the SDC does not apply for fuel handling and storage events.

► Practical elimination can be considered as part of a general approach and as an enhancement of the Defence-in-Depth principle. The design should restrain practical elimination to a very limited list of situations.

Related description in the SDC Report
Section 2.2.4 Prevention of cliff edge effect: **Severe accidents that could lead to a significant and sudden radioactive release, not reasonably manageable by design improvement, shall be practically eliminated by appropriate design provisions.**

3.5.2. Identification of Situations to be Practically Eliminated

All potential situations which might lead to early or large radioactive release should be considered as the situations to be practically eliminated. Some examples of situations to be practically eliminated are as follows:

(1) Severe events with mechanical energy release exceeding the capability of the containment

► Power excursions for intact core situations (during power operation)

- ✓ **Large gas flow through the core**
- ✓ **Large-scale core compaction**
- ✓ Collapse of the core support structures

Note: As far as reasonably possible, severe accidents have to be managed with mitigation means, before implementing a practical elimination demonstration.

(2) Situations leading to failure of the containment with a risk of fuel damage

► Complete loss of the decay heat removal functions, leading to core damage and failure of the reactor coolant boundary

► **Core uncovering due to sodium inventory loss**

(3) Fuel degradation in the fuel storage or when the containment is not functional due to maintenance (e.g. opening containment for replacement of large equipment)

- Core damage without a functioning containment
- Spent fuel melting in the storage

Table 1 shows the reasons for choosing the specific examples for an SFR. An example of an approach to identify situations to be practically eliminated is provided in Appendix (A). **In practice, situations to be practically eliminated will be determined by using a concrete scope to the design team by step. Identification of situations to be practically eliminated should be made as the results of tradeoff between practical elimination and mitigation. Robust design revision shall be provided for the demonstration of the practical elimination. Design modification shall be made, if it is not sufficient to the demonstration.**

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Number	Author	bois	Subject	Highlight	Date	10.09.2021	00:01:46	+02:00
1								
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Not exist citation.
Risk failure of severe accident management systems.
Not relevant.
Not relevant.

5.3.3 Design Considerations for Situations to be Practically Eliminated

(1) Power excursions for intact core situations

a) Large gas flow through the core

The primary coolant system should be designed to prevent or limit cover gas entrainment from the sodium free surface in the reactor vessel and to limit or prevent gas accumulation in structures and components submerged in sodium. In addition to the gas entrainment from the sodium free surface, any gas generation and release consequences which possibly occur in the reactor coolant system, including fission gas release, should be considered. Gas release paths are necessary where gas accumulation might occur. An evaluation should be made to show that an accidental gas ingress, entrainment and transport through the core does not cause prompt criticality. Such accidental gas ingress might happen in abnormal conditions, e.g. as a consequence of a primary pump over-speed.

Related Criteria in the SDC Report
Criterion 43bis: Plant system performance of a sodium-cooled fast reactor
Criterion 45: Control of the reactor core

b) Large-scale core compaction

Reactivity insertion, due to motion and deformation of fuel assemblies and other assemblies such as control rod assembly and shielding assembly, should be limited to prevent core damage following any possible causes, including earthquakes. Due consideration should be taken to the fuel assembly design to prevent compaction. Adequate stiffness of the core should be ensured for each plant state, through the suitable design for the core subassemblies, core support plates and the core restraint system, if provided. The gap between the core subassemblies should be adequate taking any core conditions assumed in normal operation, e.g., reactor shutdown state, nominal power operation, and partial power operation into account.

Control rod insertion should be assured with sufficient geometrical clearance margin for a large earthquake. In order to maintain subcritical conditions after control rod insertion, when the control rods disconnected from the drive mechanisms, the upward movement of the inserted control rods should be limited during an earthquake.

Related Criteria in the SDC Report
Criterion 42bis: Plant system performance of a sodium-cooled fast reactor
Criterion 44: Structural capability of the reactor core

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Number	Author	bois	Subject	Highlight	Date	5_OB_2021_1140Z1	+0200
Number 1	Author: bois	Subject: Highlight	Date: 5_OB_2021_1140Z1	+0200	As a comparison, according to the Hungarian Nuclear Safety Code (NSC), the situations to be practically eliminated are the following:		
<ul style="list-style-type: none"> - breach of the pressure vessel - prompt criticality - loads endangering the containment integrity - loss of heat removal from SF storage causing fuel damage - loss of coolant accident with open containment building. 							
Number 2	Author: bois	Subject: Highlight	Date: 5_OB_2021_1140Z1	+0200	Not relevant.		
Number 3	Author: bois	Subject: Highlight	Date: 5_OB_2021_1145Z6	+0200	The consideration of core compaction should be re-evaluated. The elimination of prompt supercriticality is more general requirement. The loss of the shutdown system is a typical DEC event for which design provisions should be made.		

a) Collapse of the core support structure

The core support structure should be designed to ensure a sufficient safety margin, subject to design rules required by the regulator and including the uncertainties against mechanical and thermal loads. In addition, it should, during its lifetime, cope with environmental conditions and aging of core support structure materials. The core support structure shall not undergo high radiation dose. Operation of this structure in brittle domain is forbidden. A minimum material ductility should be preserved until the end of life of the reactor. In normal operation, temperature of the core support structures should remain in the domain of negligible creeping of the stainless steel. Detection of potential core support deformation or failure should be provided, e.g. inspection capability by using ultrasonic detectors and monitoring plant parameters, such as flow rates and temperatures.

Related Criteria in the SDC Report
Criterion 42bis: Plant system performance of a sodium-cooled fast reactor
Criterion 44: Structural capability of the reactor core

b) Complete loss of the decay heat removal function, leading to core damage and failure of the reactor coolant boundary

- For DBA, a decay heat removal system should be provided to deal with initiating events typically caused by single failure of an SSC.
- For DEC, design measures should be provided against initiating events, which are more severe than DBAs, or which originate from multiple failures of SSCs.
- Proven technology, based on the design, construction and operation experience of SFRs, should be applied to the basic design of decay heat removal systems.
- Extension of capabilities to deal with DEC, e.g. additional decay heat removal systems, increased capacities of heat removal, and operation with natural as well as forced circulation, should be considered. Application of mobile power sources and manual operation can reinforce the design in case of extreme external hazards of low probability even with their uncertainty of availability.
- Ensuring diversity in systems is essential for improving the overall reliability. Redundancy of systems does not bring the same reliability benefits. It is required to maintain heat removal functions, even under postulated severe external hazards, such as earthquakes, flooding, tsunamis and missiles leading to a common cause failure. Physical separation of the diverse or redundant systems can further reduce the potential for common cause failures.

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Number	Author	bois	Subject	Highlight	Date	5_OB_2021_1146Z0	+0200
Number 1	Author: bois	Subject: Highlight	Date: 5_OB_2021_1146Z0	+0200	See the previous item.		
Number 2	Author: bois	Subject: Highlight	Date: 6_OB_2021_1139Z5	+0200	See previous comments. If this situation includes the loss of passive ECC (safety system), it shall be eliminated. Other loss of BFR situations are not subject of practical elimination.		
Author: bois	Subject: Study Note	Date: 18_OB_2021_2145Z4	+0200	The scenario itself should not lead to large early release, and as a consequence, it is not subject of practical elimination.			
Number 3	Author: bois	Subject: Highlight	Date: 18_OB_2021_2152Z2	+0200	The role of diversity and redundancy is not the same, their reliability benefits cannot be compared.		

- ▶ In SFR should proactively utilise its natural circulation capability to an ultimate heat sink (atmosphere), since this can significantly contribute to improving the reliability of the heat removal capability, even under long-term loss of power supplies. Natural circulation can be used as a measure for DBAs, as well as for DECs.
- ▶ Robust demonstrations of practical elimination should consider independence between safety systems for DBAs and decay heat removal capabilities for DECs. If necessary, additional independent decay heat removal systems should be installed.
- ▶ It is necessary to clarify all credible factors leading to loss of decay heat removal function and to demonstrate that measures can be implemented to overcome all of them. "Credible factors" are the initiating events concurrent with potential subsequent failures of required safety related systems, structures and components (including common cause failures and dependent failures of redundant systems). Broad range of situations including internal and external events and associated plant states (AOCs, DBAs and DECs) should be comprehensively considered to assess their influence on decay heat removal function.
- ▶ Each system, related to decay heat removal, should be able to demonstrate that it can perform its function as expected.

Related Criteria in the SDC Report
Criterion 42bis: Plant system performances of a sodium-cooled fast reactor
Criterion 51: Decay heat removal system

4) Core uncovering due to sodium inventory loss

- ▶ The RVs and GVs should be designed, manufactured, installed, maintained and inspected to have the highest level of reliability.
- ▶ Due design considerations should be taken to prevent dependent failures and common cause failures between RVs and GVs, even under postulated severe external hazards, such as earthquakes.
- ▶ The reliability of these components may also be challenged by a failure of any adjoining SSCs. The impact of such a failure should be prevented wherever possible by layout optimisation and/or provision of adequate protection by features designed to withstand the global and local loads which may be placed on them.
- ▶ In case of a RV leakage, the GV should be designed considering a single failure in the pressure control between the inside and outside RV so that the reactor core remains covered and the decay heat removal paths are maintained.

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Not relevant.			

Table 1: Examples of Situations to be practically eliminated for an SFR

SFR	
Situations to be practically eliminated	Reason for choice
Power excursions from intact core situations (Extreme initiating events)	In case an accident involving uncontrollable reactivity insertion of more than 1% occurs, it may lead to prompt criticality and a severe power excursion either LWR or SFR. Due to physics of fast neutron spectrum reactors, the SFR cores are not in their most reactive configuration. Therefore, the possible causes of such reactivity insertion in SFR, such as large-scale gas entrainment through the core, large-scale core compaction, and collapse of core support structure, should be practically eliminated because prevention and mitigation measures for them may not be effective to prevent an early or large release of radioactivity.
Complete loss of heat removal function that could lead to core damage and failure of the reactor coolant boundary	If no heat sink is available, a coolant boundary failure will occur, either due to creep damage anywhere in the primary and secondary coolant systems, or from melt-through by degraded core materials.
Core uncovering due to sodium inventory loss	If the core becomes uncovered, it is impossible to avoid a core melt. Depending on the course of the accident and under some circumstances, significant radioactive materials would be released into the containment atmosphere.
Core damage during maintenance, or spent fuel melting in the storage	If a core damage or fuel melting occurs during maintenance (when the containment is not functional), or at the fuel storage outside the containment, significant radioactive release might happen.

This table provides examples not intended to be comprehensive and not implied to be generally applicable to all SFRs. It should be established in the frame of detailed design and safety analysis.

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Not relevant, but a useful summary.			

Control rod insertion should be assured even in case of deformation of core components due to, for instance, irradiation or design basis earthquakes. The reactor shutdown system, which functions as reactor controller under normal operation, e.g. reactor start-up and power regulation, should also be designed so that any failure of the control function, such as control rod position change by motor drive mechanism, shouldn't affect the reactor shutdown function.

Related consideration
The reactor core should be designed to have favourable inherent safety characteristics, e.g. a negative power coefficient, to self-limit any reactor power increase in case of reactivity insertions.

► **Design consideration**

- ✓ Duration of a signal transmitting and processing to actuate control rod insertion
- ✓ Duration of control rod insertion
- ✓ Reactivity worth of control rods
- ✓ (Reactivity characteristics of reactor core)

(3.2) Diversity and independence

► **Functional recommendations**

Two shutdown systems should be designed to have independence and diversity to the extent practicable to prevent common cause failures. Examples of design measures are:

- ✓ Different structures/mechanisms for rod insertion, different designs for the control rods and their guide tubes
- ✓ Physical separation of electric distribution boards and cables, isolated arrangements, divided by walls, etc.
- ✓ Electrical isolation of protection system

Detection parameters for the reactor shutdown systems should be diverse to the extent practicable.

At least one of the two reactor shutdown systems should be designed according to a single failure criterion that applies to components of actuation mechanisms, detectors and signal processing systems.

Fail safe features, such as control rod insertion by loss of the holding function in case of electric power supply failure, should be considered.

► **Design consideration**

- ✓ Electrical and physical separation

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- ✓ Diversification of design (e.g. detection parameters for reactor shutdown, a combination of programmable digital I&C system and diverse system with a different operation principle.)
- ✓ Fail safe measures

(3.3) Consideration for environmental conditions

► **Functional recommendations**

Throughout their life in the reactor including conditions during DBAs, two shutdown systems should be designed to withstand environmental conditions, such as irradiation, temperature, chemical effects, geometrical changes.

► **Design consideration**

- ✓ Temperature, irradiation effects, liquid sodium environmental conditions, cover gaps, plenum with sodium aerosols.

Reactor Shutdown for DECs

(1) SDC
"For design extension conditions, passive or inherent reactivity reduction capabilities shall be provided to prevent severe core degradation and to avoid re-criticality in the long run."

(2) Design concept
Passive reactivity reduction or feedback capabilities, several different mechanisms, including those responding to coolant temperature increase or coolant hydraulic force change.
Inherent reactivity reduction in balance with heat removal capabilities due to negative reactivity feedback from Doppler effect, axial and radial expansion of core and structure, and even sodium expansion effects, if designed accordingly.

(3) Functional recommendations and Design consideration

(3.1) Capability of passive or inherent reactivity reduction mechanisms

► **Functional recommendations**

Adequate combination of passive reactivity reduction, passive feedback and inherent power reduction capabilities should be provided in case of an AOO with a failure of the active reactor shutdown systems, to avoid exceeding the design limits for DECs.

1) Passive reactivity reduction or feedback mechanisms should be designed to provide sufficient negative reactivity within an allowable time and to be activated and

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Measures to achieve and maintain a safe shutdown state and to monitor the plant conditions should be provided in case of failure of the active shutdown systems:

- ▶ Design consideration
 - ✓ Reactivity needed for safe shutdown
 - ✓ Parameters for monitoring the reactor condition after an accident

(3-4) Consideration for environmental conditions

- ▶ Functional recommendations
 - Passive reactivity reduction or feedback mechanisms should be designed to withstand environmental conditions and to maintain insertion function throughout the lifetime of the reactor and DBAs/DECs.
 - Reactivity effects of inherent reactivity reduction capabilities should be assessed throughout the lifetime of the reactor, considering environmental conditions.
- ▶ Design consideration
 - ✓ Effects of irradiation and temperature, liquid sodium environmental conditions, and geometrical changes

3.1.2 Mitigation of Core Leakage

The mitigation functions of core damage are defined in the Criteria 44, 45, 47 and 51 in the SDC for GIF SFRs. The related design guidelines are presented below. Objectives are to prevent prompt criticality in the course of core degradation and to ensure the coolability of core debris. Design measures against core damage from unprotected transients should be implemented for the following accident phases, according to the progression of a core damage:

- ▶ Initiating phase, accident phase from intact state up to inter-subassembly material motion onset, i.e. subassembly duct failure
- ▶ Transition phase, accident phase after initiating phase up to the establishment of stable cooling conditions
- ▶ Post accident heat removal, stable cooling condition for a long term

Mitigation in Initiating Phase

(1) SDC

"To avoid significant mechanical energy release during a core disruptive accident, the reactor core shall be designed to have favourable neutronics, thermal, and physical characteristics, considering all reactivity feedbacks, including sodium void worth, to mitigate the consequences of such design extension conditions."

(2) Design concept

Various core designs with different output power, different fuel material, i.e. oxide, metal and nitride. A common core geometry consists of fuel subassemblies containing the fuel pins in ducts of hexagonal cross-sections, typically referred to as subassembly ducts or hex-cans.

(3) Functional recommendations and Design consideration

(3-1) Limiting the total reactivity during unprotected transients

- ▶ Functional recommendations
 - Core reactivity characteristics should be designed so as to prevent prompt criticality, i.e. $\rho_{\text{net}} < 1\beta$ during the initiating phase of unprotected transients. Positive reactivity effects, such as sodium boiling, should be limited so that negative reactivity effects e.g. Doppler effect, fuel expansion and fission gas dispersion are sufficient to counteract the positive reactivity effects. Design parameters, such as sodium volume fraction, core height and other geometric parameters, should be properly chosen based on the effects of sodium void worth during transients.
- ▶ Design consideration
 - ✓ Core reactivity characteristics: sodium void reactivity, Doppler effect, fuel axial expansion, core radial expansion, control-rod drive-line expansion, etc.)
 - ✓ Sodium volume fraction and core height and geometry (e.g. heterogeneous arrangements, sodium plenum)

(3-2) Facilitating fuel reactivity effects

- ▶ Functional recommendations
 - Core design parameters, such as core height, should be properly chosen to obtain effective negative feedback due to fission gas dispersion. Fuel reactivity feedback is dependent on the choice of fuel type for the reactor. The effects should be appropriately included in transient analysis of an accident.
- ▶ Design consideration
 - ✓ Choice of fuel type, core height

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Number:2	Author:bois	Subject:Highlight	Date:26.08.2021	113343	+02'00'
This chapter is very specific for SFRs; it has to be re-evaluated for SCWRs.					

characteristics, considering all reactivity feedbacks, including sodium void worth, to mitigate the consequences of such design extension conditions."

(2) Design concept

Various core designs with different output power, different fuel material, i.e. oxide, metal and nitride. A common core geometry consists of fuel subassemblies containing the fuel pins in ducts of hexagonal cross-sections, typically referred to as subassembly ducts or hex-cans.

(3) Functional recommendations and Design consideration

(3-1) Limiting the total reactivity during unprotected transients

- ▶ Functional recommendations
 - Core reactivity characteristics should be designed so as to prevent prompt criticality, i.e. $\rho_{\text{net}} < 1\beta$ during the initiating phase of unprotected transients. Positive reactivity effects, such as sodium boiling, should be limited so that negative reactivity effects e.g. Doppler effect, fuel expansion and fission gas dispersion are sufficient to counteract the positive reactivity effects. Design parameters, such as sodium volume fraction, core height and other geometric parameters, should be properly chosen based on the effects of sodium void worth during transients.
- ▶ Design consideration
 - ✓ Core reactivity characteristics: sodium void reactivity, Doppler effect, fuel axial expansion, core radial expansion, control-rod drive-line expansion, etc.)
 - ✓ Sodium volume fraction and core height and geometry (e.g. heterogeneous arrangements, sodium plenum)

(3-2) Facilitating fuel reactivity effects

- ▶ Functional recommendations
 - Core design parameters, such as core height, should be properly chosen to obtain effective negative feedback due to fission gas dispersion. Fuel reactivity feedback is dependent on the choice of fuel type for the reactor. The effects should be appropriately included in transient analysis of an accident.
- ▶ Design consideration
 - ✓ Choice of fuel type, core height

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Number	Author	bois	Subject	Highlight	Date
Number:1	Author:bois	Subject:Highlight	Date:26.08.2021	113016	+02'00'
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Number:2	Author:bois	Subject:Highlight	Date:10.09.2021	184605	+02'00'
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Number:3	Author:bois	Subject:Highlight	Date:10.09.2021	184616	+02'00'
Not relevant					
Number:4	Author:bois	Subject:Highlight	Date:10.09.2021	184621	+02'00'
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► Design consideration

- ✓ Molten fuel relocation characteristics
- ✓ Molten fuel relocation path
- ✓ Coolant flow and re-entry paths to the core.

4 Prevention of Reactor Coolant Boundary Failures

(1) SDC
"Components, which constitute the reactor coolant boundary, shall be designed to maintain the boundary function and to maintain a sufficient sodium inventory in the primary coolant system in case of a core disruptive accident."

(2) Design concept
Prevention of reactor coolant boundary failures against mechanical load. (Countermeasures depend on the reactor structure design)

(3) Functional recommendations and Design consideration

(3-1) Prevention of reactor coolant boundary failures following mechanical loads

► Functional recommendations
The reactor coolant boundary should maintain its boundary function following pressure loads, including any loads induced by fuel-coolant interaction (FCI).

► Design consideration

- ✓ Choice of fuel type, inlet and/or outlet design of core assemblies, plenum geometry, structural design of reactor coolant boundary components, material strength.

(3-2) Prevention and mitigation of sodium ejection from a reactor cover gas boundary due to mechanical loads

► Functional recommendations
Reactor cover gas boundary components should withstand pressure loads to prevent sodium ejection into the containment, including any loads induced by FCI. Design measures, such as strengthened seals of plug structures, including rotating plugs for fuel handling, should be considered.
Mitigation provisions against sodium ejection should also be implemented as needed.

► Design consideration

- ✓ Structural design of reactor cover gas boundary components, material strength.

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(3-3) Prevention of cover pressure

► Functional recommendations
Measures should be provided to cope with temperature and pressure increases due to heat generation and accumulation of fission gases, released from a degraded core.

► Design consideration

- ✓ Installation of pressure relief devices, such as safety relief valves at the reactor cover gas boundary

4 Post Accident Heat Removal

(1) SDC
"Means shall be provided for the capability of core cooling under postulated plant conditions with core degradation."

(2) Design concept
Retention and cooling of a degraded core
Depending on reactor core characteristics, perform the retention function with an existing component or structure, or a dedicated structure, e.g. core catcher.

(3) Functional recommendations and Design consideration

(3-1) Retention of a degraded core

► Functional recommendations
Measures should be provided to retain degraded core materials to facilitate post-accident heat removal. Re-criticality of a retained degraded core should be prevented during the post-accident heat removal phase. The retention structure should resist the thermal load from a degraded core, as well as mechanical loads, including any loads from FCI.

► Design consideration

- ✓ Design of any structure intended to retain degraded core materials, including the core inlet plenum, core support structure (form, strength), material strength characteristics;
- ✓ Coolability and re-criticality of degraded core materials;
- ✓ Fuel retention capacities for such structures;
- ✓ Characteristics and relocation of degraded core materials.

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5.2.1 Prevention of Core Uncovering

(1) SDC
"Guard vessels and guard pipes shall be designed so as to maintain the sodium surface of the primary coolant system at a level necessary for decay heat removal in the case of a sodium leak accident in the primary coolant system. Due considerations shall be taken of a dependent failure and a common cause failure between the reactor vessel and the guard vessel, as well as between main coolant pipes and guard pipes. Provisions shall be made to reduce the amount of sodium that leaks from the primary coolant system in case of a failure of the reactor coolant boundary."

(2) Design concept
 Guard vessel,
 Guard pipes (option for loop type)

(3) Functional recommendations and Design consideration
5.2.1.1 Guard vessel

- ▶ Functional recommendations
 A Guard Vessel (GV) should be installed to enclose the Reactor Vessel (RV) so that the reactor sodium level can be maintained in the RV following a sodium leak. The gap volume between the RV and the GV should be limited, so that the sodium surface level inside the RV should, during a safe shutdown state, always be above the design limit level for sodium circulation (E.S.L., emergency sodium level).
- ▶ Design consideration
 - ✓ Gap/volume between the reactor coolant boundary and the GV (guard pipe as loop type option)
 - ✓ E.S.L. in RV

5.2.1.2 Prevention of double failure of the RV and the GV

- ▶ Functional recommendations
 In order to substantiate the practical elimination of core uncovering, the conditions for preventing double failure of the RV and the GV are given as follows.
Ensure the reliability
 The RVs and GVs should be designed, manufactured, installed, maintained and inspected to have the highest level of reliability.

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prevent core damage under severe leak conditions, such as multiple leaks in the primary loops, decay heat removal measures should be provided, e.g. enabling an in-vessel cooler to be operable in case of low sodium levels without coolant circulation in the primary coolant loops.

- ▶ Design consideration
 - ✓ Arrangement of the primary coolant system (e.g. vertical position of the horizontal parts of primary pipes), sodium quantity

5.2.2 Decay Heat Removal for DBA

(1) SDC
*"The decay heat removal system shall be designed as follows:
 (a) To provide diversity to the extent practicable and redundancy for reducing common cause failures, including external events.
 (b) To prevent freezing of the sodium coolant to avoid blockage of coolant circulation, and
 (c) To provide detection and mitigation measures against postulated decay heat fluid leaks."*

(2) Design concept
 Various options for system configuration, number of sub-systems (DRACS, FRACS, IRACS, RVACS, SGAHRS etc.) as safety systems. A secondary sodium loop connected to an air cooler is a typical DBA countermeasure.
 DRACS, Direct Reactor Auxiliary Cooling System
 FRACS, Primary Reactor Auxiliary Cooling System
 IRACS, Intermediate Reactor Auxiliary Cooling System
 RVACS, Reactor Vessel Auxiliary Cooling System
 SGAHRS, Steam Generator Auxiliary Heat Removal System

(3) Functional recommendations and Design consideration
5.2.2.1 Basic capability of a DHRS

- ▶ Functional recommendations
 A Decay Heat Removal System (DHRS) should be able to cool the reactor core immediately after reactor shutdown and for as long as needed time. The system configuration and heat removal capacity of DHRSs, as well as transient characteristics,

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such as flow coastdown of the primary pumps, should be set not to exceed design limits of AOOs and DBAs, as discussed in Section 4.3, assuming a single failure

- ▶ Design consideration
 - ✓ Heat removal capacity of DBA measures
 - ✓ Decay heat characteristics
 - ✓ Pump coastdown
 - ✓ Primary coolant system heat capacity

(3-2) Ensuring reliability of a DHRS

- ▶ Functional recommendations

In order to avoid a common cause failure in case of DBAs, including internal and external hazards, redundancy, diversity, physical separation and independence should be adequately provided. DHRSs should be designed to have redundancy and diversity for ensuring sufficient core cooling capacity against AOOs and DBAs, considering loss of off-site power and single failure of components. For instance, three systems at 100% of the capacity required for decay heat removal, or four systems at 50% capacity, would be provided, considering one system failure as an initiating event and a single failure in another system, depending on the design.

Diversity in the system configuration and/or operation mode (forced circulation and natural circulation), as well as physical separation, should be introduced.
- ▶ Design consideration
 - ✓ Redundancy (number of subsystems and their heat removal capacity)
 - ✓ Diversity (system level, component level, mechanism level (i.e., forced/natural circulation))
 - ✓ Layout, physical separation and independence (protection against internal and external hazards)
 - ✓ Emergency power supply system, if required (the capacity and the activation time)

B-5) Prevention of coolant freezing

- ▶ Functional recommendations

In order to prevent sodium (or other types of coolant such as NaK) freezing in DHRSs, design measures, such as keeping a minimum flow rate, providing an electric heater, or hot gas blow heating, should be provided. To prevent freezing in the air coolers, design measures should be provided, e.g. keeping sufficient circulation in the sodium circuits and

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independent control of subsystems for prevention of common cause failures due to malfunction of the air flow regulators and blowers.

- ▶ Design consideration
 - ✓ Operating temperature (standby temperature)
 - ✓ Heat exchange characteristics of air cooler (control characteristics)

B-4) Measures against sodium leaks

- ▶ Functional recommendations

Design provisions should be made for the prevention and early detection of sodium (or other types of coolant such as NaK) leaks, e.g. aerosol and contact type detectors, and for mitigation of the effects of chemical reactions between sodium and air or water, e.g. guard pipes or enclosures, sodium drain systems.
- ▶ Design consideration
 - ✓ Design measures for prevention, detection, control, and mitigation

4.2.3 Decay Heat Removal for DEC

(1) SDC

"In design extension conditions, means for decay heat transfer shall be provided, in addition to a decay heat removal system for anticipated operational occurrence and design-basis accidents, with the conditions listed below

(a) *The cooling of the reactor core is possible even under extreme external hazards and their consequences, such as long-term loss of all AC power supplies,*

(b) *Passive mechanisms are used to the extent practicable, and*

(c) *Decay heat removal system has diversity to the extent practicable."*

(2) Design concept

Functional extension of the design measures dedicated to AOOs and DBAs to cope with DEC.

Providing alternative cooling measures, in addition to the measures for DBAs

(3) Functional recommendations and Design consideration

(3-1) Enhancement of decay heat removal capabilities

- ▶ Functional recommendations

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Multiple barriers against the release of radioactivity consist of fuel elements, fuel cladding, reactor coolant and **Over gas boundaries, Guard vessels (and pipes in a loop configuration),** and containments. These functions need to be maintained for AOCs. For DBAs, the reactor core geometry is preserved and should be coolable, and the reactor coolant boundary, **Over gas boundary,** and containment should uphold their barrier functions.

Typical indices for design limits of the reactor core are the maximum temperatures of the fuel, cladding, and coolant. **Boiling should be prevented** to maintain the cooling of the reactor core, and to **limit its contribution to the net reactivity.**

Cladding failure during normal operation and AOCs should be avoided and specified acceptable fuel design limits are not exceeded. Cladding time-temperature during AOCs and DBAs, cladding stress and strain due to internal pressure are used (e.g. to evaluate the cladding cumulative damage).

The integrity of the reactor coolant boundary should be maintained during a postulated accident. Typical indices are the maximum coolant temperature and the duration of the accident when the reactor coolant boundary is exposed to elevated temperature.

Typical indices for the containment are the containment temperature and internal pressure.

Concerning release of radioactive materials, the potential public radiation exposure dose should be limited according to the criteria for operational states and accident conditions.

4.3.2. EBC

Core damage prevention under DECc

(1) Typical initiating events

Typical initiating events include ATWS that require other means of controlling or reducing the reactor power, such as passive shutdown or provisions for favourable inherent reactivity feedback. Postulated ATWSs, i.e., AOCs associated with failures of active reactor shutdown systems, comprise: Loss of Flow (LOF), e.g. loss of power in all the primary pumps, Transient Over Power (TOP), e.g. uncontrolled withdrawal of a control rod, and Loss of Heat Sink (LOHS), e.g. loss of main heat removal system typically via the secondary coolant system or the water-steam system.

Typical events, challenging the "reactor core heat removal" safety function, are multiple failures in the coolant system or multiple failures of the decay heat removal function, which might result in a reduction of the reactor coolant level or loss of the safety systems for decay heat removal. For these events, core degradation, due to core uncovering or complete loss of decay heat removal capabilities, should be practically eliminated by design measures.

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Number:1	Author:bois	Subject:Highlight	Date:10.09.2021 19:47:11 +02'00'
Not relevant			
Number:2	Author:bois	Subject:Highlight	Date:10.09.2021 19:47:18 +02'00'
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Number:3	Author:bois	Subject:Highlight	Date:10.09.2021 19:47:27 +02'00'
Not relevant			
Number:4	Author:bois	Subject:Highlight	Date:26.06.2021 10:38:56 +02'00'
For SCWR, this requirement should be formulated for film boiling (boiling crisis)			
Number:5	Author:bois	Subject:Highlight	Date:26.06.2021 10:42:46 +02'00'
Not relevant for water cooled reactors.			

(2) Design limits

As for the reactor core, core coolable geometry should be maintained by averting or limiting possible causes of cooling deficiencies due to coolant boiling, cladding failure or deformation (e.g. ballooning). **Fuel melting should be averted or limited to prevent propagation of fuel failures.**

As for the coolant boundary, creep failure caused by a temperature increase should be prevented. Hence, temperature increase and duration of the events should be adequately limited.

Containment failures should be avoided, i.e. temperature and pressure limits should not be exceeded.

Any release of radioactive materials from the containment should be limited and should not exceed acceptable regulatory dose limits to achieve "elimination of the need for off-site emergency response" for these events [2]. It should be noted that an off-site emergency plan is still envisioned for Gen-IV SFRs, even if the goal is elimination of the need for off-site emergency response.

Mitigation of consequences of core damage under DECc

(1) Typical initiating events

Unprotected transients resulting in core degradation, are representative of DECc with core melt. If inherent power reduction capabilities are insufficient, core degradation can occur, and analyses should include considering the uncertainty of individual reactivity feedback effects.

The potential for propagation of damage from local fuel failures should be considered as a core degradation mode and different from that of core damage from unprotected transients.

Provisions for achieving IVR should mitigate challenges to the containment structure from either type of core degradation modes.

For most SFRs, the containment design basis include a fire from sodium leakage and combustion. The containment function should be able to withstand pressures, temperatures, chemical reactions, aerosol deposition etc. which arise as a result of such a sodium fire.

(2) Design limits

Severe mechanical energy release, resulting in a reactor coolant boundary failure, should be prevented for unprotected transients that may lead to core degradation. Possible indices are:

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Number:1	Author:bois	Subject:Highlight	Date:29.08.2021 11:46:02 +02'00'
This requirement should be limited to DEC1 events.			
Number:2	Author:bois	Subject:Highlight	Date:29.08.2021 11:46:06 +02'00'
Not relevant for SCWR.			

the limitation of the fuel temperature or the prevention of prompt criticality, i.e. maximum net reactivity should be less than 1\$.

To prevent coolant boundary failures, the strain on the RV should be limited so that the failure won't occur during a dynamic load. In addition, creep damage, due to thermal stress at an elevated temperature, should be prevented.

Containment failures should be avoided, i.e. the temperature and pressure should not exceed specified limits, and dynamic effects should not challenge the containment boundary. Hydrogen generation should be prevented as far as possible, and its concentration should be limited to avoid reaching deflagration and detonation limits.

Release of radioactive materials should not exceed the regulatory limits for the public radiation exposure dose that initiates off-site response.

4.4. Testability

The safety functions of SSCs mentioned in 4.1 and 4.2 should regularly be inspected and tested throughout the plant's lifetime **according to their safety classifications**. Examples of tests and inspections to be performed during the plant's lifetime are given below.

Particular attention should be paid to the following aspects:

- ▶ integrity of the barriers between radioactive materials and the environment (such as fuel cladding, the reactor coolant boundary and the containment),
- ▶ availability of safety systems, such as protection systems, safety actuation systems and safety system support features,
- ▶ availability of items, whose failure could adversely affect safety,
- ▶ **accessibility and operability for monitoring and inspection of components under submerged in sodium coolant.**

(1) Reactivity feedback

- ▶ Measurement of reactivity feedback during start-up to verify both sign and magnitude for normal operation and accidents.

(2) Reactor shutdown

- ▶ Functional tests of reactor shutdown systems e.g. control rod positions and insertion times to ensure that the tested system or component is capable of performing its design function.

(3) Decay heat removal

- ▶ Functional tests of active components, such as pumps and blowers.

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Number: 1	Author: bots	Subject: Highlight	Date: 29.08.2021 11:53:56 +02:00
Rules for safety classification of SSCs should be described (in general) in the guideline.			
Number: 2	Author: bots	Subject: Highlight	Date: 29.08.2021 11:53:41 +02:00
Not relevant for SCWR			

- ▶ Confirmation of natural circulation heat removal capabilities, for instance, by checking temperature and flow rate changes from a safe shutdown state in case of termination of forced convection. This could be done as a performance test in the commissioning stage or pre-start-up phase in service.

4.5. Demonstration

The effectiveness of the safety functions of SSCs, mentioned in 4.1 and 4.2, should be demonstrated according to their safety classification before starting any power operation of a nuclear power plant. Examples of information required for the safety demonstration are given in this section.

For DBAs and DEC's without core damage, experiments, using scale models of the actual SSCs or actual reactor tests during commissioning stage, can be possible and useful for the safety demonstration.

For DEC's with core damage, direct performance demonstration on the actual reactor is impossible. Thus, numerical simulations have an important role. In addition, experiments should be set up to validate the analytical tools and to understand the basic phenomena having a bearing on the accident analysis.

1) Passive shutdown mechanism (example for SASS)

- ▶ Material tests: understanding physical properties of sensing alloys (Curie point temperature, thermal aging and irradiation effect), productivity.
- ▶ Component tests: control rod insertion performance by model tests, transient response tests, thermal hydraulic tests at the reactor outlet by scale models.
- ▶ Reactor tests: stability of the in-vessel holding force, operability and material tests.
- ▶ Development of analytical methods for design and evaluation.
- ▶ Validation of analytical methods based on above-mentioned experiments.
- ▶ Evaluation of the core damage frequency from reactor shutdown failures using PSA.

(2) Decay heat removal by natural circulation

- ▶ Validation of analytical models by water and sodium tests and applying the results to the evaluation of the reactor core.
- ▶ The performance of the actual plant should be confirmed during the performance test phase.
- ▶ Evaluation of the core damage frequency from loss of heat removal using PSA.

(3) Mitigation of core damage

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The exact guidelines are not relevant for SCWR; but the general requirements are applicable.			

4 CONSIDERATIONS FOR SFR REACTIVITY CHARACTERISTICS

In the course of the international reviews of the "SDC Phase I Report", a request was made for a technical clarification of the "sodium void reactivity in the fast reactor core reactivity characteristics" to be included in the SDG. In general, the idea and design concept of a so-called "negative sodium void reactivity reactor core" has been proposed and examined from many angles during the some half-century long SFR development history.

This chapter summarises the general reactivity characteristics of an SFR and contains concluding remarks on the sodium void reactivity in relation to SFR safety, emphasising that it is the overall reactivity feedback that is important and not any single part, such as sodium void. In addition, integrated transient analysis is required to determine if the sodium void of the design is acceptable or not since the entire core typically does not void during a postulated transient, but sodium voiding is usually predicted to initially occur locally with a much lower reactivity effect. The core reactivity can be dominated by reactivity change induced by motion of molten core materials such as fuel and steel in a CDA (Core Disruptive Accident). Positive sodium void reactivity should be appropriately limited but the sodium void is not necessarily negative.

General view

Regardless of the LWR or the fast reactor including the SFR, the reactor core should have the inherent reactivity feedback characteristics so as to achieve stable control in operational states and terminate an abnormal event in an accident condition without resulting in core damage. In addition to the inherent reactivity feedback characteristics, the nuclear facility should have an instrumentation and control system that ensures stable control of the reactor in operational states, and safety systems such as a reactor shutdown system against a DDA to prevent the events from exceeding the design limits. Furthermore, the facility should utilise the inherent reactivity feedback, and also a passive mechanism complementary to the function of the feedback if needed, as a preventive measure against core damage under a DEC.

A fast reactor core has a critical geometry for which moderation of fast neutrons generated by nuclear fission is not required in principle. The core reactivity is affected by the change in temperature of or in distribution of coolant and structural materials, which have the capability to moderate neutron, and fuel itself. Therefore, the safety design should prevent excessive power increase caused by re-criticality in a CDA.

The core of a commercial power-generating fast reactor generally has regions of positive void reactivity (interior) and negative void reactivity (the periphery), sodium voiding hardens the neutron spectrum, leading to positive void reactivity in the interior, while it facilitates neutron leakage, leading to negative void reactivity in the periphery. In a fast reactor core except very

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Number: 1 Author: bots Subject: Highlight Date: 29.08.2021 12:48:50 -0200
The chapter deals with a special SFR problem (sodium void coefficient), not relevant for SCWR. However, similar specific issues for SCWR-type reactors should be evaluated here.