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Abstract:

This deliverable outlines a pre-licensing study for the demonstration of the safety in the further development stages of the Supercritical Water-cooled Small Modular Reactor (SCW-SMR). First, generic and specific safety criteria and requirements for the SCW-SMR concept have been developed by following and further developing the science-based recommendations by the International Atomic Energy Agency (IAEA), Generation IV International Forum (GIF) for SMRs and other relevant guidance. The requirements and criteria from the relevant documents have been related to the challenges, issues and phenomena gaps in knowledge regarding the safety-related behaviour of SCW-SMR. These documents are e.g. IAEA safety principles and standard for design, GIF safety goals and Western European Nuclear Regulators Association (WENRA) objectives and key positions for safety of new Nuclear Power Plant (NPP) designs. First set of safety-related features have been derived from literature review of Supercritical Water-cooled Reactor (SCWR), High Performance Light Water Reactor (HPLWR), Boiling Water Reactor (BWR) and SMR technology. Then issues to safety-related features and phenomena based on expert opinions and the project results from materials testing, thermal hydraulic and safety, and neutron physics research on SCW-SMR have been identified. The expert opinions about the phenomena have been collected in a Phenomena Identification and Ranking Table (PIRT) analysis, which was also used to underline safety aspects needed in future detailed investigations. The generic results of the SCW-SMR pre-licensing study identified three sets of the safety criteria and requirements: (1) fulfilled with reasonable certainty

within the limits of already available knowledge; (2) fulfilment is probable, further research needed; (3) fulfilment is improbable, design changes are needed.

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Executive Summary

The purpose of the pre-licensing study was to relate the safety criteria and requirements for the Supercritical Water Small Modular Reactor (SCW-SMR) concept developed in WP5.1 [4] with the phenomena gaps in knowledge regarding the safety-related behaviour of SCW-SMR as identified in WP5.2 [6] and to the available level of detail. No specific experimental verifications have been planned for this pre-licensing study.

Section 1 introduces the project's objectives and methodology. The pre-licensing study, a critical component of WP5.3, connects safety criteria and requirements for the SCW-SMR with the challenges, issues, and knowledge gaps identified in WP5.1 and WP5.2, respectively. The study categorized the safety criteria into three groups: those that can be met with existing knowledge, those that likely can be met with further research, and those that will require design changes. This classification aims to address specific knowledge gaps, particularly regarding structural materials (of the reactor technology as well as the fuel cladding), radiation effects, validation of engineering simulation tools, core design, and the licensing process itself. As future SCW-SMR conceptual design is in the scope of follow-up projects, an example of the conceptual design of European High-Performance Light Water Reactor (HPLWR) developed in the frame of EU project HPLWR PHASE 2 (grant agreement 36230) is provided. The HPLWR, a fourth-generation nuclear reactor concept, is cooled and moderated with supercritical water. The conceptual design serves as a reference to guide future development and identify critical areas for further research in SCW-SMR technology.

Section 2 of the report discusses the pre-licensing study approach for the SCW-SMR technology. The section outlines the primary inputs for the study, D5.1 [4] and D5.2 [6] deliverables. D5.1 [4] deliverable concludes that no specific safety elements have been developed for the SCW-SMR, and no regulations for SMRs, except in Canada and China, are in force. Specific requests for SCW-SMR are also absent in these regulations. Thus, safety criteria must be developed alongside ECC-SMART's design features. Four guiding principles were developed, emphasizing compatibility with new reactor safety elements, case-by-case application of SMR-specific elements, adherence to Generation IV International Forum (GIF) safety requirements, and the need for robust safety demonstrations. ECC-SMART's safety principles should integrate 3rd Gen., SMR, and 4th Gen. standards, particularly using IAEA SSR-2/1 (Rev. 1) as a design basis. D5.2 [6] deliverable outlines the "safety-related features of the SCW-SMR design" by evaluating literature, technical work packages, and expert opinions. It highlights challenges from Supercritical Water-cooled Reactor (SCWR), HPLWR, and Boiling Water Reactor (BWR) technologies, as well as SMR-specific issues. Key findings include identified phenomena with high importance and knowledge gaps. The PIRT analysis, which evaluates phenomena based on importance and knowledge levels, indicates which areas require further research to enhance the safety of SCW-SMR technology. The scope of the pre-licensing study for SCW-SMR is also outlined. It is to relate to the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 with the issues and gaps in knowledge regarding the safety-related behaviour of SCW-SMR as identified in D5.2. This is in agreement with the methodology defined in the ECC-SMART grant agreement [9] which states that the pre-licensing study will summarize both the already achieved safety criteria and the potential gaps to be further investigated in future research projects.

Section 3 outlines the conformance of the future SCW-SMR technology to two relevant IAEA standards, the Fundamental Safety Principles SF-1 [26] and SSR-2/1 (Rev. 1) [1] for design. The ECC-SMART project does not cover SCW-SMR's detailed design; nevertheless, this report

assesses conformance to SF-1 Principle 3 (leadership and safety assessment) and SF-1 Principle 8 (accident prevention). Then, the description of IAEA SSR-2/1 (Rev. 1) [1] standard for design is provided, selection process of relevant requirements of IAEA SSR-2/1 (Rev. 1) [1], applicability of selected requirements of IAEA SSR-2/1 (Rev. 1) [1] (i.e. Requirements 42 through 58) to SCWR, safety considerations and options to enhance the performance of the engineered safety features of water-cooled SMRs incorporating lessons learned from the Fukushima Daiichi accident [48]. Finally, the requested information needed for future SCW-SMR conceptual design compliance to selected requirements of SSR-2/1 (Rev. 1) [1], relation of each selected requirement with safety-related behaviour of future SCW-SMR conceptual design, and compliance of knowledge is provided.

Section 4 of the report discusses the conformances to 23 fuel safety criteria by Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) [10]. For each fuel criterion, a brief description is given first, followed by the requested information needed for future SCW-SMR conceptual design compliance judgement, the relation of each selected criterion with safety-related behaviour of future SCW-SMR conceptual design, and compliance of knowledge.

Section 5 of the report examines the alignment of future SCW-SMR technology with Generation IV (Gen IV) nuclear energy system goals. The Gen IV goals serve multiple purposes: they establish criteria for technology assessment, drive innovation in nuclear energy systems, and guide collaborative research and development (R&D) efforts. Among the eight established goals, three focus on safety and reliability: ensuring operational excellence, minimizing reactor core damage likelihood and severity, and eliminating the need for offsite emergency response. As information provided in D5.2 [6] is very general (based on a literature review) for SCWR, HPLWR, BWR and SMRs, and very little detailed information exists for future SCW-SMR conceptual design, the applicability of these GIF goals to future SCW-SMR conceptual design is discussed rather judgment of design compliance to requirements and criteria is done.

Section 7 of the report summarizes the outcomes of the pre-licensing study conducted for the future SCW-SMR conceptual design. Pre-licensing study results for IAEA standards requirements and criteria, and OECD/NEA fuel safety criteria fulfilment are shown. For Gen IV goals and Western European Nuclear Regulators Association (WENRA) safety objectives for SMRs the applicability to SCW-SMR is discussed. Relevant IAEA SSR-2/1 (Rev. 1) [1] requirements and criteria are related with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR and knowledge judgement is done regarding further research and design refinement to achieve full compliance. In addition, the pre-licensing study results list the recommendations of IAEA specific safety guides SSG-2 [17], SSG-52 [16], SSG-53 [37] and SSG-56 [36], based on which requested information needed to judge the compliance of future SCW-SMR conceptual design to the selected safety criteria and requirements of IAEA SSR-2/1 (Rev. 1) is determined. Relevant OECD/NEA fuel safety criteria are related to challenges, issues and phenomena of safety-related behaviour of SCW-SMR and knowledge judgement is done regarding further research and design refinement to achieve full compliance. As little detailed design information is available about future SCW-SMR conceptual design, it is difficult to examine the level of support for the conformance to safety goals. Therefore, the applicability of safety goals to future SCW-SMR conceptual design has been discussed. For WENRA safety objectives discussion on features of SMRs that differ from the present-day reactors is judged with respect to applicability of objectives to future SCW-SMR conceptual design.

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List of acronyms and abbreviations

AOO	Anticipated Operational Occurrences
ABWR	Advanced Boiling Water Reactor
BWR	Boiling Water Reactor
CABRI	a pool-type research reactor
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CPR	Critical Power Ratio
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design Basis Accident
DiD	Defence-in-Depth
DNB	Departure from Nucleate Boiling
DSA	Deterministic Safety Analysis
EAC	Environmentally Assisted Cracking
ECC-SMART	(Joint) European Canadian Chinese development of Small Modular Reactor Technology
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
DEC	Design Extension Conditions
GIF	Generation IV International Forum
FGR	Fission Gas Release
HPLWR	High Performance Light Water Reactor
IAEA	International Atomic Energy Agency
IL	Importance Level
IL _w	Importance Level - weighted
ISAM	Integrated Safety Assessment Methodology
KL	Knowledge Level
KL _w	Knowledge Level - weighted
KR	Knowledge Ranking
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCST	Maximum Cladding Surface Temperature
MOX	mixed-oxide
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
OECD/NEA	Organisation for Economic Co-operation and Development/Nuclear Energy Agency
PCI	Pellet-Clad Interaction
PCMI	Pellet-Clad Mechanical Interaction
PIE	Postulated Initiating Events
PIRT	Phenomena Identification and Ranking Table
PR	Phenomena Ranking
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
R&D	Research and Development

RCS	Reactor Coolant System
RD	Relative Dispersion
RIA	Reactivity Initiated Accident
RPV	Reactor Pressure Vessel
RR	Relative Relevance
SCC	Stress Corrosion Cracking
SCC-PCI	Stress Corrosion Cracking – Pellet Cladding Interaction
SCW	Supercritical Water
SCWR	Supercritical Water-cooled Reactor
SCW-SMR	Supercritical Water-cooled Small Modular Reactor
SDC	Safety Design Criteria
SFR	Sodium cooled Fast Reactor
SMR	Small Modular Reactor
SSC	Structures, Systems and Components
SSC SDG	Safety Design Guidelines on Structures, Systems and Components
SSE	Safe-Shutdown Earthquake
SSG	Specific Safety Guide
SSR	Specific Safety Requirements
STUK	Säteilyturvakeskus (Finland's Radiation and Nuclear Safety Authority)
U.S.	United States
WENRA	Western European Nuclear Regulators Association
WGFS	Working Group on Fuel Safety
WNA	World Nuclear Association

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N.A.

Related documents

- [Deliverable D5.1](#) Safety criteria and requirements for the SCW-SMR concept, JSI (leader), 31/08/2021,
- [Deliverable D5.2](#) Safety related features of the SCW-SMR concept, BME (leader), 02/02/2024.

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). The main objectives of the Work Package 5 (WP5) are to develop generic and specific safety criteria and requirements for the SCW-SMR concept (see D5.1 [4]), to synthesize the main safety-related findings and conclusions of WPs 2-4 (see D5.2 [6]), develop a pre-licensing study demonstrating the feasibility of the design to be licensed (purpose of this report) and to develop guidelines for the demonstration of safety in the further development stages of the SCW-SMR concept (in the scope of D5.4).

1.2 Purpose of the document

This document is the third among the four deliverables to be developed within the WP5 Synthesis and guidelines for safety standards of the ECC-SMART project. It documents the work performed in WP5 Task 5.3, which relate the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 with the challenges, issues and gaps in knowledge regarding the safety-related behaviour of SCW-SMR as identified in WP5.2 and to the available level of detail. No specific experimental verifications are planned for this pre-licensing study. The information collected and developed in the following two deliverables was used as input:

- Deliverable D5.1 Safety criteria and requirements for the SCW-SMR concept;
- Deliverable D5.2 Safety-related features of the SCW-SMR concept.

A pre-licensing study identifies three sets of safety criteria and requirements:

- (1) Fulfilled with reasonable certainty within the limits of already available knowledge – in the following text FULFILLED,
- (2) Fulfilment is probable, further research is needed – in the following text FULFILMENT IS PROBABLE, and
- (3) Fulfilment is improbable, design changes are needed – in the following text FULFILMENT IS IMPROBABLE.

Such judgment is limited to the challenges, issues and gaps in knowledge regarding the safety-related behaviour of future SCW-SMR conceptual design as identified in WP5.2. Specific knowledge gaps are primarily the assessment of structural materials with special attention to the influence of radiation, validation of engineering simulation tools like system, subchannel and computational fluid dynamics (CFD) codes, core design as well as licensing process itself.

For selected requirements and criteria related to safety-related behaviour of future SCW-SMR conceptual design for which information has not been provided as input through D5.2 [6], requested information needed to judge compliance of design will be specified.

If some criterion or requirement is not applicable to future SCW-SMR conceptual design or no information is available for SCW-SMR conceptual design, judgement on compliance will not be done - in the following text NOT JUDGED.

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

- D5.4 Guidelines for the demonstration of the safety in the further development stages of the SCW-SMR concept.

1.3 Design information needed for pre-licensing review

As we are considering requirements and criteria for future SCW-SMR conceptual design, which is in the scope of the follow-up project, an example of SCWR conceptual design is provided, illustrating the level of design information available for reviewer. Namely, according to WENRA Reactor Harmonization Working Group (RHWG) the following is stated [42]: "*As little detailed design information is available about many of the innovative designs that are being considered or proposed, it is difficult to examine the level of support for the claims of enhanced safety. However, the purpose of this report is not to evaluate the proposed concepts or the validity of the safety claims but to evaluate the applicability of the WENRA Safety Objectives to SMRs.*" From this example it can be seen that even for existing innovative designs, which are more mature than conceptual design, it is difficult to judge compliance of design with safety requirements. In our opinion as a bare minimum a conceptual design is needed to judge such compliance, therefore an example of conceptual design is presented and compared against ECC-SMART pre-conceptual studies.

The European HPLWR PHASE 2 project [19] can serve as an example of such innovative design (another could be the Japanese SCWR concept [5]). The passive safety features and the material requirements still remained to be challenging for the HPLWR. Based on the common research activities within GIF, a few knowledge gaps were defined. In ECC-SMART project, open issues which were defined by project partners are to be addressed in particular, materials, water chemistry, heat transfer, neutronics, development and validation of codes. Special attention is to be paid to the topic of passive safety of the system, especially within the scope of the activities related to thermal-hydraulics and safety. The main safety-related findings and conclusions in other work packages were planned to facilitate the development of the pre-licensing study, which would summarize both the already fulfilled safety criteria and the potential gaps to be further investigated in future research projects. The pre-licensing study would benefit a lot if passive safety features were already proposed and their performance tested by deterministic safety analyses. For example, a pre-licensing review per Canadian REGDOC 3.5.4 [2], commonly referred to as a vendor design review, is an optional service that the Canadian Nuclear Safety Commission (CNSC) provides for the assessment of a vendor's reactor design. A vendor design review consists of three phases. *Phase 1* review intends to determine, at the general level, the vendor design and design processes demonstrate implementation of Canadian design requirements. *Phase 2* is prelicensing assessment, with a focus on identifying if any potential fundamental barriers to licensing exist or are emerging with respect to the reactor's design. *Phase 3* is pre-construction follow-up. *Phase 3* should not be initiated by a vendor until the design's detailed engineering program is underway [2]. A vendor design review can begin once a vendor has, at a minimum, made reasonable progress in the basic engineering phase of the design. This means that the basic architecture of systems important to safety has been laid out following the vendor's reactor design guides and design requirements. The following documents should be approaching a state of completion, such that the vendor is ready to proceed with the detailed design phase [2]:

- *"Design guides that contain design philosophies, safety philosophies and rules that designers must follow when performing their design work, including safety requirements such as applicable codes and standards"*
- *Design requirements for systems important to safety that establish such aspects as minimum performance requirements and reliability targets reflect significant progress made in any safety-related research and development*
- *The vendor's overall management system as it applies to the design of structures, systems and components (SSC)*
- *Design and safety analysis representative of a preliminary safety analysis report"*

A vendor can initiate a *Phase 1* review once the conceptual design is complete. However, the conceptual design of SCW-SMR itself is not the objective of the ECC-SMART project. Therefore, the pre-licensing study¹ could not consider as input documentation design and safety representative of a preliminary safety analysis report. Also, the basic architecture of systems important to safety has not yet been laid out following the reactor design guides and design requirements.

Also, D3.3 entitled "Report on the pre-conceptual studies on the core layout and passive safety concept of the SCW-SMR" [7] and D3.6 [8] entitled "Report on pre-conceptual design requirements for ECC SCW-SMR" shall serve as a basis for requirements of a future SCW-SMR conceptual design project, were not input document to D5.2 [6] as they were not available at the time of D5.2 [6] publication. This means that very limited information on the safety features of the SCW-SMR concept is available for the pre-licensing study. On the other hand, safety requirements and criteria have been elaborated in D5.1 [4] entitled "Safety criteria and requirements for the SCW-SMR concept". From the title of the report it is clear that the identified criteria are for the future SCW-SMR concept. For example, the Canadian pre-licensing review described in REGDOC 3.5.4 [2] requires to meet requirements in REGDOC 2.5.2 [3], which to a large degree, represents the CNSC's adoption of the principles set forth by the International Atomic Energy Agency (IAEA) in SSR-2/1 [1] standard for design.

1.3.1 HPLWR as an example of conceptual design

To illustrate what is conceptual design, the reader can refer to the book entitled "The High Performance Light Water Reactor" [19], dealing with HPLWR Phase 2. After the first HPLWR in 2003 [18], HPLWR Phase 2 was planned in which the HPLWR reactor and power plant should be designed in more detail to assess the future potential, based on ideas and concepts of the first phase. HPLWR Phase 2 is a nuclear reactor concept of the 4th generation which is cooled and moderated with supercritical water. The concept has been worked out by a consortium of European industry, research centres and universities, co-funded by the European Commission. It features a once through steam cycle, a pressure vessel type reactor, and a compact containment with pressure suppression pool. The conceptual design described in [19] shall enable to assess its feasibility, its safety features and its economic potential.

Chapter 2 of [19] provides a comprehensive analysis of the HPLWR Phase 2 core design. The chapter begins by outlining the core's general design strategy, focusing on the thermal core design assembly design proposal, involving 40 fuel rods and a single water box. The coolant flow path is meticulously detailed, with illustrations indicating the feedwater and coolant trajectories

¹ 'pre-licensing study' is much smaller in the scope than 'pre-licensing review', which review the preliminary safety analysis report

through various heating stages: the evaporator, first superheater, and second superheater regions. The fuel assembly design incorporates wire-wrapped fuel rods and honeycomb structures for improved stability. Neutronic design considerations include iterative coupling of neutronic and thermal-hydraulic codes to predict power and coolant density profiles accurately. Thermal-hydraulic design addresses the heat-up steps and flow distribution within the core. Stability analyses reveal the necessity for inlet orifices in evaporator fuel assemblies to ensure operational stability against density wave oscillations.

Chapter 3 of [19] focuses on the design and analysis of the primary system of the HPLWR Phase 2. It outlines the system components, including the reactor pressure vessel (RPV), steam generators, recirculation pumps, and pressurizer. The design strategy emphasizes the necessity for a robust RPV to handle the high system pressure (25 MPa) and temperature (up to 500 °C). The chapter details the RPV's structure, including the core barrel, control rod guide tubes, and steam plenum, highlighting the materials used, such as 20 MnMoNi 55 steel and P91 for the outlet flange. Thermal expansion management and coolant mixing processes are described, stressing the importance of minimizing thermal stresses and ensuring uniform coolant flow. The safety and performance analyses include evaluations of the system's ability to handle steady-state conditions and transient scenarios. Additionally, the in-core instrumentation for monitoring neutron flux and power distribution is discussed, featuring systems like the Power Distribution Detector and aeroball probe measurement systems. These systems ensure accurate monitoring and control of reactor operations.

Chapter 4 of [19] provides a detailed analysis of the containment design and safety systems for the HPLWR Phase 2. It outlines the general strategy for safety, emphasizing both active and passive safety mechanisms. The containment is designed to withstand internal pressure increases up to 0.5 MPa, and external threats such as earthquakes and airplane crashes. The chapter details the mechanical analyses demonstrating the containment's robustness. Key safety systems include the reactor shutdown system, containment isolation valves, steam pressure relief, automatic depressurization, and coolant injection systems. Accident scenarios, such as loss of coolant accidents (LOCAs) and control rod ejection, have been analysed using coupled codes like APROS, ATHLET-KIKO3D, CATHARE, RELAP5 and SMABRE/TRAB-3D to evaluate system responses and ensure core cooling under various conditions. The chapter also discusses the limitations and uncertainties in the current safety analyses, highlighting the need for further studies and experimental data to validate numerical simulations, especially during the transition from supercritical to subcritical conditions.

Chapter 5 of [19] delves into the design and layout of the HPLWR's Phase 2 steam cycle and the overall power plant configuration. It starts with an overview of the design strategy, aiming for a net power output of 1000 MWe and a net plant efficiency of 43.5 %. The chapter discusses the thermodynamic options, highlighting the use of a high-pressure, medium-pressure, and low-pressure turbine train, with a reheater between the high-pressure and medium-pressure turbines. The schematic diagram and the temperature-entropy diagram illustrate the steam cycle's operation and efficiency optimization. The major components of the steam cycle include steam turbines, reheaters, feedwater pumps, feedwater tanks, and preheaters. The chapter details the performance and functionality of each component, ensuring efficient and reliable plant operation. Control strategies for the steam cycle during various load ranges and start-up/shut-down procedures are also covered. The layout of the power plant is based on the Advanced Boiling Water Reactor (ABWR) design, with specific modifications to accommodate the HPLWR's Phase 2 requirements. The reactor building, containment, safety systems, and auxiliary systems

are described, emphasizing their roles in ensuring plant safety and operational efficiency. The chapter concludes with a cost estimate and a discussion of the overall plant design.

1.3.2 ECC-SMART SCW-SMR feasibility assessment and identification of safety features

As stated in NEI 18-04 document [27] for no-LWR reactors, the design development is performed in phases and often includes a pre-conceptual, conceptual, preliminary, and final design phase and may include iterations within phases. Design development and analysis includes definition of the elements of the safety design approach, the design features to meet the top-level design requirements for energy production and investment protection, and analyses to develop sufficient understanding to perform a Probabilistic Risk Assessment (PRA) and the deterministic safety analyses. As was explained in Section 1.1, one of the objectives of ECC-SMART is to derive design requirements for an SCW-SMR design concept. This design requirement document shall serve as a basis for a future conceptual design project. The conceptual design of such a reactor itself is thus not the objective of the ECC-SMART project. ECC-SMART project is oriented towards assessing the feasibility and identification of safety features of an intrinsically and passively safe SCW-SMR, considering specific knowledge gaps related to the future licensing process – especially the assessment of the structural materials with special attention to the influence of irradiation, validation of engineering simulation tools like system, subchannel, and CFD codes, core design as well as the licensing process itself.

The objective of Work Package 2 (WP2) is to complete the existing knowledge and deepen the understanding of corrosion processes that occur in candidate materials for fuel cladding under both non-irradiated and irradiated conditions. It will also deepen the knowledge of not well-known processes such as radiolysis and electrochemistry in SCW, both decisive to define the chemistry in the reactor.

Material selection is critical for the SMR-SCW design. WP2 aims to verify and complete previous findings on materials such as Alloy 800H and Stainless Steel 310S. This verification will be extended through long-term immersion corrosion tests and Slow Strain Rate Tensile (SSRT) tests. These tests will be carried out under different tests conditions, encompassing corrosion processes around the critical point as well as under average operating conditions for this type of reactor. Additionally, the corrosion behavior of an advanced material, specifically an Alumina-Forming Alloy (AFA) based on 310S steel, will be also studied. The studies conducted in this work package will be complemented by investigations into the effects of neutron irradiation on the corrosion resistance of these candidate materials in supercritical water, as well as the study of radiolysis processes that would occur in supercritical water. The primary objective is to preliminarily assess whether these processes could be suppressed under the selected test conditions. If possible, this would confirm the viability of the SCW-SMR. The WP2 is divided into 4 sub-tasks which have the following objectives:

- WP 2.1: Selection and characterization of materials. The selection is based on available information.
- WP 2.2: Investigating corrosion and Environmentally Assisted Cracking (EAC) behaviour of non-irradiated materials under SMR-SCWR selected environmental conditions.
- WP 2.3: Studying how neutron radiation affects material oxidation in SCW.

- WP 2.4: Exploring how changes in SCW chemistry influence material corrosion through specialized electrochemical measurements and to explore the possibility of suppressing radiolysis processes in supercritical water.

The objectives of WP3 address thermal-hydraulic investigations, safety considerations, and the effect of ongoing corrosion processes on heat transfer. The main objectives can be further specified as follows:

- The first objective involves system design analysis and the creation of a design requirement document based on preliminary SCW-SMR concepts developed under the Generation IV International Forum. This task aims to optimize the design specifications essential for future SCW-SMR conceptual design phases.
- The second objective of WP3 focuses on modelling surface-coolant interaction to predict corrosion under SCW conditions. Corrosion is a critical issue that impacts system longevity and safety, necessitating accurate prediction models validated against experimental data.
- The third objective comprises numerical and experimental investigations of thermal-hydraulic phenomena crucial for SCW-SMR safety and feasibility. This includes improving computational tools for scenarios like LOCAs and enhancing heat transfer models for CFD simulations.

Throughout WP3, efforts are also directed towards assembling and utilizing existing experimental data to validate models and simulations. This involves tasks such as direct numerical simulations of flow-induced corrosion, development of heat transfer correlations, and validation of engineering models for turbulent heat and mass transfer under SCW conditions.

WP4 focuses on the neutron physics of a SCW-SMR. This work package aims to assess design and safety aspects crucial for the pre-conceptual design phase of the SCW-SMR.

- In WP4.1, the effects of different neutron-physics codes and cross-section sets are evaluated through benchmarking and comparative analysis. This phase addresses the complexities of modelling a SCWR including 3D modelling of rapidly changing moderator properties and cross-section availability.
- WP4.2 delves into analytical investigations of neutron physics parameters pertinent to SCW-SMR safety and feasibility. This includes studying reactivity feedback coefficients, fuel and moderator properties under varying temperatures and pressures, and integrating thermal-hydraulic models to capture spatial coolant density variations. Additionally, the impact of corrosion on neutron physics and power distribution is explored.
- WP4.3 conducts preliminary core design calculations using Monte Carlo simulations to optimize reactor core layouts and dimensions. This phase assesses cycle lengths, power density distributions, and shutdown margins, crucial for determining operational feasibility and safety margins. Inputs from WP3 studies guide these analyses, ensuring alignment with overall project objectives and design requirements.

Table 1 shows the topics covered by ECC-SMART and HPLWR Phase 2 project. The major difference is that ECC-SMART does not propose the SCW-SMR conceptual design (see above WP2, WP3 and WP4 objectives), rather it is oriented towards assessing the feasibility and identification of safety features of an intrinsically and passively safe SCW-SMR. On the other hand, HPLWR Phase 2 started with the review of design proposal of HPLWR from 2003 [18]. Design tasks included core design, reactor design, safety systems and steam cycle components, and analyses covered neutronic analyses, thermal-hydraulic analyses, structural analyses and

system analyses for steady state and transient conditions. The technology program was focused on heat transfer of supercritical water in the reactor core, on materials for fuel claddings, and on the water chemistry needed. A first design proposal of a containment for the HPLWR Phase 2 had been also worked out within the first year [19].

Table 1. Topics covered by HPLWR Phase 2 and ECC-SMART

System	HPLWR Phase 2 project	ECC-SMART project	
2. Core design	Chapter 2 provides a detailed examination of the core's mechanical and thermal design elements, along with stability considerations essential for the efficient and safe operation of the HPLWR.	WP2 Materials Testing	WP2 objective is to enhance understanding of the corrosion behaviour of cladding materials intended for SMRs cooled by Supercritical Water (SCW), under both non-irradiated and irradiated conditions.
		WP4 Neutron physics of SCW-SMR	WP4 aims at studying the design- and safety-related neutronic parameters and reactor physics behaviour of the SCW-SMR in order to support the pre-conceptual design. It also contains preliminary core design calculations in order to obtain and evaluate some potential core layouts.
3. Primary System	Chapter 3 provides a detailed examination of the primary system's design considerations, materials, and safety mechanisms essential for the HPLWR's efficient and safe operation.	WP3 Thermal Hydraulics and Safety of the SCW-SMR	WP3 seeks to enhance the scientific foundation necessary for the design and safety assessment of SCW-SMRs. It aims to address key challenges such as corrosion, thermal-hydraulic behaviour, and safety margins, thereby contributing to the advancement of next-generation nuclear reactor technology.
4. Containment and Safety Systems	Chapter 4 underscores the importance of a multi-faceted safety approach combining robust design, thorough analysis, and ongoing research to address potential safety challenges in the HPLWR.		
5. Steam Cycle and Layout of the Power Plant	Chapter 5 describes the design and layout of the HPLWR's steam cycle and the overall power plant configuration.	N.A.	

2 Pre-licensing study approach

2.1 Inputs to pre-licensing study review

2.1.1 Input from D5.1 deliverable

Deliverable D5.1 [4] concludes that 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. Therefore, safety criteria covering particular aspects of the future ECC-SMART conceptual design will have to be developed in parallel with the specific design features featured by the designs under ECC-SMART.

The following four guiding principles have been developed, summarized in Table 2.

Table 2. Four guiding principles for the review and selection of the relevant safety elements

ID	Guiding Principle
1P	All compatible ² safety elements ³ of new reactors (3rd Gen. NPPs) apply to ECC-SMART.
2P	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.
3P	All the top-level safety requirements identified by GIF apply to ECC-SMART, in particular, the elimination of DEC-B scenarios.
4P	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.

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: SF-1 [26], SSR-2/1 (Rev. 1) [1], SSG-53 [37], SSG-56 [36], SSG-62 [49] and SSG-64 [50] classified as 1P, TECDOC-1936 [47] and TECDOC-1785 [48] classified as 2P and 3P. Especially, IAEA SSR-2/1 (Rev. 1) [1] should be used during the SCW-SMR development as a starting point for the design criteria development. Specific safety guides (SSGs) provide guidelines to fulfil the top-level requirements drawn from SSR-2/1 (Rev. 1) [1]. In this pre-licensing study SSG-2 [17], SSG-52 [16], SSG-53 [37], SSG-56 [36], TECDOC-1785 [48] and TECDOC-1936 [47] have been used, while SSG-62 [49] and SSG-64 [50] have not been used due to not

² Within the current context, 'compatible' is intended as applicable to ECC-SMART in terms of design and performance.

³ generic safety principles, objectives and criteria (hereafter simply named 'safety elements')

available information on hazards in D5.2 [6] and not available detailed information on future SCW-SMR conceptual design.

The SCWR is among the selected GIF designs. As such, ECC-SMART should meet the relevant GIF safety goals:

- The frequency of DEC-A⁴ scenarios shall be very low and its damage extension limited.
- DEC-B⁴ scenarios shall be eliminated.

The Safety Design Criteria (SDC) document [46] was developed for Sodium-Cooled fast reactors (SFR) 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 (Rev. 1) [1]) and guidelines can be applied to a Generation IV system through suitable adaptation of IAEA documents. A preliminary screening of the applicability of criteria developed in the SDC document [46] is given in Annex 1 and in Section 3.2.3 of D5.1 [4].

2.1.2 Input from D5.2 deliverable

D5.2 deliverable [6] has outlined the "safety related features of SCW-SMR design" based on the review of the available literature, the conclusions available from technical work packages (at month 42) and the expert opinion of WP2-4 participants.

General knowledge was collected from the literature review, which covered challenges of safety-related features derived from SCWR technology and BWR technology, and SMR special issues.

Project specific knowledge was collected from the evaluation of WP2-4 results in the deliverables and elaboration of other reports of the work packages. Answers to the potentially outstanding issues and gaps in knowledge regarding the safety-related behaviour of SCWR were obtained by expert opinion through "Safety-related findings and conclusions live Excel file" for WP2, WP3 and WP4.

In order to get a broader view about material-related, thermal hydraulics-related and neutronics-related safety features of SCW-SMR an expert evaluation (in the framework of a PIRT analyses) has been initiated for the project partners of WP2, WP3 and WP4.

Challenges and issues to safety-related features of future SCW-SMR and most significant phenomena with safety-related knowledge gaps from D5.2 [6] are:

- "Safety-related features derived from SCWR technology" – 6 challenges
- "Safety-related features derived from HPLWR technology" – 7 challenges
- "Safety-related features derived from BWR technology" – 4 challenges
- "SMR special issues" – 3 challenges
- "Safety-related features based on the results and expert opinion of WP2":
 - 4 important issues by expert opinion;
 - 23 phenomena were identified by WP2 in PIRT analysis of safety related knowledge gaps.
- "Safety-related features based on the results and expert opinion of WP3":

⁴ According to WENRA **Chyba! Nenalezen zdroj odkazů.** there are two categories of DEC:
– DEC A for which prevention of severe fuel damage in the core or in the spent fuel storage can be achieved;
– DEC B with postulated severe fuel damage.

- 2 important issues by expert opinion;
- 21 phenomena (also called thermal-hydraulics related safety issues) were identified by WP3 in PIRT analysis of safety related knowledge gaps.
- "Safety-related features based on the results and expert opinion of WP4":
 - 5 important issues by expert opinion;
 - 22 phenomena were identified by WP4 in PIRT analysis of safety related knowledge gaps.

PIRT tables in D5.2 deliverable [6] present the values of phenomena importance level (IL) and knowledge level (KL), relative relevance (RR) and relative dispersion (RD). The phenomena will not be judged for compliance with safety requirements criteria for safety-related features (namely, identified phenomena need to be first related to safety-related features that could be affected by them), but with respect to knowledge compliance of phenomena occurring in SCW-SMR technology, and identifying phenomena requiring further research.

Table 3. Challenges (C) to safety-related features derived from SCWR technology [6]

ID	Description
C1_SCWR	<i>"application of novel manufacturing processes"</i>
C2_SCWR	<i>"aggressive chemical effects of SCW in the reactor core"</i>
C3_SCWR	<i>"SCW has not been used in highly radiative environment"</i>
C4_SCWR	<i>"correctness of assumptions and extensions do require confirmation"</i>
C5_SCWR	<i>"most significant knowledge gap related to the fuel technology may be the change in material properties of cladding material as a function of the irradiation damage"</i>
C6_SCWR	<i>"material issues identified (specific issues for the investigated reactor designs)"</i>

Table 4. Challenges (C) to safety-related features derived from HPLWR technology [6]

ID	Description
C1_HPLWR	<i>"core power distribution is heavily influenced by the coolant density distribution through the neutron moderation parameters"</i>
C2_HPLWR	<i>"coolant and moderator flow stability"</i>
C3_HPLWR	<i>"possible Xenon oscillation instabilities"</i>
C4_HPLWR	<i>"possible larger stresses because of the high temperature difference"</i>
C5_HPLWR	<i>"high peak cladding temperature, low fuel burn-up and high hot channel factors require further analyses"</i>
C6_HPLWR	<i>"large uncertainties of heat transfer predictions"</i>
C7_HPLWR	<i>"the safety systems are quite similar to BWRs but „in case of a SCWR the residual heat can be removed only by forced convection inside the RPV”"</i>

Table 5. Challenges (C) to safety-related features derived from BWR technology [6]

ID	Description
C1_BWR	"recirculation pump (not relevant for SCWRs)"
C2_BWR	"large changes in neutron flux, coolant density along the FAs"
C3_BWR	"one-circuit design → radioactive steam arrives to turbines (less physical barriers available, no leak tight primary circuit) → steam line isolation valves are essential"
C4_BWR	"Special containment arrangement (drywell / wetwell)"

Table 6. SMR special issues (I) [6]

ID	Description
I1_SMR	"safety features of large reactors are mainly applicable, but more extensive use of passive safety systems"
I2_SMR	"small core inventory → small decay heat (new safety systems – possibility of air cooling for residual heat removal function) and small source terms (lower release possible to environment, resulting in decreasing or elimination of emergency preparedness zones)"
I3_SMR	"differences in reactor physics parameters (higher enrichment, new fuel types, etc.)"

Table 7. WP2 most important issues (I) and most significant phenomena (P) [6]

ID	Description
I1_WP2	"Oxidation behaviour at long time exposures"
I2_WP2	"Effect of irradiation on cladding materials"
I3_WP2	"Effect of geometry and manufacture process of tubes in the crack initiation and corrosion behaviour of candidate materials" Note by authors of this report: Rods were used for electrochemistry; geometry is also important since WP2 partners try to study "real" components or close to the reality as much as possible.
I4_WP2	"Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature"
P1_WP2	"Radiolysis processes"
P2_WP2	"Resistance of cladding materials under LOCA"
P3_WP2	"Pellet cladding interaction"
P4_WP2	"Overheating of the cladding"
P5_WP2	"Irradiation Assisted Stress Corrosion Cracking (IASCC)"

Table 8. WP3 most important issues (I) and most significant phenomena (P) [6]

ID	Description
I1_WP3	"Party solved issues are: models, methods and correlations for turbulent heat transfer at SC conditions; modelling of deterioration of heat transfer (DHT)"
I2_WP3	"Knowledge gap exists for: formulation of design and safety concept"
P1_WP3	"Heat and mass transfer along corroded and rough surfaces"
P2_WP3	"Heat transfer in water under supercritical pressure conditions"
P3_WP3	"Deterioration of heat transfer (DHT)"
P4_WP3	"Turbulent heat and mass transfer in water under supercritical pressure conditions"
P5_WP3	"Transition from supercritical to subcritical pressure"

Table 9. WP4 most important issues (I) and most significant phenomenon (P) [6]

ID	Description
I1_WP4	"all temperature reactivity coefficients are negative, but differences in the different stages of coolant flow"
I2_WP4	"uncertainties related to ensuring the reactivity reserve (use of HA-LEU or MOX fuel)"
I3_WP4	"shaping of power profile (large number of different FA would be necessary)"
I4_WP4	"moderation optimization (wider FA gap and lower moderator temperature or design modification)"
I5_WP4	"set of refuelling strategy"
P1_WP4	"Simulation methodology issues (lacking parameters or material composition, boundary conditions etc.) – however, expert ranking evaluation could not be performed because of lack of expert activity"

In the original Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) PIRT study [13] the results for the screening parameters RR and RD are colour-coded. The colour coding is not intended to categorise the phenomena, but merely to help the reader to identify at a glance the most relevant/significant phenomena with regard to importance or lack of knowledge [13]. Based on the PIRTs, the expert panel identified phenomena that are of both high importance and high uncertainty, and thus of primary interest for further research. From [13] it is clear that RR is a fairly good indicator for phenomena with high importance and low knowledge level. However, in contrast to IL and KL, RR is a normalised and relative measure, and a high value for RR does not necessarily mean that the phenomenon is important or poorly known in absolute terms. Phenomena that received highly dispersed votes by the panel, as indicated by high values for RD of a phenomenon i , were brought up to discussion.

Following the voting in the frame of OECD/NEA PIRT study [13], the panel was convened to discuss the outcome. The discussion focused on phenomena that had received high IL and/or low KL, and for which there seemed to be significant disagreement between the panellists. These phenomena were identified by use of two screening parameters [13]. The second screening parameter addressed the RD of votes for each phenomenon, i.e. the scatter in experts' opinion regarding importance level and knowledge level [13].

In D5.2 [6] regarding the scales the following is stated: "*For the participants, the ranking possibilities for importance were High / Medium / Low / Insignificant. These categories have been represented in the statistical evaluation with IL numbers 1 / 2 / 3 / 4, respectively. The ranking possibilities for knowledge level were 1 (very limited knowledge) / 2 (partially known) / 3 (known, moderate uncertainty) / 4 (fully known). With these categories the highest significance (i.e. with low importance level and / or low knowledge level) means low numerical scores. High numerical scores for IL/KL mean less significant issues.*"

The PIRT analysis results from annexes 2, 3 and 4 of D5.2 [6] were also taken as input.

Annex 2 – PIRT analysis results, WP2, comprise of the following tables:

- Table A2-1: Significant phenomena identified by WP2 experts
- Table A2-2: Expert ranking by WP2 members
- Table A2-3: Statistics of WP2 ranking results

Annex 3 – PIRT analysis results, WP3, comprise of the following tables:

- Table A3-3: Statistics of WP3 ranking results
- Table A3-1: Significant phenomena identified by WP3 members
- Table A3-2: Expert ranking by WP3 members
- Table A3-3: Statistics of WP3 ranking results

Annex 4 – PIRT analysis results, WP4, comprise of the following tables:

- Table A4-1: Significant phenomena identified by WP4 members
- Table A4-2: Significant phenomena identified by WP4 members for rod ejection accident (REA) conditions

2.2 PIRT WP2 and WP3 analyses recalculation

In D5.2 [6] the values of votes from 1 to 4 were used (see Section 2.1.2 above), while both OECD/NEA [13] and ELSMOR [15] (which follow OECD/NEA [13], see Section 2.1.1.5 of D5.2 [6] for detailed description) approaches use weights for the votes. Using weights between 0 and 1 give the weighted values of IL_w and KL_w also in the range between 0 and 1, making interpretation easier. Also, IL_w and KL_w are input values for calculation of RR. As these values were not presented in D5.2 [6], they were recalculated from tables A2-2 and A3-2 of D5.2 [6], which show expert ranking by WP2 and WP3 members, respectively. How to recalculate the values, the reader can refer to ELSMOR study [15]. The weights in this recalculation were 0, 1/3, 2/3 and 1 for Insignificant, Low, Medium and High importance of phenomenon, respectively, and 0, 1/3, 2/3 and 1 for very limited knowledge, partially known, known and fully known, respectively. Besides RR also IL_w and KL_w values provide useful information for judging the need for further research of the phenomena and helps in interpretation of R&D needs.

Integrated Safety Assessment Methodology (ISAM) [23] explains that the PIRT process with prioritized list of phenomena and the adequacy of the knowledge allows gaps to be defined that then need to be filled by a priority R&D effort. This approach was followed in this study (for details the reader can refer to Figure 4 of ISAM document [23], which shows the knowledge base gap determination). The gaps are for high or medium important phenomena with very limited or partial knowledge plus low important phenomenon with very limited knowledge. Four importance levels and four knowledge levels suggest to use four ranges for colour coding: 0-0.25, 0.25-0.5, 0.5-0.75 and 0.75-1). To consider transition zone (for example, phenomenon with $IL_w=0.51$ is considered medium important and with $IL_w=0.49$ is considered low important), five colour code

ranges were used for IL_w and KL_w (0-0.2, 0.2-0.4, 0.4-0.6, 0.6-0.8 and 0.8-1). This gives the following 9 categories for R&D needs (see also Table 10): H1 - high priority level 1 (high important phenomenon and very limited knowledge), H2 - high priority level 2 (high important phenomenon, which is partially known), M1 - medium priority level 1 (medium important phenomenon and very limited knowledge), M2 – medium priority level 2 (medium important phenomenon, which is partially known), L1 – low priority level 1 (low important phenomenon and very limited knowledge), L2 – low priority level 2 (high important phenomenon, which is partially known to know), L3 – low priority level 3 (medium important phenomenon, which is partially known to know), B – beneficial research (medium important phenomenon, which is partially known to know or low important phenomenon, which is very limited knowledge to partially known), and NN – not needed research (all fully known phenomena, all insignificant phenomena, low important phenomenon, which is partially known to known, low important phenomenon, which is known), where H, M, L, B and NN mean high, medium, low, beneficial and not needed, respectively and the numbers indicated the levels, where 1 indicates the highest and 3 the lowest level.

Table 10. R&D priority for knowledge base gaps

	Importance level				
Knowledge level	0.8-1	0.6-0.8	0.4-0.6	0.2-0.4	0-0.2
0.8-1	NN	NN	NN	NN	NN
0.6-0.8	B	B	NN	NN	NN
0.4-0.6	L2	L3	NN	NN	NN
0.2-0.4	H2	M2	B	NN	NN
0-0.2	H1	M1	L1	L4	NN

Tables 11 and 12 show the colour coded results for WP2 and WP3 PIRT, respectively. The orange coloured cells for IL and KL (proposed in the D5.2 deliverable [6]) denote the most significant phenomena and the lowest knowledge level. As can be seen from the colour legend of Tables 11 and 12, the red colour denotes weighted high importance level (IL_w), weighted low knowledge level of the phenomenon (KL_w), high RR and RD. RR is in contrast to IL_w and KL_w normalized to the maximum value of RR obtained for considered phenomena of selected WP. Therefore, RR is (as the name relative relevance suggest) relative measure and therefore the phenomenon with the highest RR has always the value of RR equal 1 (for example, in Table 11 this is phenomenon no. 8 – 'radiolysis process'), but this does not necessarily mean that the phenomenon is important or poorly known in absolute terms (we should not forget that RR is screening parameter). For radiolysis process the values for $IL_w = 0.90$ and $KL_w = 0.14$ suggest that the phenomenon is indeed very important and very poorly known. It is used for normalization of all RR's as it has the maximum value for product of IL_w times $(1 - KL_w)$, which is $0.9 * (1-0.14) = 0.774$ (absolutely the maximal possible value is of such product is 1). RR value is equal to zero only in that case, when phenomenon is fully insignificant ($IL_w = 0$) and/or fully known ($KL_w = 1$). As can be seen there is no such case in Tables 11 and 12.

Also, in Tables 11 and 12 ranks are provided for RR, IL_w and KL_w . The colour coding is not intended to categorise the phenomena, but merely to help the reader to identify at a glance the most interesting phenomena with regard to importance or lack of knowledge. Finally, R&D needs were judged based on the phenomenon importance level and knowledge level. The phenomena with priority research needs with high importance level, but deemed to be sufficiently well known, do not need to be considered for future research (for example, phenomenon no. 9 in Table 11,

'physicochemical properties of water within the SC region', are rather important, but has the lowest RR value 0.35, because it is known). The results for WP2 PIRT shown in Table 11 demonstrate that RR is good measure as two H1 categories have the highest RR values (1.0 and 0.95), while H2 categories have the second highest values (RR in the range 0.70-0.75). Similarly, the results for WP3 PIRT shown in Table 12 demonstrate that RR for one H1 category have the highest RR value (1.0), while H2 categories have the second highest values (RR in the range 0.71-0.85).

Tables 11 and 12 also show that the highest RD has been obtained for phenomenon no. 3, 'oxide release from the cladding surface' and no. 12, the 'effect of the presence of large and hot structural components', respectively. These two phenomena are not among the most important phenomena (phenomenon no. 3 in Table 12 was even ranked as one of the least important). On the other hand, for the WP2 PIRT analysis, the RD is rather low for the first 5 most important phenomena, while for the WP3 PIRT analysis, the RD is judged to be zero for the first 4 most important phenomena. Zero value of RD is obtained when all experts rank the same either IL or KL, or both.

Table 11. PIRT for WP2

ID	Phenomenon	Ranking - w/o weights		Ranking - weights		Screening parameters		Ranks			R&D needs
		IL	KL	IL _w	KL _w	RR	RD	RR rank	IL _w rank	KL _w rank	
1	Through wall penetrations produced by general or localized corrosion	2.00	2.71	0.67	0.57	0.37	0.27	22	16	2	L3
2	Oxide build-up that impedes heat transfer	1.57	2.29	0.81	0.43	0.60	0.17	12	7	6	L2
3	Oxide release from the cladding surface	2.00	2.50	0.67	0.50	0.43	1.00	20	16	4	L3
3a	Oxide release by dissolution / evaporation	1.83	2.17	0.72	0.39	0.57	0.21	15	11	9	M2
4	Pellet cladding interaction	1.57	1.83	0.81	0.28	0.75	0.15	3	7	19	H2
5	Environmental Assisted cracking (EAC)	1.29	2.29	0.90	0.43	0.67	0.25	7	1	6	L2
6	Changes in the mechanical properties of the materials produced by ageing and/or irradiation	1.57	2.14	0.81	0.38	0.65	0.20	9	7	10	H2
7	Changes in the geometry of tubes produced by irradiation, creep	2.29	1.86	0.57	0.29	0.53	0.19	16	21	17	B
8	Radiolysis processes	1.29	1.43	0.90	0.14	1.00	0.40	1	1	23	H1
9	Physicochemical properties of water within the SC region	1.86	2.86	0.71	0.62	0.35	0.65	23	12	1	NN
10	Resistance of cladding materials under LOCA conditions SCWR	1.29	1.57	0.90	0.19	0.94	0.27	2	1	22	H1
11	Impurity enrichment	1.86	1.86	0.71	0.29	0.66	0.42	8	12	17	M2
12	Oxide release from the cladding surface by spalling	1.57	2.00	0.81	0.33	0.70	0.00	6	7	14	H2
13	Irradiation embrittlement due to He	1.29	2.71	0.90	0.57	0.50	0.16	18	1	2	L2
14	IASCC	1.29	2.14	0.90	0.38	0.72	0.29	5	1	11	H2
15	Hydriding	2.29	2.14	0.57	0.38	0.46	0.29	19	21	11	B
16	Cladding collapse	2.00	2.00	0.67	0.33	0.57	0.45	13	16	14	M2
17	Overheating of the Cladding	1.43	2.00	0.86	0.33	0.74	0.43	4	6	14	H2
18	Overheating of Fuel Pellets	1.86	2.33	0.71	0.44	0.51	0.48	17	12	5	L3
19	Cladding rupture	1.86	2.14	0.71	0.38	0.57	0.23	14	12	11	M2
20	Fuel Rod Mechanical Fracturing	2.00	1.83	0.67	0.28	0.62	0.45	10	16	19	M2
21	Strain Fatigue	2.00	1.83	0.67	0.28	0.62	0.36	11	16	19	M2
22	Fretting Wear	2.33	2.29	0.56	0.43	0.41	0.39	21	23	6	NN

Colour legend adopted in the PIRT:

Importance level	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1
Knowledge level	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1
RR, RD	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1

Table 12. PIRT for WP3

ID	Phenomenon	Ranking - w/o weights		Ranking - weights		Screening parameters		Ranks			R&D needs
		IL	KL	IL _w	KL _w	RR	RD	RR rank	IL _w rank	KL _w rank	
1	Steep non-linear change of SCW fluid material properties	1.20	3.40	0.93	0.80	0.23	0.41	21	5	1	B
2	Heat transfer in water under supercritical pressure conditions	1.00	2.00	1.00	0.33	0.82	0.00	3	1	15	H2
3	Pressure drop (Δp) in water under supercritical pressure conditions	2.00	2.40	0.67	0.47	0.44	0.00	13	18	10	L3
4	Turbulent heat and mass transfer in water under supercritical pressure conditions	1.20	1.80	0.93	0.27	0.85	0.34	2	5	20	H2
5	Heat and mass transfer along corroded and rough surfaces	1.20	1.40	0.93	0.13	1.00	0.41	1	5	21	H1
6	Deterioration of heat transfer (DHT)	1.00	2.00	1.00	0.33	0.82	0.00	4	1	16	H2
7	Transition from supercritical to subcritical pressure	1.20	2.00	0.93	0.33	0.77	0.53	5	5	16	H2
8	Steam and liquid water two phase flow	1.40	2.80	0.87	0.60	0.43	0.41	14	11	6	L2
9	Natural circulation of water under super - or sub - critical pressure conditions	1.40	2.00	0.87	0.33	0.71	0.00	6	11	16	H2
10	Strong coupling between the thermal hydraulics and the reactor physics	1.25	3.00	0.92	0.67	0.38	0.69	18	10	2	B
11	Depressurisation of the primary loop and the travelling depressurisation wave	1.60	2.40	0.80	0.47	0.53	0.51	11	15	10	L3
12	The effect of the presence of large and hot structural components	2.20	3.00	0.60	0.67	0.25	1.00	20	21	2	NN
13	Flow instability under supercritical pressure conditions	1.40	2.40	0.87	0.47	0.57	0.51	10	13	10	L2
14	Allowable maximum cladding temperature	1.00	2.40	1.00	0.47	0.66	0.00	8	1	10	L2
15	Flow stratification in horizontal channels	2.00	2.20	0.67	0.40	0.49	0.53	12	18	14	M2
16	Flooding	1.20	3.00	0.93	0.67	0.38	0.53	17	5	2	B
17	TH and Neutronic instabilities	1.00	2.50	1.00	0.50	0.62	0.00	9	1	8	L2
18	CHF near the critical point	1.50	2.00	0.83	0.33	0.69	0.59	7	14	19	H2
19	Flow induced vibration	2.00	2.50	0.67	0.50	0.41	0.00	15	18	8	L3
20	Mechanical deformation	1.75	2.75	0.75	0.58	0.39	0.81	16	17	7	L3
21	Pellet/cladding interaction	1.67	3.00	0.78	0.67	0.32	0.00	19	16	2	B

Colour legend adopted in the PIRT:

Importance level	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1
Knowledge level	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1
RR, RD	0 - 0.2	0.2 - 0.4	0.4 - 0.6	0.6 - 0.8	0.8 - 1

2.3 Scope of the work

In accordance with WP5.3 description [9], the pre-licensing study will relate the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 (see D5.1 [4]) with the safety-related behaviour and features of the SCW-SMR as identified in WP5.2 [6] and to the available level of detail. No specific experimental verifications are planned for this pre-licensing study [9].

Before defining the scope, the wording "**safety related feature**" had to be clarified. In the amendment to grant agreement [9] this term is used just in the title of D5.2 [6] deliverable, which is entitled "Safety related features of the SCW-SMR concept" [6]. Such title suggests that conceptual design on SCW-SMR safety features is available as input to this report (refer to Section 1.3.2, where it is described that NEI 18-04 [27] suggests pre-conceptual, conceptual, preliminary, and final design). As it was shown in Table 1 above, the ECC-SMART project does not propose the SCW-SMR concept, rather "*it is oriented towards assessing the feasibility and*

identification of safety features of an intrinsically and passively safe SCW-SMR, taking into account specific knowledge gaps related to the future licensing process – especially the assessment of the structural materials with special attention to the influence of irradiation, validation of engineering simulation tools like system-, subchannel, and CFD codes, core design as well as the licensing process itself." [9]. This statement is supported by the description of D5.2 deliverable [9], where it is stated that the *"aim of this the deliverable is to provide answers to the potentially outstanding issues and gaps in knowledge regarding the safety related behaviour of supercritical water reactor"*. Also, WP5.2 task is entitled *"Safety-related findings and conclusions of the WPs 2-4"* [9], and in its description of work it is stated that the *"aim of the task is to provide answers to the potentially outstanding issues and gaps in knowledge regarding the safety-related behaviour of supercritical water reactor"*. In other words, this report did not receive as input **safety related features of SCW-SMR** as suggested by the title of D5.2 [6], rather the **issues and gaps in knowledge regarding the safety-related behaviour of SCW-SMR** were the aim of WP5.2 [6]. This is in line with the amendment of grant agreement where under section "Expected impacts" it is stated [9]:

"The proposed project is oriented toward the identification of a feasibility concept and safety features identification of an intrinsically and passively safe SCW-SMR core, taking into particular account residual heat removal, but also spent fuel management issues, decommissioning, cogeneration and competitiveness."

Therefore in D5.2 [6] deliverable, based on the literature review, challenges to specific features based on SCWR system safety assessment document [25], HPLWR design and analyses book [17], BWR designs (second generation designs, ABWR, ESBWR and especially BWRX-300 [28]), and SMR special issues [29] have been collected. In other words, challenges of safety related features for existing SCWRs and BWRs, which are applicable to future SCW-SMR conceptual design, have been identified (these include material issues, technology issues, reactor physics issues and thermal-hydraulic issues [6]). However, as emphasized in D5.2 [6]: *"However, because of the differences between SCW-SMR and large SCWR designs (e.g. horizontal fuel assemblies or 7-step heat-up section) some of the gained experience cannot be applied for our case"*. In addition, for WP2, WP3 and WP4 most important issues by expert opinion and most significant phenomena based on the results of PIRT analyses have been identified in D5.2 [6] deliverable.

Finally, D5.2 [6] states that based on the "Work Package Periodic Report M18" of WP3, one of the main achievements of WP3 at the time of preparing the D5.2 deliverable [6] was also the SCW-SMR design concept which has been developed based on the concept of HPLWR (see [30] and [31]) and forms the basis for further analysis and design in the frame of ECC-SMART project. It should be noted that according to [30] this design proposal is more a sketch, and a lot more details would still need to be designed to make it more realistic. It means that pre-conceptual design proposal in [30] was the starting point to enable WP3 design- and safety-related thermal-hydraulic investigations for the SCW-SMR and WP4 studying the design- and safety-related neutronic parameters and reactor physics behaviour of the SCW-SMR in order to support the pre-conceptual design (WP4 also contains preliminary core design calculations).

Some of the results were provided after D5.2 [6] publication (e.g. in D3.3 [7], which summarises the results of the pre-conceptual studies on the core layout and passive safety concept of the SCW-SMR as elaborated during the work performed in the WPs of the ECC-SMART project). These are early analyses of passive safety aspects, analysis of pre-conceptual core layout and passive safety concept for SCW-SMR, and safety and design analysis of the SCW-SMR concept).

The scope of this pre-licensing study is therefore to relate the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 with the issues and gaps in knowledge regarding the safety-related behaviour of SCW-SMR as identified in WP5.2. This is in an agreement with the methodology defined in the grant agreement [9] which states that the pre-licensing study will summarize both the already achieved safety criteria and the potential gaps to be further investigated in future research projects.

As explained above, the scope is related to safety-related behaviour of future SCW-SMR. This means that not all safety design requirements as identified in D5.1 [4] could be related to future SCW-SMR, but only those one related to the potentially outstanding issues and gaps in knowledge regarding the safety-related behaviour of future SCW-SMR as identified in D5.2 [6] and to the available level of detail. As can be seen from Section 2.1 of this report, these are the challenges to applicable safety-related features of future SCW-SMR, derived from SCWR, HPLWR and BWR technology and WP2-4 expert opinion issues with respect to future SCW-SMR. D5.2 deliverable [6] also identified the important phenomena in the PIRT analysis of safety related knowledge gaps. The phenomena require different treatment than issues and gaps in the knowledge regarding the safety-related behaviour of SCW-SMR, as explained in the following.

In accordance with Integrated Safety Assessment Methodology (ISAM) [23], Phenomena Identification and Ranking Table (PIRT) is a technique that has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a **spectrum of safety-related phenomena or scenarios that could affect those systems** (i.e. Gen IV nuclear systems), and to rank order those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e., sources and magnitudes of phenomenological uncertainties).

The PIRT method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM [23]. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA). OPT focuses on identifying design provisions intended to prevent, control or mitigate those phenomena. However, it is intended to be applied early in the pre-conceptual design phase, and iteratively through conceptual design, which is not in the scope of this project. The PIRT is also essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

This means, that phenomena should be first related to affected future SCW-SMR safety-related features intended to prevent, control or mitigate those phenomena, before judgment on compliance to safety criteria could be done. Therefore, the in general compliance of future SCW-SMR design could not be done at this stage of deriving requirements on SMR SCWR conceptual design, rather information, which will be needed in future to judge compliance to requirements and criteria has been described.

In addition, the pre-licensing study assesses the feasibility of future SCW-SMR design taking into account specific knowledge gaps related to the future licensing process – especially the assessment of the constructional materials with special attention to the influence of irradiation, validation of engineering simulation tools like system-, subchannel, and CFD codes, core design as well as the licensing process itself. This was done through assessing the provided D5.2 [6]

input for WP2, WP3 and WP4 results and expert opinion, and results of PIRT analysis, which identify the phenomena that are both of high importance and for which knowledge is very limited.

3 Conformance of future SCW-SMR technology to relevant IAEA standards

3.1 IAEA SF-1 safety fundamentals

The IAEA Safety Fundamentals document SF-1 [26] outlines essential principles for ensuring nuclear safety. The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation. This objective is supported by ten safety principles:

Principle 1: Responsibility for Safety

"The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks."

The intent and application of this principle is described in paragraphs 3.3-3.7 of IAEA SF-1 [26].

Principle 2: Role of Government

"An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained."

The intent and application of this principle is described in paragraphs 3.8-3.11 of IAEA SF-1 [26].

Principle 3: Leadership and Management for Safety

"Effective leadership and management for safety must be established and sustained in organizations concerned with, and facilities and activities that give rise to, radiation risks."

The intent and application of this principle is described in paragraphs 3.12-3.17 of IAEA SF-1 [26].

Principle 4: Justification of Facilities and Activities

"Facilities and activities that give rise to radiation risks must yield an overall benefit."

The intent and application of this principle is described in paragraphs 3.18-3.20 of IAEA SF-1 [26].

Principle 5: Optimization of Protection

"Protection must be optimized to provide the highest level of safety that can reasonably be achieved."

The intent and application of this principle is described in paragraphs 3.21-3.24 of IAEA SF-1 [26].

Principle 6: Limitation of Risks to Individuals

"Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm."

The intent and application of this principle is described in paragraphs 3.25-3.26 of IAEA SF-1 [26].

Principle 7: Protection of Present and Future Generations

"People and the environment, present and future, must be protected against radiation risks."

The intent and application of this principle is described in paragraphs 3.27-3.29 of IAEA SF-1 [26].

Principle 8: Prevention of Accidents

"All practical efforts must be made to prevent and mitigate nuclear or radiation accidents."

The intent and application of this principle is described in paragraphs 3.30-3.33 of IAEA SF-1 [26].

Principle 9: Emergency Preparedness and Response

"Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents."

The intent and application of this principle is described in paragraphs 3.34-3.38 of IAEA SF-1 [26].

Principle 10: Protective Actions to Reduce Existing or Unregulated Radiation Risks

"Protective actions to reduce existing or unregulated radiation risks must be justified and optimized."

The intent and application of this principle is described in paragraphs 3.39-3.40 of IAEA SF-1 [26].

These 10 principles collectively provide a robust framework for maintaining a high level of safety in the use of nuclear energy and materials, ensuring that radiation risks are properly controlled and mitigated.

In accordance with D5.1 [4], IAEA SF-1 [26] top-level principles apply to ECC-SMART fully.

The Fundamental Safety Principles [26] establish one fundamental safety objective and ten safety principles that provide the basis for requirements and measures for the protection of people and the environment against radiation risks and for the safety of facilities and activities that give rise to radiation risks. As conceptual design of SCW-SMR is not the objective of the ECC-SMART project, information available in D5.2 [6] allow to judge conformance only for some of the of safety principles. The first is SF-1 [26] Principle 3 on leadership and management for safety, which also requires that safety of facility must be assessed. The second is SF-1 [26] Principle 8 on prevention of accidents, which is related to future safety-related features of SCW-SMR. The other SF-1 [26] safety principles are more related to nuclear power plant operation and are therefore not considered in this report.

Section 2 of SSR-2/1 (Rev. 1) [1] elaborates on the safety objective, safety principles and concepts that form the basis for deriving the safety function requirements that must be met for the nuclear power plant, as well as the safety design criteria. SSR-2/1 (Rev. 1) [1] establishes requirements that apply those safety principles, which are particularly important in the design of nuclear power plants. These are radiation protection in design, safety in design, the concept of defence in depth and maintaining the integrity of design. Safety in design requires measures to prevent and mitigate accidents, limit radiological consequences and that occurrence of an accident with serious radiological consequences is extremely low and that radiological consequences of such accidents are mitigated to the fullest extent practicable.

From the above it can be concluded that compliance of safety principles could be judged based on the compliance to design requirements, which are not in the scope of this report (see Section 2.3). Therefore, no judgement on compliance to SF-1 [26] principles has been. Nevertheless, information is provided on relation of selected SF-1 [26] principles 3 and 8 with safety related behaviour of future SCW-SMR conceptual design.

3.1.1 SF-1 Principle 3 (paragraphs 3.15 and 3.16 on safety assessment)

Safety assessment is required by IAEA SF Principle 3 on leadership and management for safety, which in paragraphs 3.15 and 3.16 describes what is safety assessment and when safety assessment process is performed (as described, safety has to be assessed for facility). IAEA GSR Part 4 standard [22] in paragraph 1.8 defines stages in the lifetime of the facility when the safety assessment is performed. One of these stages is also "development of the design". The initial development and use of the safety assessment provide the framework for the acquisition of the necessary information to demonstrate compliance with the relevant safety requirements.

The safety assessment process for nuclear facilities, as described in the IAEA GSR Part 4 standard [22] and shown in Figure 1, involves several key steps. Initially, the scope and purpose of the assessment are defined, focusing on identifying radiation risks and ensuring compliance

with safety standards. The process mandates that the responsibility for conducting the assessment lies with the legally responsible entity, which could be the facility's operator or owner.

The safety assessment begins with thorough preparation (see Requirement 5 of IAEA GSR Part 4 [22] in Figure 1), including assembling a qualified team, gathering necessary information about the facility, and ensuring the availability of analytical tools and safety criteria. The core of the safety assessment involves a detailed safety analysis, which employs both deterministic and probabilistic methods to evaluate potential safety challenges. This analysis must be verified and benchmarked to ensure its accuracy and reliability.

Furthermore, the assessment examines several critical aspects, including radiation protection measures, engineering features, human factors, and long-term safety considerations. The results of the safety assessment are used to inform improvements in facility design and operation, establish maintenance and inspection programs, and support regulatory submissions. Periodic reviews and updates of the safety assessment are required to account for changes in the facility or new safety information.

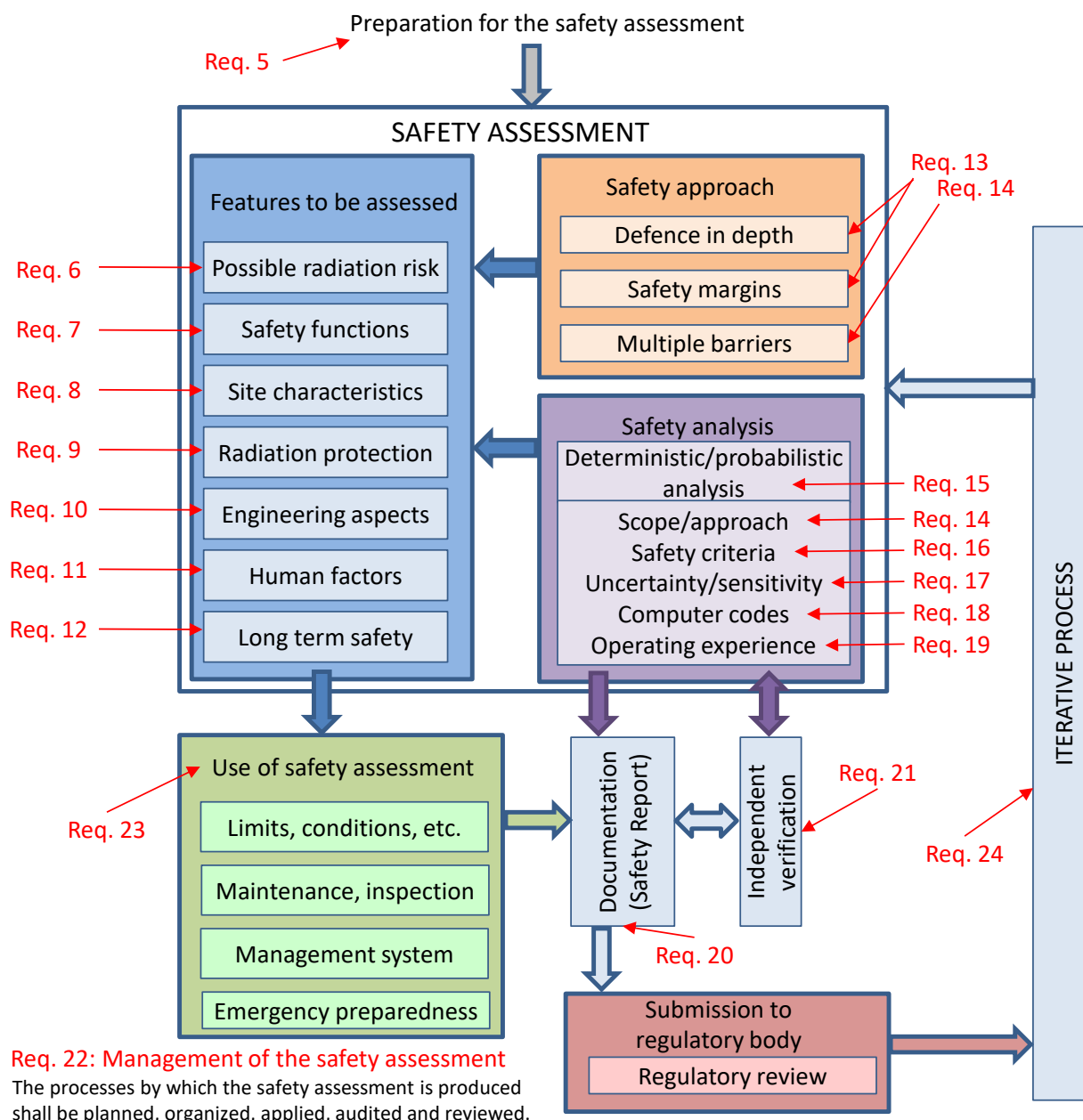


Figure 1. Overview of the safety assessment process (adapted per FIG. 1. of [22])

One of the prerequisites to perform safety assessment of nuclear facility is comprehensive and accurate data on the facility's design, construction, and operational history. This includes detailed information on reactor systems, components, safety features, and operational procedures. From this it is clear that this prerequisite is not satisfied for SCW-SMR as it is not in the scope of ECC-SMART, which is to define set the design requirements for future conceptual design (see also Table 1). Therefore, 'pre-licensing study' has been performed, which is not based on completed conceptual design as is in the case of safety assessment in the frame of 'pre-licensing review' (see also Section 1.3).

An example of already performed safety assessment is SCWR System Safety Assessment document [25]. 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.

The objective of SSA [25] was to perform an assessment of the adequacy of safety provisions incorporated in the conceptual design completed so far for the SCWR systems. Improvements required to demonstrate defence-in-depth (DiD) and additional R&D as required were the expected outcome of the study. The document describes also the ISAM methodology. As design matures and proof-of-concept R&D outcomes become available, tools described in the ISAM methodology will be applied to advance the maturity of the design. Namely, at least pre-conceptual design is needed to use these tools.

SSA [25] presents an overview of the SCWR concept and describes the application of some of the ISAM tools and their outcome. This SSA focuses on the safety of the Canadian SCWR and only a supplementary description is provided on the HPLWR (European SCWR), because the HPLWR project ended before the first version of the ISAM became available.

As explained in the SSA [25] 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 light water reactors (LWRs). Basically, the design of the safety systems adheres to the DiD safety principle [25].

D5.2 [6] list multiple fields where progress is needed as has been identified in SSA document [25]:

- *"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, PuO_2) fuel to enable the implementation of this kind of fuel in different SCWR concepts."*

Relation of principle with safety-related behaviour: According to the work performed in work packages WP2, WP3 and WP4 it was found work has a clear link to the important features related to the licensing process (i.e. qualification of materials, validation of simulation tools, improved reactor core design to ensure the smooth licensing).

Compliance of design: NOT JUDGED

Justification: D5.2 [6] does not provide complete information on future SCW-SMR conceptual design, which is the main reason that the requirement on the safety assessment could not be fulfilled. In general, safety assessment is a well-established process set in IAEA GSR Part 4 [22]. Safety assessments are to be submitted to the regulatory body as part of the licensing or authorization process. This has been done for operating reactors. In general, the knowledge level of the process is sufficient to perform such safety assessment.

The potential gap could be the need to review the regulatory and legal framework. D5.1 [4] provides information on national legal frameworks. 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 NuScale SMR is even completed. BWRX-300 design has cleared the first two phases of the CNSC's VDR process (see Section 1.3), marking a first for a SMR in Canada. Although none of these SMR designs is a SCW-SMR design, some conclusions could be drawn from these processes for the SCW-SMR. For example, the licensing of NuScale shows that adapting existing nuclear regulations to accommodate SMR technology, which differs significantly from traditional large reactors, can be complex. Unique design aspects of SMR, such as passive safety systems and modular construction, also require thorough evaluation and approval processes.

According to OECD/NEA [32] the primary challenge with novel designs is the limited experience base, making it difficult to demonstrate and approve their safety case. This is based on more efficient passive safety features, fewer and less severe failure modes, and reduced off-site emergency planning zones. Additionally, changes to the fuel and/or coolant result in greater deviations from previous regulatory paradigms and may necessitate more flexible licensing approaches. This also requires the development of significant new expertise within nuclear safety regulatory organizations.

The deployment of SMRs requires significant adjustments to the existing legal and regulatory frameworks [33]. This includes shifting from a prescriptive regulatory approach to one that acknowledges the technological innovations of SMRs, harmonizing international regulatory standards, and developing an in-factory certification process. Additionally, it is essential to adjust regulatory fees, enhance the capabilities of regulatory bodies, and reduce the licensing duration. Political / legislative measures are necessary to support these changes and establish the proper infrastructure for SMR licensing.

3.1.2 SF-1 Principle 8 (paragraphs 3.30 through 3.33 on prevention of accidents)

Paragraphs 3.30-3.33 describe the intent and application of SF-1 [26] Principle 8 on prevention of accidents. It is explained that the most harmful consequences arising from facilities and activities have come from the loss of control over a nuclear reactor core, nuclear chain reaction,

radioactive source or other source of radiation [26]. Measures have to be taken to ensure that the likelihood of an accident having harmful consequences is extremely low. The primary means of preventing and mitigating the consequences of accidents is DiD. Finally, accident management procedures must be also developed in advance.

The design for safety of a nuclear power plant (i.e. SSR-2/1 (Rev. 1) [1]) applies the safety principle that to minimize the likelihood of an accident and that practical measures must be taken to mitigate the consequences for human life and health and for the environment of nuclear or radiation accidents (Principle 8 of the Fundamental Safety Principles).

Therefore, compliance to SF-1 [26] Principle 8 is partly judged in Section 3.2.8.

Compliance: see Section 3.2.

Justification: D5.2 [6] regarding defence in depth refer to very general information in SSA document [25], where it is written: "*Basically, the design of the safety systems adheres to the "Defence-in-Depth" (DiD) safety principle.*" It refers also to the HPLWR book [17]: "*The concept of design in depth (DiD) is applied for all plant states. The preliminary targets for different design basic conditions can be seen in Table 2*".

3.2 IAEA SSR-2/1 (Rev. 1) standard

3.2.1 Description of IAEA SSR-2/1 (Rev. 1) standard

IAEA SSR-2/1, (Rev. 1) standard entitled "Safety of Nuclear Power Plants: Design" [1] with top level requirements has been judged fully applicable to ECC-SMART [4]. It contains 82 numbered requirements with additional 199 requirements as appropriate in paragraphs. The requirements are described in Sections 3-6 of IAEA SSR-2/1 (Rev. 1) and are shown in Table 13. Sections 3-6 outline the overarching numbered requirements (highlighted in bold), along with additional requirements in the subsequent paragraphs. Section 3 sets the general requirements for the design organization to manage safety in the design process. Section 4 covers the requirements for principal technical design criteria for safety, including fundamental safety functions, DiD application, and construction provisions. It also addresses safety interfaces with nuclear security and the State system for nuclear material accounting and control, ensuring that radiation risks from the plant are kept as low as reasonably achievable. Section 5 details requirements for general plant design, supplementing the principal technical design criteria to ensure safety objectives are met and safety principles are applied. These general plant design requirements apply to all items (SSC) important to safety. Section 6 specifies requirements for the design of specific plant systems, such as the reactor core, reactor coolant systems (RCSs), containment system, and instrumentation and control systems.

When GIF will provide safety design criteria for SCWR that safety criteria are recommended to be used instead of those in IAEA SSR-2/1 (Rev. 1) [1].

Table 13. IAEA SSR2/1 (Rev. 1) requirements (Sheet 1 of 2)

Section of IAEA SSR 2/1 (Rev. 1)	Area	Requirement
Section 3: Management of safety in design		Requirement 1: Responsibilities in the management of safety in plant design (3.1)
		Requirement 2: Management system for plant design (3.2–3.4)
		Requirement 3: Safety of the plant design throughout the lifetime of the plant (3.5–3.6)
Section 4: Principal Technical Requirements		Requirement 4: Fundamental safety functions (4.1–4.2)
		Requirement 5: Radiation protection in design (4.3–4.4)
		Requirement 6: Design for a nuclear power plant (4.5–4.8)
		Requirement 7: Application of defence in depth (4.9–4.13A)
		Requirement 8: Interfaces of safety with security and safeguards
		Requirement 9: Proven engineering practices (4.14–4.16)
		Requirement 10: Safety assessment (4.17–4.18)
		Requirement 11: Provision for construction (4.19)
Section 5: General Plant Design requirements	Design basis	Requirement 12: Features to facilitate radioactive waste management and decommissioning (4.20)
		Requirement 13: Categories of plant states (5.1–5.2)
		Requirement 14: Design basis for items important to safety (5.3)
		Requirement 15: Design limits (5.4)
		Requirement 16: Postulated initiating events (5.5–5.15)
		Requirement 17: Internal and external hazards (5.15A–5.22)
		Requirement 18: Engineering design rules (5.23)
		Requirement 19: Design basis accidents (5.24–5.26)
		Requirement 20: Design extension conditions (5.27–5.32)
		Requirement 21: Physical separation and independence of safety systems (5.33)
		Requirement 22: Safety classification (5.34–5.36)
		Requirement 23: Reliability of items important to safety (5.37–5.38)
		Requirement 24: Common cause failures
		Requirement 25: Single failure criterion (5.39–5.40)
	Requirement 26: Fail-safe design (5.41)	
	Requirement 27: Support service systems (5.42–5.43)	
	Requirement 28: Operational limits and conditions for safe operation (5.44)	
	Design for safe operation over the lifetime of the plant	Requirement 29: Calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety (5.45–5.47)
		Requirement 30: Qualification of items important to safety (5.48–5.50)
		Requirement 31: Ageing management (5.51–5.52)
	Human factors	Requirement 32: Design for optimal operator performance (5.53–5.62)
	Other design considerations	Requirement 33: Safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant (5.63)
Requirement 34: Systems containing fissile material or radioactive material		
Requirement 35: Nuclear power plants used for cogeneration of heat and power, heat generation or desalination		
Requirement 36: Escape routes from the plant (5.64–5.65)		
Requirement 37: Communication systems at the plant (5.66–5.67)		
Requirement 38: Control of access to the plant (5.68)		
Requirement 39: Prevention of unauthorized access to, or interference with, items important to safety		
Requirement 40: Prevention of harmful interactions of systems important to safety (5.69–5.70)		
Requirement 41: Interactions between the electrical power grid and the plant		
Safety analysis	Requirement 42: Safety analysis of the plant design (5.71–5.76)	

Table 13. IAEA SSR2/1 (Rev. 1) requirements (Sheet 2 of 2)

Section 6: Design of specific plant systems	Reactor core and associated features	Requirement 43: Performance of fuel elements and assemblies (6.1–6.3)
		Requirement 44: Structural capability of the reactor core
		Requirement 45: Control of the reactor core (6.4–6.6)
		Requirement 46: Reactor shutdown (6.7–6.12)
	Reactor coolant systems	Requirement 47: Design of reactor coolant systems (6.13–6.16)
		Requirement 48: Overpressure protection of the reactor coolant pressure boundary
		Requirement 49: Inventory of reactor coolant
		Requirement 50: Cleanup of reactor coolant (6.17)
		Requirement 51: Removal of residual heat from the reactor core
		Requirement 52: Emergency cooling of the reactor core (6.18–6.19)
	Containment structure and containment system	Requirement 53: Heat transfer to an ultimate heat sink (6.19A–6.19B)
		Requirement 54: Containment system for the reactor
		Requirement 55: Control of radioactive releases from the containment (6.20–6.21)
		Requirement 56: Isolation of the containment (6.22–6.24)
	Instrumentation and control systems	Requirement 57: Access to the containment (6.25–6.26)
		Requirement 58: Control of containment conditions (6.27–6.30)
		Requirement 59: Provision of instrumentation (6.31)
		Requirement 60: Control systems
		Requirement 61: Protection system (6.32–6.33)
		Requirement 62: Reliability and testability of instrumentation and control systems (6.34–6.36)
		Requirement 63: Use of computer based equipment in systems important to safety (6.37)
		Requirement 64: Separation of protection systems and control systems (6.38)
		Requirement 65: Control room (6.39–6.40A)
	Requirement 66: Supplementary control room (6.41)	
	Requirement 67: Emergency response facilities on the site (6.42)	
	Emergency power supply	Requirement 68: Design for withstanding the loss of off-site power (6.43–6.45A)
	Supporting systems and auxiliary systems	Requirement 69: Performance of supporting systems and auxiliary systems
		Requirement 70: Heat transport systems (6.46)
Requirement 71: Process sampling systems and post-accident sampling systems (6.47)		
Requirement 72: Compressed air systems		
Requirement 73: Air conditioning systems and ventilation systems (6.48–6.49)		
Requirement 74: Fire protection systems (6.50–6.54)		
Requirement 75: Lighting systems		
Requirement 76: Overhead lifting equipment (6.55)		
Other power conversion systems	Requirement 77: Steam supply system, feedwater system and turbine generators (6.56–6.58)	
Treatment of radioactive effluents and radioactive waste	Requirement 78: Systems for treatment and control of waste (6.59–6.60)	
	Requirement 79: Systems for treatment and control of effluents (6.61–6.63)	
Fuel handling and storage systems	Requirement 80: Fuel handling and storage systems (6.64–6.68A)	
Radiation protection	Requirement 81: Design for radiation protection (6.69–6.76)	
	Requirement 82: Means of radiation monitoring (6.77–6.84)	

3.2.2 Selection of IAEA SSR-2/1 (Rev. 1) requirements for pre-licensing study

From the above requirements and criteria, not all are related to safety features. In this respect the SDC for SMR are followed. The objective of the SSC-SDG from 2024 [34] is to provide detailed guidelines for SFR designers to support the practical application of the SDC in design process to ensure the highest level of safety in SFR design. Note that SSC-SDG from 2024 [34] focuses on the reactor and, therefore, excludes consideration of out-reactor fuel handling and fuel storage.

The SSC-SDG [34] describes the three fundamental safety systems: the core system, the coolant system, and the containment system, which particularly includes selected 14 focal points regarding the SFR-specific safety features as listed in Table 14. As can be seen, they are not directly applicable to SCW-SMR, but same systems could be considered with some adaptation of safety features and focal points.

Therefore, in the following the requirements for the design of specific plant systems of ECC-SMART such as the reactor core, coolant system, and containment systems will be claimed regarding current status of fulfilment with respect to the sets, as specified in Section 1.2 (FULFILLED, FULFILMENT IS PROBABLE, FULFILMENT IS IMPROBABLE, and NOT JUDGED).

Here it should be emphasized that document [34] gives guidelines for SSCs of Gen-IV SFRs, what is not in the scope of ECC-SMART project (see Section 2.3). However, as we are dealing with challenges to safety-related SSCs, which is objective of ECC-SMART, the selection of requirements based on Table 14 is relevant.

It should be noted, that in the previous version SDG from 2019 [35], instead of wording 'Categories' and 'Issues' the terms 'Safety features' and 'Focal points' have been used, respectively - see Table 1 with 14 focal points in [35] and [34], respectively. However, in the text explaining the structure of SSC-SDG from 2024 [34], the terms 'safety features' and 'focal points' are still used:

"Table 1 shows the SFR-specific safety features for each system; the selected 14 focal points are also described in this document."

Table 14. 14 focal points in the SSC-SDG for Gen-IV SFRs (Table 1 of [34])

Systems	Categories	Issues	SDC	Safety approach SDG	SDG statement's number
Reactor Core	Integrity maintenance of core fuels	1. Fuel design to withstand high temperature, high inner pressure, and high radiation conditions	✓		2.1, 2.2, 2.4~2.15
		2. Core design to keep the core coolability	✓	✓	2.18~2.23, 2.28~2.34, 2.43~2.46
	Reactivity control	3. Active reactor shutdown	✓	✓	2.47~2.59
		4. Reactor shutdown using inherent reactivity feedback and passive mechanisms	✓	✓	2.16, 2.17, 2.24~2.29, 2.32~2.35, 2.60~2.75
		5. Prevention of significant energy release during a core damage accident, In-Vessel Retention	✓	✓	2.32~2.37, 2.44~2.46
Reactor Coolant System and associated systems	Integrity maintenance of components	6. Component design to withstand high temperature and low pressure conditions	✓		3.1~3.16, 3.22~3.26, 3.33~3.35, 3.49, 3.52~3.54, 3.56~3.71
	Primary coolant system	7. Cover gas and its boundary	✓		3.18, 3.72~3.81
		8. Measures to keep the reactor coolant level	✓	✓	3.3, 3.9, 3.48, 3.62, 3.68, 3.70, 3.82~3.97
	Measures for prevention and mitigation of sodium chemical reaction	9. Sodium leakage and combustion	✓		3.17, 3.33~3.36, 3.48~3.50, 3.110, 3.133~3.146
		10. Sodium-water reaction	✓		3.36, 3.147~3.155
	Decay heat removal	11. Securing decay heat removal by natural circulation of sodium	✓	✓	3.1, 3.3, 3.9, 3.39, 3.123~3.128, 3.132
		12. Reliability maintenance (diversity and redundancy)	✓	✓	3.11, 3.22~3.32, 3.102, 3.103, 3.105~3.109, 3.111~3.122, 3.129~3.131
Containment and its associated systems	Design concept and loading	13. Formation of containment boundary and loads on it	✓		4.1~4.39, 4.49~4.52, 4.57~4.59
	Containment boundary	14. Containment function of secondary coolant system	✓		4.13(d), 4.53~4.56

Following the table with 14 focal points in the SSC-SDG for Gen-IV SFRs [34] the table with 16 focal points was developed for SCW-SMR, based on IAEA SSGs for LWR.

As can be seen from Table 15, the systems covered are reactor core and associated features, RCSs and containment structure and containment system. For the reactor core and associated features design the applicable safety requirements for design are Requirements 43-46 of SSR-2/1 (Rev. 1) [1], for which the IAEA specific safety guidelines (SSG) are given in IAEA SSG-52 [16]. For the RCSs design the applicable safety requirements for design are Requirements 47-53 of SSR-2/1 (Rev. 1) [1], for which the IAEA specific safety guidelines are given in IAEA SSG-

56 [36]. For the reactor containment systems design the applicable safety requirements for design are Requirements 54-58 of SSR-2/1 (Rev. 1) [1], for which the IAEA specific safety guidelines are given in IAEA SSG-53 [37].

3.2.3 Applicability of selected IAEA SSR-2/1 (Rev. 1) Requirements 42 through 58 to the SCWR

IAEA TECDOC-1936 [47] presents considerations for the applicability of the requirements of IAEA SSR-2/1 (Rev. 1) for near-term deployment SMRs [1], what has been already done in the frame of D5.1 [4] (see below). Requirements 42 to 58 of IAEA SSR/2-1 (Rev. 1) [1] apply to LWR SMRs as is, with the exception of Requirement 57, for which interpretation is needed [47]: "*In case of SMR designs that do not need access to the containment during power operation or in case of accident conditions, this requirement would not be applicable.*".

Annex 1 and 2 of D5.1 [4] show the results of the review on SDC document for Generation IV SFR system [46] (note: at the time of preparation D5.1 [4] and this report the SDC for SCWR was not available). The results in Annex 1 and 2 of D5.1 [4] show that Requirements 42, 45-47, 51, 53-55 and 57-58 follow the IAEA SSR/2-1 (Rev. 1) [1] original text. For requirements 43, 48-50 and 56 the modifications are not relevant for SCWR, therefore original requirements of IAEA SSR/2-1 (Rev. 1) [1] are used in this document. For Requirement 52 the modification is not accepted (needed) and again original IAEA SSR/2-1 (Rev. 1) [1] text is followed.

The Requirement 44 associated paragraphs 6.3bis and 6.3quater for SFR [46] are not relevant for SCWR, while 6.3ter has been accepted: "*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.*" (note that IAEA SSG-52 [36] provide recommendations 3.72 and 3.152 with respect to flow blockage and mis-loading).

The Requirement 45 associated paragraph 6.6 for SFR [46] is not relevant for SCWR.

The Requirement 46 associated paragraph 6.9 for SFR [46] which states "*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.*" needs re-evaluation. The following comment was given [4]: "*The modification requires re-evaluation. For DEC conditions, usually no additional shutdown systems are required in case of Gen3 reactors. Apart from ATWS, diverse shutdowns systems are capable of stopping the chain reaction.*"

The Requirement 47 associated paragraph 6.13 for SFR is not accepted. The original term 'pressure boundary of reactor coolant system' is more appropriate. The Requirement 47 new associated paragraph 6.15bis, 6.15ter, 6.16bis, 6.16ter, 6.16quater and 6.16quinquies for SFR are not relevant for SCWR.

The Requirement 50 associated paragraph 6.17 for SFR [46] is not relevant for SCWR.

The Requirement 51 associated paragraph 6.18 for SFR [46] is not accepted (modification not necessary, the original IAEA requirement is suitable), modifications to paragraph 6.19 for SFR [46] are not relevant to SCWR and the new 6.19bis paragraph for SFR [46] is not relevant to SCWR.

The Requirement 54 new associated paragraph 6.19C is proposed for SCWR:

"Insertion 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.)"

The Requirement 55 associated paragraph 6.20C modifications for SFR [46] are not relevant for SCWR.

The Requirement 56 associated paragraphs 6.22 and 6.23 modifications for SFR [46] are not relevant for SCWR.

The Requirement 58 associated paragraphs 6.28B and 6.30 omission for SFR [46] is not accepted (the original IAEA requirement is necessary), while 6.29 modifications for SFR [46] are not relevant for SCWR.

From above judgment it could be concluded that paragraphs 5.71 to 5.76, 6.1 to 6.28A are applicable as proposed originally by IAEA SSR-2-1 (Rev. 1) [1] except that paragraph 6.9 needs re-evaluation. Two new paragraphs are suggested, first to accept 6.3ter and new 6.19C.

Therefore, according to the scope of ECC-SMART project assessing the feasibility and identification of safety features of future SCW-SMR conceptual design, taking into account specific knowledge gaps in the assessment of the constructional materials, validation of engineering simulation tools, and core design, the Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1] on safety analysis and above selected Requirements 43-58 of SSR-2/1 (Rev. 1) [1] (based on Table 15 focal points) are considered in the following. The specific safety guidelines for these requirements refer to several other IAEA for SFR [46] requirements, shown in Table 13, which are not in the scope of pre-licensing study, as they are related to the management of safety in design, principal technical requirements and general plant design, for which little or no detailed design information is available about future SCW-SMR conceptual design. Because of little or no detailed design information, also the compliance to Requirements 42-58 of SSR-2/1 (Rev. 1) [1] design safety requirements could not yet be judged, therefore the information needed to perform such safety assessment is provided. Additionally, the compliance of knowledge for the safety-related behaviour is judged.

Table 15. 16 focal points selected from IAEA SSGs for LWRs

Systems	Categories	Issues	SSR-2/1 (Rev. 1)	SSG no.	Paragraphs
Reactor core and associated features	Integrity maintenance of fuel rods and fuel assemblies	1. Fuel design to withstand high temperature, high inner pressure, and high radiation level	Req. 43	SSG-52	2.1-2.27, 3.4-3.76
		2. Core design to keep core coolability and control rod insertion	Req. 44	SSG-52	3.77-3.88
	Reactivity control	3. Control of the reactor core	Req. 45	SSG-52	3.89-3.102
		4. Reactor shutdown	Req. 46	SSG-52	3.103-3.127
Reactor coolant systems	Integrity maintenance of RCS and associated systems	5. SSC design	Req. 47	SSG-56	3.2-3.143, 5.4-5.16
		6. Overpressure protection	Req. 48	SSG-56	3.117-3.120, 5.29-5.41
	Primary coolant system	7. Control of conditions (inventory, temperature, pressure)	Req. 49	SSG-56	5.18-5.19
		8. Coolant cleanup	Req. 50	SSG-56	7.2-7.4
	Decay heat removal	9. Heat removal in operational states	Req. 51	SSG-56	7.6-7.16
		10. Heat removal in accident conditions	Req. 52	SSG-56	7.17-7.26
		11. Heat transfer to ultimate heat sink	Req. 53	SSG-56	4.22-4.27
Containment structure and containment system	Design concept	12. SSC design to ensure containment functions	Req. 54	SSG-53	2.2-2.3, 3.1-3.90
		13. Confinement of radioactive material	Req. 55	SSG-53	2.4, 4.90-4.130
	Mechanical features	14. Isolation of containment	Req. 56	SSG-53	4.154-4.166
		15. Access to the containment	Req. 57	SSG-53	4.174-4.180
	Containment	16. Control of containment conditions	Req. 58	SSG-53	4.131-4.150

For the clarity reasons, the IAEA SSR-2/1, (Rev. 1) [1] overarching requirements will be explicitly stated in this report, while for brevity reasons this practice will not be followed for additional requirements as appropriate in the paragraphs that follow them.

As is it too early to judge compliance of non-existing future SCW-SMR conceptual design, instead of judgement on compliance to IAEA SSR-2/1, (Rev. 1) [1] design requirements, first some guideline on requested information needed for judging compliance is provided, and then selected requirements of IAEA SSR-2/1, (Rev. 1) [1] are related with the safety related behaviour as identified in WP5.2. Based on the knowledge compliance, three sets of the safety criteria and requirements are then identified as described in Section 1.2.

3.2.4 Safety considerations and options to enhance the performance of the ESF of water cooled SMRs (Requirements 42 through 58)

IAEA TECDOC-1785 [48] discussed design safety considerations and options to enhance the performance of the engineered safety features of water cooled SMRs incorporating the lessons learned from the Fukushima Daiichi accident.

Relevant countermeasures to address the lessons learned from the Fukushima Daiichi accident in the design of water cooled SMRs in relation to Requirements 42 through 58 of IAEA SSR-2/1 (Rev. 1) [1] are:

- Ensure robust measures for reactor core cooling and ultimate heat sink;
- Ensure design of safety-related SSCs;
- Ensure measures for prevention and mitigation of hydrogen explosions;
- Use PSA effectively for risk assessment and management.

To ensure robust measures for reactor core cooling and ultimate heat sink relevant safety requirements of SSR-2/1 (Rev. 1) [1] are Requirements 47, 51, 52, 53 and relevant Paragraphs.

For enhancing design of safety-related SSCs relevant safety requirements of IAEA SSR-2/1 (Rev. 1) [1] are Requirements 45, 46, 48, 49, 55, 58 and relevant Paragraphs.

For ensuring measures for prevention and mitigation of hydrogen explosions relevant safety requirements of SSR-2/1 (Rev. 1) [1] are Requirement 58 and relevant Paragraphs.

For consideration of the use of PSA effectively for risk assessment and management relevant safety requirements of SSR-2/1 (Rev. 1) [1] are Requirement 42 and relevant Paragraphs.

3.2.5 Requirements 43 through 46 for design of reactor core and associated features

The fuel and core design are the central issue for a nuclear power plant (NPP). The core design should fulfil the design requirements and criteria. The core thermal-hydraulic characteristics are unique and strongly coupled with the neutronic characteristics of the core [5]. Typical parameters for fuel rod design are e.g. fuel rod heated length, fuel rod diameter, fuel rod cladding material, limits for cladding stress, design conditions, cladding thickness, initial gap size and initial pellet density. In the following, the requested information to judge compliance of licensing requirements of IAEA SSR-2/1, (Rev. 1) [1] for design of the reactor core and associated features is listed, following IAEA SSG-52 guidelines for design of the reactor core for nuclear power plants [16].

In the grant agreement for WP4 objectives the following is stated:

"The following three main objectives have been set for this work package:

1. Selection of proper neutron/reactor physics code for modelling the complex behaviour of SCW-SMR, 2. Calculation of safety related neutron physics parameters, 3. Reactor physics analysis core layouts."

D5.2 [6] includes the results of D2.1 (for on the test matrix and a detailed procedure for testing), D2.2 (summarising basic characterization materials and specimens machining), D4.1 (neutron physics) and D4.2 (investigation of neutron physics parameters relevant to the safety and feasibility of the SCW-SMR with the aid of calculations), while WP4.3 results for preliminary / pre-conceptual core design calculations will be available till the end of project. During the preliminary core design studies, reactor physics relationships, such as beginning of life (BOL) k_{eff} values vs. cycle length, power density distribution and necessary shutdown margins will be determined. Therefore, the Requirements 43 to 46 could not be judged based on the ECC-SMART design information, as this was not objective of the project. Instead, challenges to safety-related features derived from other technologies are used for judgment. Here it should be stressed that for these safety-related features only challenging issues have been provided for input (therefore judgement FULFILLED is almost not relevant).

Requirements 43 to 46 of IAEA SSR-2/1, (Rev. 1) [1] standard are related to the reactor core and associated features.

Requirement 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, in combination with all the processes of deterioration that could occur in operational states."

The processes of deterioration are specified in paragraph 6.1 of IAEA SSR-2/1, (Rev. 1) [1] and include:

- *"Differential expansion and deformation;*
- *External pressure of the coolant;*
- *Additional internal pressure due to fission products and the buildup of helium in fuel elements;*
- *Irradiation of fuel and other materials in the fuel assembly;*
- *Variations in pressure and temperature resulting from variations in power demand;*
- *Chemical effects;*
- *Static and dynamic loading, including flow induced vibrations and mechanical vibrations;*
- *Variations in performance in relation to heat transfer that could result from distortion or chemical effects."*

Compliance of design – requested information: Information on fuel rod analysis is not available in the D5.2 [6]. Therefore, some guideline is given on information needed to demonstrate compliance of future conceptual design with above requirement for performance of fuel rods and fuel assemblies.

To judge the fulfilment of the above requirements the information on fuel and core design should be available, including information on design bases, performance of the fuel system during normal operation, anticipated operational occurrences (AOOs), and accident conditions. Information is also needed on performed testing and inspection of new fuel to ensure that the fuel is fabricated in accordance with the design.

First, general safety considerations in the design of the reactor core should be taken into account, especially management system, design objectives; design basis for SSCs; design for safe operation and reactor core safety analysis as recommended in paragraphs 2.1-2.27 of IAEA SSG-52 guideline [16]. For reactor core safety analysis (Requirement 42 of IAEA SSR-2/1, (Rev. 1) [1]) see Section 3.2.8. Design objectives include information on fundamental safety functions, adequate design based on the concept of defence in depth, proven engineering practices, safety assessment in the design, and features to facilitate radioactive waste management. Design basis for SSCs include information on plant states and postulated initiating events (PIEs), external hazards, design limits, safety classification, engineering design rules, design for reliability, and operational limits and conditions.

Second, specific recommendations for the safe design of fuel rods and fuel assemblies based on specific design Requirement 43 of IAEA SSR-2/1, (Rev. 1) [1] are given in Section 3 (paragraphs 3.4 through 3.76) of IAEA SSG-52 [16]. Based on these recommendations the information on fuel type, coolant, moderator, neutronic design, thermohydraulic design, thermomechanical design of

fuel rods and fuel assemblies is needed to judge compliance to Requirement 43 of IAEA SSR-2/1, (Rev. 1) [1].

For neutronic design the information on design considerations of the core, and nuclear design limits is needed. Nuclear key safety parameters influencing the neutronic design of the core and fuel management strategies should be established from the safety analyses. Other needed information for neutronic design is core reactivity characteristics, maximum reactivity worth and reactivity insertion, control of global and local power, and shutdown margin.

For thermohydraulic design the information on design considerations and thermohydraulic design limits is needed.

For thermomechanical design of fuel rods and fuel assemblies the information on design considerations and fuel design limits is needed. Design considerations include thermal and heatup effects on fuel rods, effects of irradiation on fuel assembly structures, effect of variations of power level, mechanical effects in fuel rods, effects of burnable absorber in the fuel, corrosion and hydriding, crud, hydraulic effects in fuel assemblies, considerations of mechanical safety in the design, and fuel pellet-cladding interaction. Fuel design limits should be established based on all physical, chemical and mechanical phenomena that affect the performance of fuel rods and fuel assemblies. Information is needed for all applicable plant states.

As an example, from fuel rod analyses with the maximum peak steady state condition⁵ the behaviours of the fuel rod and mechanical strength requirement for the fuel rod cladding can be evaluated. It is very important to accurately evaluate the maximum peak cladding temperature. The results in study [5] show that, for the average coolant core outlet temperature of 500 °C, the maximum peak cladding surface temperature can be as high as 740 °C with 95 % confidence and 95 % probability. However, optimizing the design and improving the fabrication qualities can reduce the maximum peak cladding surface temperature. The thermal analysis and mechanical analysis in the same study [5] have been performed to get prediction of pellet-clad gap size and pellet-clad mechanical interaction (PCMI) in high burnup fuel rods (during normal operation and transient conditions). For more details on PCMI see OECD/NEA fuel safety Criterion 9 in Section 4.

Relation of requirement with safety-related behaviour: Requirement 43 of IAEA SSR-2/1, (Rev. 1) [1] is related to the following challenges: C1_SCWR, C2_SCWR, C3_SCWR, C5_SCWR and C6_SCWR.

Compliance of knowledge:

C1_SCWR: FULFILMENT IS PROBABLE

Comment: novel manufacturing processes carry the risk of a first-of-its-kind design and may introduce new technical challenges. However, significant R&D activities have been and are currently underway to qualify novel manufacturing processes (in particular additive manufacturing of safety-classified components) with issuing of associated code cases and nuclear design code evolutions. Thus, FULFILMENT is highly likely.

C2_SCWR: FULFILMENT IS PROBABLE

⁵ The maximum peak denotes a hot spot at the peak state when all core parameters are at their nominal design values. The nominal peak depends on spatial fluctuations of the core parameters. The maximum peak further takes into account various engineering uncertainties. [5]

Comment: the aggressive chemical effects of SCW will require experimental testing and demonstration.

C3_SCWR: FULFILMENT IS PROBABLE

Comment: SCW as a coolant in highly radiative environment requires significantly high level of experimental testing and demonstration before adopting.

C5_SCWR: FULFILMENT IS PROBABLE

Comment: the most significant knowledge gap related to the fuel technology of the SCWR is the change in material properties of cladding material as a function of the irradiation damage - this knowledge gap should be closed during the development of SCW-SMR concept.

C6_SCWR: FULFILMENT IS PROBABLE

Comment: maximum “diametral strain” estimated for the pressure tube of the Canadian SCWR concept after 75 years of full power operation would require validation. In-core irradiation experiments are required at SCWR conditions to validate the presented estimation.

Requirement 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 other than severe accidents, a geometry that allows for adequate cooling is maintained and the insertion of control rods is not impeded."

Compliance of design – requested information: Information on structural capability of the reactor core has not been provided in D5.2 [6]. To judge the fulfilment of the above requirement the information of the fuel assembly structural analysis should be available demonstrating that geometry allows adequate core cooling and that control insertability is maintained.

Specific recommendations for the structural capability of the reactor core for Requirement 44 of IAEA SSR-2/1, (Rev. 1) [1] are given in Section 3 (paragraphs 3.77 through 3.88) of IAEA SSG-52 [16]. The information on mechanical design considerations of core structures and components, design limits for the mechanical design of the core structures and components is needed to judge compliance to Requirement 44 of IAEA SSR-2/1 (Rev. 1) [1].

The information is needed how structural integrity of reactor core structures and components is maintained for all applicable plant states, under various damage mechanisms. In accordance with IAEA SSG-52 [16] damage mechanism may be caused by vibration (mechanical vibration or flow induced vibration) and fatigue; debris effects; thermal, hydraulic and mechanical loads; and chemical and irradiation effects. Especially, damage to reactivity control devices and shutdown, devices, and damage to the reactor coolant pressure boundary are of concern. Design limits are specified in appropriate codes and standards, selected in accordance with the safety class.

Relation of requirement with safety-related behaviour: Requirement 44 of IAEA SSR-2/1, (Rev. 1) [1] is related to C4_HPLWR.

Compliance of knowledge:

C4_HPLWR: FULFILMENT IS IMPROBABLE

Comment: The studies in the frame of HPLWR indicated areas of design optimization for stress and deformation analyses of the RPV, the major reactor internals and of the assembly boxes [19]. Also, the following it is stated [19]: *"A great challenge has been to design the internals of the*

pressure vessel such that they can freely expand under the increased temperature differences, but to seal each component against the others such that cold feedwater cannot penetrate into the hot steam. It is still an open question, how close these sealing systems can be built, and how durable they will be stay under long term operation."

From studies of HPLWR [19] it can be concluded that design changes may also be needed for future SCW-SMR conceptual design, before structural capability of the reactor core will be demonstrated.

Requirement 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."

Compliance of design – requested information: Information provided in D5.2 [6] identified large changes in the neutron flux as a challenge. Also, information on the distributions of neutron flux that can arise in any state of the reactor core in nuclear power plant is insufficient to judge the compliance. Finally, D5.2 [6] states the following: *"For REA transients Figures-of-Merit were power history during REA transient (FoM1) and pin fuel enthalpy during REA transient (FoM2). FoM2 includes the peak cladding temperature for which an acceptance limit is defined. In this phenomenon group, some high-importance issues have been identified with very limited knowledge. These are the rod worth of the ejected control rod and the rate of reactivity insertion during the accident. The common cause of these phenomena is the lack of control rod system design."*

To judge the fulfilment of the above Requirement 45 of IAEA SSR-2/1, (Rev. 1) [1] the recommendations are given in IAEA SSG-52 [16], paragraphs 3.89 to 3.102 for reactor core control system. Information on adequate means of detecting the neutron flux distributions in the reactor core and their changes, instrumentation and detectors for monitoring the core parameters, and reactivity control devices to enable the power level and the power distribution to be maintained within safe operating limits is needed.

Relation of requirement with safety-related behaviour: Requirement 45 of IAEA SSR-2/1, (Rev. 1) [1] is related to the following challenges: C1_HPLWR, C2_HPLWR, C3_HPLWR, C2_BWR.

Compliance of knowledge:

C1_HPLWR: FULFILMENT IS PROBABLE

Comment: As written in D5.2 [6] this is feature of HPLWR. For HPLWR reactor analysis of the core power distribution iterative analysis is required [17]: *"Like with boiling water reactors, the core power distribution is significantly influenced by the coolant density distribution, which is responsible for neutron moderation, together with the moderator water inside the water boxes and between the assembly boxes. The coolant density, in turn, is decreasing by the fissile power so that both, the coolant and moderator heat up and the core power distribution must be analysed iteratively to yield a consistent, coupled solution."*

C2_HPLWR: FULFILMENT IS PROBABLE

Comment: As written in D5.2 [6] in Section 2.2.1.1 Literature review of SCWR reactor designs, High Performance Light Water Reactor - Design and Analyses:

"Another important issue is the problem of coolant and moderator flow stability, also well-known from BWR reactors. The preliminary analysis performed for HPLWR indicates possible Xenon oscillation instabilities that must be considered during the further design."

A Technology Roadmap for Gen IV nuclear energy systems [20] states that important SCW technology gaps are in the areas of SCWR safety, including power-flow stability during operation. In accordance with [20] an SCWR safety research activity is recommended for power-flow stability assessments.

C3_HPLWR: FULFILMENT IS PROBABLE

Comment: The preliminary analysis performed for HPLWR indicates possible Xenon oscillation instabilities that must be considered during the further design (see D5.2 [6]). To understand the behaviour of possible Xenon oscillation instabilities a full core model is required [19] *"In order to fully understand the behaviour of the HPLWR 3-pass core against xenon oscillations, a full-core model applying fast computational methods will be required."*

C2_BWR: FULFILMENT IS PROBABLE

Comment: D5.2 [6] in Section 2.2.2 'Safety features of BWR reactors' provides the following information:

"As a consequence of the in-core boiling, the moderator density and reactor power – and also the neutron flux – is much lower in the upper part of the core, resulting in a larger core volume for given thermal power (compared to PWRs). The reactor vessel volume is much larger than in case of pressurized water reactors due to the steam separation facilities and the internal jet pumps."

Regarding knowledge in large changes in neutron flux, the paper [45] concludes the following: *"Instabilities that may occur during the BWR operation constitute a widely known problem in the scientific community addressed for more than thirty years. A great deal of literature is available including data and models. The analysis of the phenomena involved requires a multidisciplinary approach comprising various areas like transient thermal-hydraulics, neutron kinetics, fuel behavior including in-core fuel management, instrumentation, plant control and monitoring, and detailed knowledge of plant features. The use of large thermal-hydraulic system codes should be promoted in this area, provided 3-D neutron kinetics modeling and suitable numerics and specific user guidelines are implemented."*

Requirement 46: Reactor shutdown

"Means shall be provided to ensure that there is a capability 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."

Compliance of design – requested information: D5.2 [6] states the following: *"The common cause of these phenomena is the lack of control rod system design."*

To judge the fulfilment of the above Requirement 46 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-52 [16], paragraphs 3.103 to 3.127. Information on reactor shutdown system should be provided that it is designed to bring the reactor to a subcritical state from all applicable plant states, and maintains it in this state. Information on reliability, rate of shutdown, different means of shutdown should be also provided. To judge the compliance,

information on wear out of the control rod cladding and the effects of irradiation, such as burnup, changes in physical properties and production of helium gas should be also provided. Information on the number and reactivity worth is also needed. Finally, information on the reliability, effectiveness of reactor shutdown system, and separation of protection systems from control systems is needed.

Relation of requirement with safety-related behaviour: Requirement 46 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related behaviour identified in D5.2 [6].

Compliance of knowledge: N/A

3.2.6 Requirements 47 through 53 for reactor coolant systems

In the following, the licensing requirements of IAEA SSR-2/1 (Rev. 1) [1] for design of reactor coolant systems are judged for compliance. In addition, IAEA guidelines for design of the reactor coolant system and associated systems for nuclear power plants [36] are also used regarding requested information.

Requirement 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."

Compliance of design – requested information: Information on RCSs design has not been provided in D5.2 [6]. To judge the fulfilment of the above Requirement 47 of IAEA SSR-2/1 (Rev. 1) [1] the generic recommendations for design of components are given in paragraphs 3.2-3.143, while specific recommendations for design of RCS are given in paragraphs 5.4-5.16 of IAEA SSG-56 [36]. Generic recommendations for design are common to the reactor coolant system and associated systems and applicable to all water reactors. A design basis should be defined for every SSC and the documentation should provide information on the following [36] (paragraphs 3.8-3.115 of IAEA SSG-56 [36]):

- a) Function(s) to be performed by the structure, system or component (paragraph 3.8);
- b) Postulated initiating events that the structure, system or component has to cope with (paragraphs 3.9-3.12, Requirement 16 of IAEA SSR-2/1 (Rev. 1) [1]);
- c) Loads and load combinations the structure or component is expected to withstand (paragraphs 3.76-3.86);
- d) Protection against the effects of internal hazards (paragraphs 3.14-3.17, Requirement 17 of IAEA SSR-2/1 (Rev. 1) [1]);
- e) Protection against the effects of external hazards (paragraphs 3.18-3.26);
- f) Design limits and acceptance criteria applicable to the design of SSCs (paragraphs 3.44, Requirements 15 and 18 of IAEA SSR-2/1 (Rev. 1) [1]);
- g) Reliability (paragraphs 3.47-3.56, Requirements 21-26, 30 of IAEA SSR-2/1 (Rev. 1) [1]);
- h) Provisions against common cause failures within a system and between systems belonging to different levels of defence in depth;
- i) Safety classification (paragraphs 3.63-3.66, Requirement 22 of IAEA SSR-2/1 (Rev. 1) [1]);
- j) Environmental conditions for qualification (paragraphs 3.68-3.75, Requirement 30 of IAEA SSR-2/1 (Rev. 1) [1]);
- k) Monitoring and control capabilities (paragraphs 5.18 and 5.19, 7.1-7.4 specific for BWR);

- l) Materials (paragraphs 3.88-3.92, Requirement 47 of IAEA SSR-2/1 (Rev. 1) [1]);
- m) Provisions for testing, inspection, maintenance and decommissioning (paragraphs 3.100-3.115, Requirement 29 of IAEA SSR-2/1 (Rev. 1) [1]).

To judge the fulfilment of the above Requirement 47 of IAEA SSR-2/1 (Rev. 1) [1] for design of RCSs, the recommendations are given in IAEA SSG-56 [36], paragraphs 5.4 to 5.16. Documentation on technical specifications for the design and manufacture of the reactor coolant pressure boundary should be available. Technical specifications for the design and manufacture of the reactor coolant pressure boundary aim to ensure high reliability by adhering to proven codes and standards. These specifications include damage mode analysis, material selection, load identification, manufacturing and inspection practices, and an in-service inspection program to maintain equipment quality. High confidence in design and manufacture prevents the need to consider large component failures as initiating events.

Documentation on design must consider failure modes such as plastic deformation, buckling, fatigue cracking, and brittle fracture. Conditions affecting equipment integrity, like corrosion and ageing, should be identified and prevented through design, manufacturing, operating provisions, and in-service inspections.

Documentation on detection systems for coolant leaks must present accuracy, reliability, and responsiveness of detection systems. From documentation it must be shown that the structural design of pressure boundaries considers a limited set of load combinations, accounting for normal operation, operational occurrences, accident conditions, and site hazards. When materials are used at low temperatures, information on allowable loads, permitted operational pressure-temperature ranges, and protection systems against brittle fracture need to be provided. Information that stress limits does not exceed limits, and that design pressure and temperature are not exceed should be provided.

Information on maximum pressure during emergency conditions and on monitoring thermohydraulic conditions throughout the plant's lifetime should be also provided.

Relation of requirement with safety-related behaviour: Requirement 47 of IAEA SSR-2/1 (Rev. 1) [1] is partly related to I2_WP3 issue.

Compliance of knowledge:

I2_WP3: FULFILMENT IS IMPROBABLE

Comment: D5.2 [6] provide the following information: "*There are no accepted formulation of design and safety concept. The detailed thermal hydraulic analysis cannot start until their availability. This issue has been regarded also partially solved (see below).*

Based on the "Work Package Periodic Report M18" of WP3, the main achievements of WP3 are so far:

1. The SCW-SMR design concept has been developed based on the concept of HPLWR;"

However, when checking SCW-SMR concept in [31], information suggests that mainly core design has been proposed: "*Based on the concept of the High Performance Light Water Reactor (HPLWR), an innovative concept of a small modular reactor is presented, which shall have smaller peak cladding temperatures than the HPLWR due to better coolant mixing, and which shall enable a passive residual heat removal by natural convection. The core is designed with horizontal fuel assemblies"*.

Requirement 48: Overpressure protection of the reactor coolant pressure boundary

"Provision shall be made to ensure that the operation of pressure relief devices will protect the pressure boundary of the reactor coolant systems against overpressure and will not lead to the release of radioactive material from the nuclear power plant directly to the environment."

Compliance of design – requested information: The only information provided in D5.2 [6] on overpressure protection is for BWRX-300: *"The isolation condensers (ICs) provide overpressure protection without the need for safety relieve valves."*

To judge the fulfilment of the above Requirement 48 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-56 [36], paragraphs 3.117-3.120 and 5.29-5.41.

For all pressure retaining components of the reactor coolant system and associated systems information on overpressure protection need to be provided. Information on the discharge capacity and location of overprotection device is also requested.

For overpressure protection devices of reactors documentation should provide information on redundancy of safety relief valves and how they open sequentially at different pressures to prevent unnecessary coolant discharge. For pressure control it must be shown how pressure is maintained below design limits during PIEs. The documentation must also provide information on discharge capacity and how industry code pressure limits are met. It must demonstrate that overpressure protection systems minimize water hammer effects and that it is powered by uninterruptible sources.

Documentation must also show that safety valves do not have shut-off valves in their discharge lines, and any relief valve used ensure reliable closing. Information on independent valve position indicators and that steam relief are monitored for leaks should be provided. Safety valves layout considerations and that valves and piping can discharge steam, steam-water mixtures, and water should be presented. The spurious opening of safety valves should be considered –with frequencies lower than those considered for loss of coolant accidents (LOCAs). It must be shown that components that can increase primary circuit pressure have systems to stop their operation to prevent inadvertent pressure increases.

Relation of requirement with safety-related behaviour: Requirement 48 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

Requirement 49: Inventory of reactor coolant

"Provision shall be made for controlling the inventory, temperature and pressure of the reactor coolant to ensure that specified design limits are not exceeded in any operational state of the nuclear power plant, with due account taken of volumetric changes and leakage."

Compliance of design – requested information: There is no information provided in D5.2 [6] with respect to provisions for controlling inventory, temperature and pressure of reactor coolant.

To judge the fulfilment of the above Requirement 49 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-56 [36], paragraphs 5.18-5.19. Information on design provisions implemented for the monitoring, display and control of important reactor coolant system parameters (e.g. reactor coolant system pressure and temperature, reactor coolant

system water inventory, steam and feedwater flow (for BWRs) to maintain these parameters within the ranges specified for normal operation and anticipated operational occurrences (AOOs), and to detect any early deviation from the normal values needs to be provided.

Relation of requirement with safety-related behaviour: Requirement 49 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

Requirement 50: Cleanup of reactor coolant

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

Compliance of design – requested information: There is no information provided in D5.2 [6] with respect to cleanup of reactor coolant. To judge the fulfilment of the above Requirement 50 of IAEA SSR-2/1 (Rev. 1) [1] the guidelines are given in IAEA SSG-56 [36], paragraphs 7.2-7.4.

The documentation on water cleanup system should include information on limitation of the concentration of contaminants and impurities in the reactor coolant.

For example, for Japan SCWR [5] the following information is provided: *"The reactor is also cleaned with the reactor clean-up system. It should be mentioned that the reactor cleanup system is used for only the startup, and the coolant is purified in the condensate system during the once-through operation as done in FPPs. Also, as in BWRs, the condenser is vacuumized in order to prepare it for startup."*

Relation of requirement with safety-related behaviour: Requirement 50 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

Requirement 51: Removal of residual heat from the reactor core

"Means shall be provided for the removal of residual heat from the reactor core in the shutdown state of the nuclear power plant such that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded."

Compliance of design – requested information: There is no information provided in D5.2 [6] with respect to means for removal of residual heat from the reactor core. Following information is given in D5.2 [6]:

- *"BWR reactors apply natural convection for decay heat removal after shutdown of the reactor, which is impossible for the HPLWR."*
- *"(The issue „decay heat“ has been added later to the phenomena list, but it was excluded from the analysis because the ranking had been already done by that time.)"*

To judge the fulfilment of the above Requirement 51 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-56 [36], paragraphs 7.6-7.16 (with recommendations on isolation condensers (if included in the design) in 7.12-7.16).

Information on residual heat removal in various shutdown conditions, ensuring the reactor coolant system can cool down quickly from hot shutdown to refuelling conditions is needed. The

documentation should demonstrate that the heat removal system removes residual heat in the event of single failure and continues cooldown if a train or division is unavailable, though at a slower rate. Documentations should provide information that residual heat removal is possible during a loss of off-site power and that it is designed in compliance with all the recommendations for design in Section 3 of IAEA SSG-56 [36], if they are operated after design basis accident (DBA) (refer also to design basis for SSCs under Requirement 47 above).

For isolation condensers, if used, the documentation must show that it meets reactor coolant system standards for design (refer also to design basis for SSCs under Requirement 47 above), have reliable control valves, ensure 72 hours of operation without refilling, handle maximum stresses, and include leak detection and gas prevention features.

Relation of requirement with safety-related behaviour: Requirement 51 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

Requirement 52: Emergency cooling of the reactor core

"Means of cooling the reactor core shall be provided to restore and maintain cooling of the fuel under accident conditions at the nuclear power plant, even if the integrity of the pressure boundary of the primary coolant system is not maintained."

Compliance of design – requested information: There is no information provided in D5.2 [6] with respect to means of cooling the reactor core. Following information is given in D5.2 [6] for HPLWR:

"The book [10] also describes the behaviour of the HPLWR in case of a loss of feed water accident and shows the necessity of an additional emergency core cooling system. This can be a steam-driven high-pressure coolant injection system as the ones applied in present BWRs. An alternative solution would be a closed loop with an emergency condenser located inside the containment (similar to older BWR isolation condensers). In order to ensure easier control of water flow, a motor driven recirculation pump is suggested for the system. These possibilities are shown in Figure 8."

To judge the fulfilment of the above Requirement 52 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-56 [36], paragraphs 7.17-7.26.

Information should be provided that means of emergency cooling of the core cool the fuel within limits for fuel and cladding integrity. For small pipe breaks, the documentation must provide information on any complementary systems which may be required to ensure adequate cooling. Also, information on preventing or limiting fuel uncovering during various primary pipe breaks should be provided for emergency core cooling system (ECCS), and that design include both for high and low-pressure capabilities to maintain coolant inventory. Information on long-term residual heat removal is also requested, even if reactor coolant system integrity is compromised. The documentation must also provide objective evidence, that ECCS components are safety class 1, following industry codes, and designed to prevent failures. Information on instrumentation to control suppression pool temperature and water level during normal and accident conditions is also needed. Information on design features to be used in design extension conditions (DEC) should be provided as well.

Relation of requirement with safety-related behaviour: Requirement 52 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

Requirement 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."

Compliance of design – requested information: There is no information provided in D5.2 [6] with respect to heat transfer to an ultimate heat sink. To judge the fulfilment of the above Requirement 53 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations are given in IAEA SSG-56 [36], paragraphs 4.22-4.27. Information that the design of heat transfer systems ensures multiple methods for transferring residual heat to the ultimate heat sink, adhering to the defence-in-depth concept, should be provided. Information should be provided that if the primary system cannot operate in hot conditions, the secondary side can release heat directly to an alternative heat sink or to the atmosphere, acting as a second ultimate heat sink. Information on independent heat transfer chains is essential, particularly for core melting accidents. Information on design and manufacture should be provided, too.

Relation of requirement with safety-related behaviour: Requirement 53 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any safety-related challenges or issues.

Compliance of knowledge: N/A

3.2.7 Requirements 54 through 58 for containment structure and containment system

In the following, the licensing requirements of IAEA SSR-2/1 (Rev. 1) [1] for design of containment structure and containment system are judged for compliance. In addition, IAEA guidelines for design of the reactor containment and associated systems for nuclear power plants [37] are also used.

Requirement 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 in accident conditions."

Compliance of design – requested information: The information provided in D5.2 [6] on containment system for the reactor does not provide evidence on containment safety functions: confinement of radioactive sources, protection against natural external events and human induced events, and radiation shielding [37]. The following information is provided in D5.2 [6]:

- *"As for the last engineering barrier, a pre-stressed cylindrical concrete containment has been designed for the HPLWR. The compact containment has an inner diameter of 20 m and inner height of 23.5 m, designed for an internal pressure of 0.5 MPa. On the top of the traditional BWR-like pressure suppression pool (wetwell) there are 4 upper pools located on the top of the containment building with a combined volume of 1100 m³. This upper pool is used for the automatic depressurization system and it contains the submerged emergency condensers as well. There are also 4 containment condensers*

mounted on the ceiling of the drywell, connected to the upper water pool. The containment systems can be seen on Figure 9."

- *"The isolation of the main steam line requires an efficient pressure suppression system being able to accept the produced steam in case of an isolation. For this purpose practically all the operating BWRs have a pressure suppression water pool, the so-called wetwell. This makes possible also having smaller containment free volumes (drywell), as it is possible to blowdown from the containment building as well in case of containment overpressure. For present BWRs the volume of the drywells is about 5000-7000 m³, while the volume of the wetwell water pool is about 3000-4000 m³.*
- *For isolated states BWRs also have passive or active cooling systems for residual heat removal. Older BWRs have isolation condensers, i.e. heat exchangers in a large in-containment water pool, while other BWR designs apply the so-called Reactor Core Isolation Cooling System (RCIC), which is a steam-driven cooling system injecting external water into the reactor. The emergency core cooling systems (ECCS) usually consist of high-pressure and low-pressure systems."*
- *ABWR introduced some advanced technology features (such as internal reactor pumps instead of external recirculation loops), as well as passive heat removal systems for reactor cooling and containment cooling for DEC conditions. The passive emergency condenser is able to remove heat from containment even after a reactor vessel melt-through.*
- *BWRX-300 has a dry containment, which can withstand to LOCA accidents. For accident conditions, the cooling of the containment is performed passively, into the equipment pool with the help of three independent heat exchangers."*

To judge the fulfilment of the above Requirement 54 of IAEA SSR-2/1 (Rev. 1) [1] the recommendations for containment safety functions and design approach of the containment and its associated systems are given in IAEA SSG-53 [37], paragraphs 2.2-2.16. Information on containment safety function need to be provided (e.g. confinement, protection and radiation shielding) and the conditions under which these safety functions have to be accomplished.

The recommendations for design basis of the containment and its associated systems is given in paragraphs 3.1-3.90 of IAEA SSG-53 [37]. Design basis comprise of PIEs, internal and external hazards, accident conditions (DBAs and DEC), design limits, reliability, defence in depth, practical elimination of conditions that could lead to an early radioactive releases or a large radioactive release, safety classification, environmental qualification, codes and standards, and use of probabilistic safety analyses in design (see also Chapter 3 of IAEA SSG-53 [37]).

Relation of requirement with safety-related behaviour: Requirement 54 of IAEA SSR-2/1 (Rev. 1) [1] is related to C4_BWR challenge.

Compliance of knowledge:

C4_BWR: FULFILLED

Comment: C4_BWR challenge is more a description than a challenge. It seems that special containment arrangement with drywell and wetwell makes it more complicated.

Requirement 55: Control of radioactive releases from the containment

"The design of the containment shall be such as to ensure that any radioactive 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."

Compliance of design – requested information: The information provided in D5.2 [6] on containment system for the reactor does not provide evidence on radioactive releases.

To judge the fulfilment of the above Requirement 55 of IAEA SSR-2/1 (Rev. 1) [1] on control of radioactive releases from the containment the recommendations are given in IAEA SSG-53 [37], paragraphs 2.4 and 4.90-4.130. Information must be provided regarding provisions designed for preventing and limiting the radioactive releases specified for different plant states, defined in Requirement 55 of IAEA SSR-2/1 (Rev. 1) [1]. Further information on the estimated source term, leaktightness of the containment, secondary confinement building, containment bypass, and reduction of radioactive material in the containment atmosphere should be provided. In general, a single system is not sufficient to reduce the concentration of airborne radioactive material, and multiple systems should be employed (e.g. deposition on surfaces, containment spray systems, pressure suppression pools, ventilation and venting systems). Information on these systems need to be provided.

Relation of requirement with safety-related behaviour: Requirement 55 of IAEA SSR-2/1 (Rev. 1) [1] is related to I2_SMR special issue.

Compliance of knowledge:

I2_SMR: FULFILLED

Comment: I2_SMR is more special safety feature rather than issue. Namely, in the D5.2 [6] the following statement is done:

"The World Nuclear Association (WNA) summarizes the special safety-related features of SMRs as follows:

- ...
- *the smaller core size and the resulting low thermal power leads to reduction of source term, i.e. smaller possible releases in accident conditions;"*)

Note: The possible multi-unit arrangement of SMR technologies (such as for the NuScale in a common water pool) can be a challenge in spite of the smaller individual core inventory.

Requirement 56: Isolation of the containment

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

Compliance of design – requested information: The information provided in D5.2 [6] on containment is not sufficient. The following information is provided for HPLWR in D5.2 [6]:

- *"The lack of separated primary and secondary circuit means also the lack of the primary pressure boundary as an engineering barrier. This special feature of BWRs also affects safety systems, as it is necessary to ensure the fast isolation of the containment building from the environment in case of an accident. For this purpose, the Main Steam Isolation Valves (MSIV) and containment isolation valves (CIV) are inevitable."*

- *"because of the use of one coolant circuit, the isolation of containment (with active or passive containment isolation valves) is inevitable for accident conditions;"*

To judge the fulfilment of the above Requirement 56 of IAEA SSR-2/1 (Rev. 1) [1] on containment isolation, the recommendations are given in IAEA SSG-53 [37], paragraphs 4.154-4.166. Information must be provided regarding containment penetrations not part of a closed loop and communicating with reactor coolant or containment atmosphere that have two isolation valves in series, one inside and one outside the containment. For each such valve information that reliably and independently actuated needs to be provided. Information must be also provided regarding number and location of isolation valves for loops that are closed either inside or outside the containment, for closed loops penetrating containment, for small dead-end instrumentation lines and for closed instrumentation lines. Information that automatic containment isolation during accidents not hinder essential accident mitigation systems need to be provided. Documentation should also describe the need on overpressure protection, performance standards on containment extensions, and measures for open containment.

Documentation should consider containment isolation features that could bypass containment. Information should be provided that systems connected to the primary circuit and containment atmosphere automatically isolate during accidents if not needed for safety, and that isolation devices are designed for specific leaktightness and closure times. Finally, information regarding leakage tests on each isolation valve is requested.

Relation of requirement with safety-related behaviour: Requirement 56 of IAEA SSR-2/1 (Rev. 1) [1] is not related to challenges or issues.

Compliance of knowledge: N/A

Requirement 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."

Compliance of design – requested information: No information is provided in D5.2 [6] on access to the containment.

To judge the fulfilment of the above Requirement 57 of IAEA SSR-2/1, (Rev. 1) [1] on access to the containment, the recommendations are given in IAEA SSG-53 [37], paragraphs 4.175-4.180.

With respect to the design of penetrations for access by personnel or equipment to the containment (containment air locks), information should be provided how the air locks equipped with doors interlocked ensure that at least one door is closed during reactor operations and in accident conditions and how are these penetrations designed to prevent undue exposure of workers to radiation in operational states. For the requested two air lock doors information is needed which plant conditions they withstand and how the chamber size between is sized. Regarding the inner door of the air locks the sealing type and provisions for testing the leaktightness of the doors and the inter-seal space should be described.

Information on equipment hatches should include information what is taken into account in the design. Finally, for containment openings (i.e. penetrations, air locks and hatches) the conditions allowing opening should be described.

Relation of requirement with safety-related behaviour: Requirement 57 of IAEA SSR-2/1 (Rev. 1) [1] is not related to any challenges or issues.

Compliance of knowledge: N/A

Requirement 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 buildup 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."

Compliance of design – requested information: No information is provided in D5.2 [6] on control of containment conditions.

To judge the fulfilment of the above Requirement 58 of IAEA SSR-2/1 (Rev. 1) [1] on control of containment conditions, the recommendations are given in IAEA SSG-53 [37], paragraphs 4.132-4.150 dealing with management of combustible gases. Information on management of combustible gases include generation of combustible gases, threats due to combustible gases in design extension conditions with core melting, and measures for the mitigation of hydrogen combustion and for the prevention of hydrogen combustion challenging the containment integrity [37].

Information on phenomena taken into account for identification of the sources of combustible gases should be provided (e.g. radiolysis of water, metal-water reactions, chemical reactions in the containment, degassing of hydrogen dissolved in the primary coolant, releases from the hydrogen tanks used for control of the primary coolant chemistry, interactions between the molten core and concrete producing hydrogen and carbon monoxide).

Threats to the containment are usually caused by high pressure and thermal loads arising from the production of large quantities of non-condensable gases and by various regimes of combustion of the combustible gases. Information on the measures for the mitigation of hydrogen combustion and for the prevention of hydrogen combustion challenging the containment integrity should be provided (for example selection of material, free space inside the containment, removal, transport, homogenization and venting). For example, for removal of hydrogen, passive autocatalytic recombiners may be used and for homogenization and venting sprays and mixing fans may be used. The documentation should provide with the information on all provisions used.

Relation of requirement with safety-related behaviour: Requirement 58 of IAEA SSR-2/1 (Rev. 1) [1] is related to I4_WP2 issue and P1_WP2 phenomenon.

Compliance of knowledge:

I4_WP2: FULFILMENT IS PROBABLE

Comment: In the report D5.2 [6] the following information is provided: *"Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature – The radiolysis processes in SCW are not well-known. Moreover, there is not much information on the effect of pressure (p) [MPa] and temperature (T) [K] in the electrochemical behaviour of SCW. This "material issue" is under investigation as well during the project by the WP2 partners"*.

At the time being it is difficult to judge that design changes of the containment are needed, as this is not yet in the scope of the project. Nevertheless, it should be kept in mind that radiolysis of the water in the core and radiolysis of the water in the sump or in the suppression pool should be taken into account in the identification of sources of combustible gases in the containment (see paragraph 4.133 of IAEA SSG-53 [37].)

Compliance of knowledge:

P1_WP2: FULFILMENT IMPROBABLE

Comment: the relative importance of radiolysis process obtained from WP2 PIRT ranking results it is 1 suggesting that this is the most important phenomenon with the lowest knowledge.

In the report D5.2 [6] the following information is provided:

- Section 2.3.2.2: "*The combination of neutron irradiation and oxidative environment needs also further evaluation as neutron radiation can modify not only the behaviour of the material but also the environment (with radiolysis process generating hydrogen, oxygen and hydrogen peroxide and free radicals, having big effect on corrosion).*"
- Section 2.3.2.2: "*8. Radiolysis processes - Production of oxygen by radiolysis must be studied in order to reduce the corrosion of cladding and structural components;*"
- Annex 2 – PIRT analysis results, WP2, 8. Radiolysis processes:
 - Short description of the item or notes: "*Production of oxygen by radiolysis must be studied in order to reduce the corrosion of cladding and structural components.*"
 - Rationale of phenomena ranking (PR) by the proposer of the item: "*May have an accelerating effect on all corrosion processes.*"
 - Rationale of knowledge ranking (KR) by the proposer of the item: "*Some modelled data exists on the radiolysis in SCW, but its effect on the corrosion phenomena in SCW is unknown.*"

From above information it can be seen that concern of this process is corrosion. At the same time radiolysis process (phenomenon no. 8 in PIRT for WP2) generate hydrogen, oxygen and hydrogen peroxide and free radicals.

3.2.8 Requirement 42 for safety analysis

Requirement 42 (with specific requirements in associated paragraphs 5.71 to 5.76) of IAEA SSR-2/1 (Rev. 1) [1] standard is related to safety analysis.

Requirement 42: Safety analysis of the plant design

"A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed."

Paragraph 5.71 of IAEA SSR-2/1, (Rev. 1) [1] standard require that design basis for items important to safety (i.e. safety features) and their links to initiating events and event sequences are confirmed. Paragraphs 5.72 and 5.73 of IAEA SSR-2/1 (Rev. 1) [1] standard require that safety analysis provide assurance that defence in depth has been implemented in the design of the plant and uncertainties have been considered. Paragraph 5.74 of IAEA SSR-2/1 (Rev. 1) [1] standard requires analytical assumptions, methods and degree of conservatism used in the design of the plant are updated and verified for the current or as built design. Paragraph 5.75 of IAEA SSR-2/1 (Rev. 1) [1] standard requires that safety analysis establish and confirm design

bases for items important to safety, characterization of plant specific PIEs, analysis and evaluation of events sequences from PIEs, comparison of the results of the analysis with acceptance criteria, and demonstration that in the accidents is possible by safety actions for the automatic actuation of safety systems in combination with operator actions actuate the safety systems. Finally, Paragraph 5.76 of IAEA SSR-2/1 (Rev. 1) [1] standard requires due account of PSA of the plant for all modes of operation and for all plant states, including shutdown.

Compliance of design – requested information: D5.2 [6] on safety analysis of plant design provides the following information:

- Section 2.2.1.1 Literature review of SCWR reactor designs, High Performance Light Water Reactor - Design and Analyses: "*The book [10] also describes the behaviour of the HPLWR in case of a loss of feed water accident and shows the necessity of an additional emergency core cooling system. This can be a steam-driven high-pressure coolant injection system as the ones applied in present BWRs. An alternative solution would be a closed loop with an emergency condenser located inside the containment (similar to older BWR isolation condensers). In order to ensure easier control of water flow, a motor driven recirculation pump is suggested for the system. These possibilities are shown in Figure 8.*"
- Section 2.3.3.2, WP3 - PIRT analysis of safety related knowledge gaps: "*Most of the identified phenomena are considered in all operational conditions, however, some of them are specifically defined for LOCA accidents.*"
- Section 2.3.3.3 WP4 - PIRT analysis of safety related knowledge gaps: "*BME NTI experts identified 22 relevant phenomena for general plant conditions (Normal Operation / All Conditions) and 8 phenomena for control rod ejection transients (REA – Rod Ejection Accidents).*"
- Section 2.3.3.3 WP4 - PIRT analysis of safety related knowledge gaps: "*For REA transients Figures-of-Merit were power history during REA transient (FoM1) and pin fuel enthalpy during REA transient (FoM2). FoM2 includes the peak cladding temperature for which an acceptance limit is defined. In this phenomenon group, some high-importance issues have been identified with very limited knowledge. These are the rod worth of the ejected control rod and the rate of reactivity insertion during the accident. The common cause of these phenomena is the lack of control rod system design.*"

Information on SCW-SMR plant design concept (and plant design) has not been provided in D5.2 [6]. In this respect the grant agreement [9] is quite clear in WP3 objectives: "*Derivation of European-Canadian-Chinese (ECC) joint design requirements for an ECC SCW-SMR design concept; This design requirement document shall serve as a basis for a future conceptual design project. The conceptual design of such a reactor itself is thus not the objective of the proposed project.*"

The objectives of WP3 [9], which besides thermalhydraulic deals with safety of SCW-SMR, are:

- "*Creation of an extended validation database....*"
- "*Model development and validation to assess turbulent heat and mass transfer and pressure drop along corroded surfaces.*"
- "*Development and validation of heat transfer correlations and models relevant for the SCW-SMR operation and safety.*"
- "*Implementation and validation of the developed correlations and models for applications in system codes;*"

- *Improvement, implementation and validation of engineering models for turbulent heat transfer under supercritical conditions;*
- *Experimental, numerical and analytical investigations of key thermal hydraulic phenomena relevant for the SCW-SMR design concept.*
- *Safety and design analysis of the SCW-SMR concept.*
- *Derivation of European-Canadian-Chinese (ECC) joint design requirements for an ECC SCW-SMR design concept."*

From above objectives it is clear that only deterministic analyses were in the scope of WP3.

The recommendations for deterministic safety analysis (DSA) of nuclear power plants (NPPs) are given in IAEA SSG-2 (Rev. 1) [17]. IAEA SSG-2 (Rev. 1) [17] guide addresses the use of DSA for design or licensing purposes, aimed at demonstrating (with adequate margins) compliance with established acceptance criteria. It focuses on neutronic, thermohydraulic, fuel (or fuel channel) and radiological analysis. For other types of analysis, in particular structural analysis of structures and components, no detailed guidance on performing such analyses is provided. IAEA SSG-2 (Rev. 1) [17] guide also does not describe specific phenomena, nor does it systematically identify the key factors essential for neutronic, thermohydraulic, fuel (or fuel channel) and radiological analysis. The structure of the guide is shown Table 16.

Table 16. Structure of IAEA SSG-2 (Rev. 1) guide

IAEA SSG-2 (Rev. 1) Guide Section	Paragraphs
2. General considerations	2.1-2.19
3. Identification, categorization and grouping of postulated initiating events and accident scenarios	3.1-3.57
4. Acceptance criteria for deterministic safety analysis	4.1-4.18
5. Use of computer codes for deterministic safety analysis	5.1-5.43
6. General approaches for ensuring safety margins in deterministic safety analysis	6.1-6.29
7. Deterministic safety analysis for different plant states	7.1-7.72
8. Documentation, review and updating of deterministic safety analysis	8.1-8.18
9. Independent verification of deterministic safety analysis by the licensee	9.1-9.21

To judge the fulfilment of the above Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1] the information on a description of the safety analyses performed to assess the safety of the plant in normal operation and in response to PIEs and accident scenarios on the basis of established acceptance criteria should be provided. These analyses include deterministic safety analyses of normal operation, AOOs, DBAs and DEC, including considerations relating to the event sequences to be 'practically eliminated', as well as the probabilistic safety assessment. More detailed description on the information needed is provided in IAEA SSG-2 (Rev. 1).

Important part of deterministic safety analysis are also acceptance criteria, which are categorized as safety criteria, design criteria and operational criteria. Safety criteria relate either directly to the radiological consequences of operational states or accident conditions, or to the integrity of barriers against releases of radioactive material. Design criteria are design limits for individual SSCs, which are part of the design basis as important preconditions for meeting safety criteria (see Requirement 28 of SSR-2/1 (Rev. 1) [1]). IAEA SSG-2 (Rev. 1) safety guide addresses only safety criteria. Operational criteria are rules to be followed by the operator during normal operation and AOOs, which provide preconditions for meeting the design criteria and ultimately the safety criteria.

In the scope of this report is mainly development and validation of correlations and models, and numerical and analytical investigations of key thermal hydraulic phenomena relevant for the SCW-SMR design concept.

IAEA SSG-2 (Rev. 1) safety guide in Chapter 5 provides recommendations for the use of computer codes, including verification (paragraphs 5.14-5.20), validation (paragraphs 5.21-5.39), qualification of input data (paragraph 5.40) and documentation of computer codes (5.41-5.43).

For verification of computer code information need to be provided both for model verification and system code verification. Verification of a computer code must ensure that the code accurately represents the mathematical model of the real system (model verification) and conforms to documentation, including numerical methods, transformation of equations, and user options. This verification is conducted through review, inspection, and audit, using checklists for review and selected audits for quality assurance. The verification process involves reviewing source code in relation to its documentation, examining the design concept, basic logic, flow diagrams, algorithms, and computational environment. If the code is run on a different platform than originally verified, its validity on the new platform must be assessed. Source coding verification ensures conformity to accepted programming practices and consistency with documentation. For complex codes integrating simpler ones, verification must ensure correct design and implementation of links/interfaces between the codes, adhering to the documentation.

Validation of a computer code ensures the mathematical models represent the real system accurately and documented information is needed. This involves information on comparing code outputs with real system observations or experimental data. Validation should build confidence in the code's predictive capabilities, suitable for the analysis type, and assess uncertainty in predicted parameter values. Validation occurs in two phases: by the code developer and independently by the user. It involves basic tests, separate effect tests, integral effect tests, and nuclear power plant level tests, ensuring comprehensive coverage of the code's intended applications. Validation also includes code-to-code comparisons or engineering judgment when experimental data is limited. Validation should cover the full range of parameters and conditions the code is designed for. Complex applications may require a validation matrix, including test data from various facilities and conditions. Boundary and initial conditions must be appropriate, and scaling effects should be considered. Validation against experimental data should account for uncertainties, transposing these to real plant applications. Documenting the code's range of validity and limitations is crucial, and results are used to determine uncertainty in calculations. For conservative codes, predictions should align conservatively with experimental data. User decisions, like model selection and nodalization, significantly affect results; hence, comprehensive procedures and guidelines are essential to minimize user effects. The nodalization must detail important phenomena and plant characteristics, and consistent nodalization philosophy should be used for both tests and full-scale analyses, with sensitivity analyses ensuring stable results.

Computer code input data, representing the nuclear plant model, must adhere to best practice guidelines and be independently checked. This data compilation should include technical drawings, operating manuals, procedures, set point lists, pump performance charts, process diagrams, instrumentation diagrams, control diagrams, and other plant documentation.

Computer code documentation is crucial for reviewing models and correlations, ensuring they are valid and appropriate for their intended range. It should detail model uncertainties, overall code uncertainties for typical applications, user guidelines, and input descriptions to ensure proper

code use. Documentation must include experimental data, validation results, and descriptions of computer options considered. Guidance should cover the influence of modelling parameters, typical code applications, nodalization types, and expected trends. Complete documentation typically comprises an abstract, theory manual, user's manual, input description, programmer's manual, and validation report. Continuous error tracking and reporting are essential, assessing impacts on completed safety assessments.

Relation of requirement with safety-related behaviour: Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1] is related also to important phenomena in Tables 11 and 12 for PIRT for WP2 and WP3, respectively. Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1] is partly related to P1_WP4 simulation methodology issues.

Compliance of knowledge:

PIRT for WP2: for knowledge judgement regarding R&D needs refer to Table 11.

Comment: Table 11 shows the R&D needs according to methodology described in Section 2.2. There are two H1 high priority phenomena and five H2 priority phenomena with respect to R&D. The PIRT for WP2 results R&D categorization is shown also in Table 17 (for details refer to Table 11).

Table 17. PIRT for WP2 R&D categorization

H1 - High priority level 1 (high important phenomenon and very limited knowledge) phenomena:

- P_WP2_8: Radiolysis processes,
- P_WP2_10: Resistance of cladding materials under LOCA conditions SCWR.

H2 - high priority level 2 (high important phenomenon, which is partially known) phenomena:

- P_WP2_4: Pellet cladding interaction;
- P_WP2_17: Overheating of the Cladding;
- P_WP2_14: IASCC;
- P_WP2_12: Oxide release from the cladding surface by spalling;
- P_WP2_6: Changes in the mechanical properties of the materials produced by ageing and/or irradiation.

M2 – medium priority level 2 (medium important phenomenon, which is partially known) phenomena:

- P_WP2_11: Impurity enrichment;
- P_WP2_21: Strain Fatigue;
- P_WP2_20: Fuel Rod Mechanical Fracturing;
- P_WP2_16: Cladding collapse;
- P_WP2_19: Cladding rupture;
- P_WP2_3a: Oxide release by dissolution / evaporation.

L2 – low priority level 2 (high important phenomenon, which is partially known to know) phenomena:

- P_WP2_2: Oxide build-up that impedes heat transfer;
- P_WP2_5: Environmental Assisted cracking (EAC);
- P_WP2_13: Irradiation embrittlement due to He.

L3 – low priority level 3 (medium important phenomenon, which is partially known to know) phenomena:

- P_WP2_18: Overheating of Fuel Pellets;
- P_WP2_3: Oxide release from the cladding surface;
- P_WP2_1: Through wall penetrations produced by general or localized corrosion.

B – beneficial research (medium important phenomenon, which is partially known to know or low important phenomenon, which is very limited knowledge to partially known) phenomena:

- P_WP2_7: Changes in the geometry of tubes produced by irradiation, creep;
- P_WP2_15: Hydriding.

NN – not needed research phenomena:

- P_WP2_22: Fretting Wear;
- P_WP2_9: Physicochemical properties of water within the SC region.

PIRT for WP3: for knowledge judgement regarding R&D needs refer to Table 12.

Comment: Table 12 shows the R&D needs according to methodology described in Section 2.2. There is one H1 high priority phenomena and six H2 priority phenomena with respect to R&D.

The PIRT for WP3 results R&D categorization is shown also in Table 18.

Table 18. PIRT for WP3 R&D categorization

<p>H1 - High priority level 1 (high important phenomenon and very limited knowledge) phenomena:</p> <ul style="list-style-type: none">• P_WP3_5: Heat and mass transfer along corroded and rough surfaces. <p>H2 - high priority level 2 (high important phenomenon, which is partially known) phenomena:</p> <ul style="list-style-type: none">• P_WP3_4: Turbulent heat and mass transfer in water under supercritical pressure conditions,• P_WP3_6: Deterioration of heat transfer (DHT),• P_WP3_2: Heat transfer in water under supercritical pressure conditions,• P_WP3_7: Transition from supercritical to subcritical pressure,• P_WP3_9: Natural circulation of water under super - or sub -critical pressure conditions,• P_WP3_16: CHF near the critical point. <p>M2 – medium priority level 2 (medium important phenomenon, which is partially known) phenomena:</p> <ul style="list-style-type: none">• P_WP3_15: Flow stratification in horizontal channels. <p>L2 – low priority level 2 (high important phenomenon, which is partially known to know) phenomena:</p> <ul style="list-style-type: none">• P_WP3_14: Allowable maximum cladding temperature,• P_WP3_17: TH and Neutronic instabilities,• P_WP3_13: Flow instability under supercritical pressure conditions,• P_WP3_8: Steam and liquid water two phase flow. <p>L3 – low priority level 3 (medium important phenomenon, which is partially known to know) phenomena:</p> <ul style="list-style-type: none">• P_WP3_11: Depressurisation of the primary loop and the travelling depressurisation wave,• P_WP3_19: Flow induced vibration,• P_WP3_20: Mechanical deformation,• P_WP3_3: Pressure drop (Δp) in water under supercritical pressure conditions. <p>B – beneficial research (medium important phenomenon, which is partially known to know or low important phenomenon, which is very limited knowledge to partially known) phenomena:</p> <ul style="list-style-type: none">• P_WP3_16: Flooding,• P_WP3_10: Strong coupling between the thermal hydraulics and the reactor physics,• P_WP3_21: Pellet/cladding interaction,• P_WP3_1: Steep non-linear change of SCW fluid material properties. <p>NN – not needed research phenomenon:</p> <ul style="list-style-type: none">• P_WP3_12: The effect of the presence of large and hot structural components.
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P1_WP4: FULFILMENT IS PROBABLE

Comment: D5.2 [6] provide the information on WP4 PIRT analysis of safety related gaps for normal operation/all conditions and rod ejection accidents (REA). The general phenomena (22 significant phenomena have been identified, see Table A4-1 of D5.2 [6]) in WP4 PIRT analysis of safety related gaps for normal operation/all conditions were further divided into three groups, related to the computation methodology (deterministic calculations – e.g. for TH coupling and Monte Carlo calculation), material and neutron physics related data and design related data. In WP4 PIRT analysis of safety related gaps for REA 8 significant phenomena have been identified

(see Table A4-2 of D5.2 [6]). In Section 2.3.3.3 of D5.2 [6], the information on WP4 PIRT analysis of safety related gaps for normal operation/all conditions and rod ejection accidents is given, including the following:

- *"In order to get a general overview about safety-related reactor-physical features of SCW-SMR, the survey for collecting relevant phenomena has been sent to the members of WP4 as well. In the first round a set of phenomena identified by reactor physics experts of BME NTI has been disseminated for WP4 members. BME NTI experts identified 22 relevant phenomena for general plant conditions (Normal Operation / All Conditions) and 8 phenomena for control rod ejection transients (REA – Rod Ejection Accidents)."*
- *"The first group of identified phenomena is connected to the computation methodology of SCW-SMR neutronics. These include issues concerning deterministic and Monte Carlo calculations. Figures-of-merit can be the ability to determine the effective multiplication factor (FoM1) or neutron flux distribution (FoM2). Some relevant phenomena – based on the two expert ranking – are the determination of time discretization for deterministic calculations and the boundary conditions and group constants for the reflector. For Monte Carlo calculations the accurate material properties seem to be the most important issue with high influence on FoM and with limited knowledge. For less extent geometric parameters and material composition are also relevant."*
- *"Material and neutron physics related data were considered as a separate set of phenomena by the experts ranked as High or Medium for importance. Some of these issues have limited knowledge as well, playing a significant role in simulation uncertainties. Such less known parameters are the material composition of control rods and structural material temperatures and densities and parameters of burnable poison materials."*
- *"The third group of relevant phenomena are the design related data. According to the experts, there are significant uncertainties concerning the design of reactivity control systems and in-core detectors, which have also a High / Medium importance ranking, so they can influence the neutronics calculations."*
- *"For REA transients Figures-of-Merit were power history during REA transient (FoM1) and pin fuel enthalpy during REA transient (FoM2). FoM2 includes the peak cladding temperature for which an acceptance limit is defined. In this phenomenon group, some high-importance issues have been identified with very limited knowledge. These are the rod worth of the ejected control rod and the rate of reactivity insertion during the accident. The common cause of these phenomena is the lack of control rod system design."*

In the computation methodology group for normal operation/all conditions determination of time discretization for deterministic calculations and the boundary conditions and group constants for the reflector are judged to be important phenomena for deterministic calculations and accurate material properties for Monte Carlo calculations. In D5.2 [6] the following information is provided for WP4 PIRT analysis phenomena no. 4, no. 5 and no. 8 of Table A4-1:

- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 4:
 - Descriptive name of reactor physical phenomena: *"Time discretisation"*
 - Short description of the item or notes: *"It is necessary to determine the appropriate temporal discretization applied in time-dependent calculations."*
- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 5:
 - Descriptive name of reactor physical phenomena: *"Reflector boundary and group constants"*
 - Short description of the item or notes: *"For deterministic codes the modelling of reflector region is not as straightforward as for neutron transport codes."*

- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 8:
 - Descriptive name of reactor physical phenomena: "*Material properties (temperature, density, etc.)*"
 - Short description of the item or notes: "*For accurate simulations, the material properties need to be known.*"

For material and neutron physics related data group for normal operation/all conditions less known parameters are the material composition of control rods and structural material temperatures and densities and parameters of burnable poison materials. In D5.2 [6] the following information is provided for WP4 PIRT analysis phenomena no. 12, no. 13 and no. 19 of Table A4-1:

- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 12:
 - Descriptive name of reactor physical phenomena: "*Composition of control rods*"
 - Short description of the item or notes: "*For precise calculations, the material composition of control rods must be determined.*"
- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 13:
 - Descriptive name of reactor physical phenomena: "*Temperature and density of structural materials*"
 - Short description of the item or notes: "*Structural material temperatures and densities need to be determined precisely for the applied neutronic (and TH) codes.*"
- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 18:
 - Descriptive name of reactor physical phenomena: "*Applied burnable absorbers*"
 - Short description of the item or notes: "*Neutron transport codes must rely upon the cross sectional data of the different materials assigned to the geometric models. Therefore, these cross sectional lists must be determined precisely in the relevant temperature and energy regions.*"

For design related data group for normal operation/all conditions there are significant uncertainties concerning the design of reactivity control systems and in-core detectors. In D5.2 [6] the following information is provided for WP4 PIRT analysis phenomena no. 20 and no. 22 of Table A4-1:

- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 20:
 - Descriptive name of reactor physical phenomena: "*Reactivity control system*"
 - Short description of the item or notes: "*The design of reactivity control system and its operation/impact during burnup as well as the applied materials have a strong effect on the global neutron flux and the reactivity.*"
- Annex 4 – PIRT analysis results, WP4, Table A4-1, phenomenon no. 22:
 - Descriptive name of reactor physical phenomena: "*In-core detectors*"
 - Short description of the item or notes: "*The design of the in-core detector system is crucial for monitoring neutronic and TH parameters and for reactor operation in both normal and accidental scenarios.*"

For REA transients high-importance issues with very limited knowledge have been identified for the rod worth of the ejected control rod and the rate of reactivity insertion during the accident. In D5.2 [6] the following information is provided for WP4 PIRT analysis phenomena no. 1 and no. 2 of Table A4-2:

- Annex 4 – PIRT analysis results, WP4, Table A4-2, phenomenon no. 1:
 - Descriptive name of reactor physical phenomena: "*Ejected control rod worth*"

- Short description of the item or notes: "*During the REA, the worth of the ejected control rod will determine the amount of reactivity that is inserted into the system. The inserted reactivity will result in power increase which will be limited by the Doppler-feedback.*"
- Annex 4 – PIRT analysis results, WP4, Table A4-2, phenomenon no. 2:
 - Descriptive name of reactor physical phenomena: "*Rate of reactivity insertion*"
 - Short description of the item or notes: "*The reactivity insertion rate will mainly depend on the ejected control rod worth.*"

It is judged that lacking materials or material composition, boundary conditions and other data is expected to be available after completed SCW-SMR conceptual design.

4 Knowledge judgement related to fuel safety criteria by OECD/NEA

The OECD/NEA Working Group on Fuel Safety (WGFS) was tasked with advancing the understanding of fuel safety issues by assessing the technical basis for current safety criteria and their applicability to high burn-up and to new fuel designs and materials [10]. With new fuel and core designs, higher-performance reactor operations, and advanced design and analysis methods, there is concern that existing fuel safety criteria may not be sufficient, especially with higher burn-up levels. Historically, fuel safety margins were defined with conservative safety criteria, choosing bounding values for model parameters and operational history. However, these conservatisms are often unquantified, making it difficult to substantiate the available safety margins. Regulators need to understand these margins and their verification, especially as the industry seeks approval for new fuels or methods. Utilities and vendors also need this information to optimize fuel design and cycle costs.

Most current fuel safety criteria were established in the 1960s and early 1970s, verified against experiments with the fuel designs of that era, often using unirradiated fuels [10]. As designs evolved, additional verification ensured compliance with existing criteria rather than establishing new limits. While these criteria have historically been effective, with no incidents reported due to their inadequacy, new demands on fuel and plant performance have reduced available margins. Optimizing fuel utilization and core performance is leading towards conditions with less operational and experimental experience, raising concerns about the adequacy of current safety margins under these new conditions.

In 2001, the considerations of the task force were published in the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) report entitled Nuclear Fuel Safety Criteria Technical Review [11]. The concept of margin in the 2003 report on “Fuel Safety Criteria in NEA member countries” [12] presented the criteria as either safety, operational or design criteria (refer to Section 3.2.8 for IAEA definitions in [17] for safety, operational and design criteria). The criteria considered in the 2003 report are presented in Table 19.

Table 19. Types of criteria and technical issues identified in the 2003 NEA report on “Fuel Safety Criteria in NEA Member Countries” [12]

Safety criteria	Operational criteria	Design criteria
DNB/CPR safety limit	DNB/CPR operating limit	Crud deposition
Reactivity coefficients	LHGR limit	Stress/strain/fatigue
Shutdown margin	PCI	Oxidation/hydriding
Enrichment	Coolant activity	Hydride concentration
Internal gas pressure	Gap activity	Transport loads
PCMI	Source term	FA fretting wear
RIA fragmentation	Control rod drop time	Clad diameter increase
Non-LOCA runaway oxidation	RIA fuel failure limit	Cladding elongation
LOCA-PCT		Radial peaking factor
LOCA-oxidation		3D peaking factor
LOCA-H release		
LOCA-long-term cooling		
Seismic loads		
Holddown force		
Criticality		
Burn-up		

In the Second Edition of the Nuclear Fuel Safety Criteria Technical Review [10], the WGFS has reviewed the fuel-related criteria, categorizing them by event type or risk significance only modestly. Each criterion is briefly described in various applications, along with the rationale for its use. Design changes, such as different cladding materials, higher burn-up, and mixed-oxide (MOX) fuels, can impact fuel-related margins and criteria. The working group highlights important effects to identify criteria needing re-evaluation. Although not all possible effects are covered, the discussion aims to pinpoint criteria that require ongoing attention.

Complete or sufficient information is not available for the following fuel (safety) criteria [10]: CRUD deposition, cladding oxidation and hydriding, rod internal gas pressure, pellet-cladding and thermal-mechanical loads, fuel melting, fuel fragmentation, cladding embrittlement, gap activity, radioactive source term and high burn-up.

Fuel safety criteria, with derivative fuel design and operational limits, are the focus of [10]. In the following 23 fuel (safety) criteria are considered, while "Other considerations" from [10] are not considered:

- Criterion 1: Critical Heat Flux
- Criterion 2: Reactivity coefficients
- Criterion 3: Criticality and shutdown margin
- Criterion 4: Fuel enrichment
- Criterion 5: CRUD deposition
- Criterion 6: Stress/strain/fatigue
- Criterion 7: Oxidation and hydriding
- Criterion 8: Rod internal gas pressure
- Criterion 9: Thermal mechanical loads and PCMI
- Criterion 10: Pellet cladding interaction (PCI)/stress corrosion cracking (SCC)
- Criterion 11: Fuel melting

- Criterion 12: Linear heat generation rate (LHGR) limits
- Criterion 13: RIA cladding failure
- Criterion 14: Fuel fragmentation and fuel dispersal
- Criterion 15: Non-LOCA cladding embrittlement/temperature
- Criterion 16: LOCA cladding embrittlement
- Criterion 17: Blowdown/seismic/transportation loads
- Criterion 18: Assembly holddown force
- Criterion 19: Fretting wear
- Criterion 20: Coolant activity
- Criterion 21: Fuel gap activity
- Criterion 22: Source term
- Criterion 23: Burn-up

In the following, the phenomena from PIRT for WP2 and WP3 are labelled as 'P_WP2_' and 'P_WP3_', respectively, plus ID number from Tables 11 and 12 (e.g. P_WP3_21 for pellet/cladding interaction phenomenon no. 21 in Table 12).

Criterion 1: Critical Heat Flux [10]

Description: The critical heat flux (CHF) criterion does not apply to the concept of a high temperature core. In Japan [5], critical heat flux design criterion was replaced by evaluating the maximum cladding temperature (see Criteria 15 and 16).

Compliance of design – requested information: N.A.

Relation of requirement with safety-related behaviour: Criterion 1 is related to WP3 phenomenon no. 18 (see Table 12). Refer to Criteria 15 and 16.

Compliance of knowledge: FULFILMENT IS PROBABLE

Comment: WP3 phenomenon no. 18 (P_WP3_18) has RR = 0.69 ($IL_w = 0.83$ and $KL_w = 0.33$), which is the 7th highest value. It is a high important phenomenon (14th rank) with rather low knowledge level (19th rank). As judged as high important phenomenon, the R&D needs were judged as high 2 priority (H2, see Table 12).

Justification: D5.2 [6] mentions CHF under:

- Section 2.3.3.2 WP3 - PIRT analysis of safety related knowledge gaps:
"The list of the 21 relevant thermal-hydraulics related safety issues (the detailed list with short explanation of the issues and with literature references can be found in Annex 3):
....
Critical heat flux (CHF) near the critical point;"
- Annex 3 – PIRT analysis results, WP3:
 - Short description of the item or notes: *"Near the critical point and in subcritical state, mainly under certain accident conditions, CHF may cause heat transfer problems."*
 - Rationale of the item: *"Please, specify."*
 - Figure of merit: *"Fuel cladding temperature (T_c)"*

With respect to peak cladding temperature (candidate to replace CHF criterion) D5.2 [6] states the following: *"Thermal-hydraulic issues (such as coolant and moderator flow stability;*

optimization of peak cladding temperatures or large uncertainties of heat transfer predictions)." Also, it is emphasized that because of the differences between SCW-SMR and large SCWR designs (e.g. horizontal fuel assemblies or 7-step heat-up section) some of the gained experience cannot be applied for ECC-SMART. For peak cladding temperature refer to Criteria 15 and 16.

Criterion 2: Reactivity coefficients [10]

Description: In terms of LWR safety criteria, there is a general requirement that either the moderator temperature coefficient or the total of all reactivity coefficients be negative when the reactor is critical, for providing negative reactivity feedback (or that the effects of any positive reactivity coefficient be inconsequential).

Compliance of design – requested information: Reactivity coefficients are an analytical matter; in terms of LWR safety criteria, there is a general requirement that either the moderator temperature coefficient or the total of all reactivity coefficients be negative when the reactor is critical, for providing negative reactivity feedback (or that the effects of any positive reactivity coefficient be inconsequential) [10].

To satisfy paragraph 3.21 of IAEA SSG-52 [16] dealing with core reactivity characteristics, the information should be provided that nuclear key safety parameters, such as reactivity coefficients, are evaluated for selected core operating conditions (e.g. zero power, full power, beginning of cycle, end of cycle and at key points relating to poison burnout) and for the corresponding fuel management strategy. Analysis results of the dependence of such nuclear key safety parameters on the core loading and on the burnup of the fuel should be available. Information should be provided that appropriate margins are included in the reactivity coefficients or within the modelling approaches used to evaluate reactivity feedback in the safety analysis for all applicable plant states.

Relation of requirement with safety-related behaviour: Criterion 2 is related to the I1_WP4 issue.

Compliance of knowledge: FULFILMENT IS PROBABLE

Comment: Regarding the I1_WP4 issue on negative temperature reactivity coefficients the document on fuel safety criteria [10] states that although the reactivity coefficients may be affected, the effects of new fuel design changes are not considered to affect the corresponding safety criteria themselves.

Justification: D5.2 [6] states the all temperature reactivity coefficients are negative, but there are differences in the different stages of coolant flow:

"The results show that all temperature reactivity coefficients are negative (with some of the local coefficients close to zero) in temperature regions relevant for normal operation. Negative temperature reactivity coefficients serve the inherent safety of the reactor, as rising temperatures would mean a reactivity decrease and consequentially a negative reactivity feedback. It is worth to mention that coolant and moderator temperature coefficients differ significantly in the different stages of coolant flow."

Here it should be noted that no further information is provided in D5.2 [6] what are the safety consequences (if any) of the above I1_WP4 issue. It seems that further research is needed to clarify why the coolant and moderator temperature coefficients differ significantly in the different stages of coolant flow.

Criterion 3: Criticality and shutdown margin [10]

Description: Attaining reactor subcriticality must be assured either by sufficient reactivity worth of control rods and/or sufficient boron concentration in the primary coolant.

Compliance of design – requested information: To satisfy paragraphs 3.24-3.25, 3.114-3.115, 3.150 of IAEA SSG-52 [16] dealing with shutdown margin and criticality, information should be provided that the insertion of control rods provides an adequate shutdown margin in all applicable plant states, information on specification and monitoring of control rod insertion limits and effects of depletion of burnable absorber on the core reactivity, effectiveness of the shutdown system (criticality analysis), any event that could cause inadvertent criticality during core loading or unloading and during handling phases. Also, D5.2 [6] provides information that the HPLWR reactor can be shut down safely with control rods or the diverse emergency boron injection system (however, it is worth noting that in supercritical water cooled reactors boric acid reactivity control is not used for normal operation - similar to BWRs -, which can make the use of emergency boron system somewhat complicated.)

Relation of requirement with safety-related behaviour: Criterion 3 is not related to any challenges, issues and most significant phenomena.

Compliance of challenge, issue or phenomenon: NOT JUDGED (information provided in D5.2 [6] is insufficient to judge the compliance).

Justification: D5.2 [6] does not provide information on criticality and shutdown margin.

Criterion 4: Fuel enrichment [10]

Description: Enrichment limits around 5 wt% U235 are used in connection with criticality considerations for fabrication, handling, and transportation. For some high burn-up applications, higher enrichments may be needed. To date, the validation of criticality safety codes and associated cross-section libraries for LWR fuel has focused on enrichments of 5 wt% and below. Neither benchmarks of code performance nor the bases for extrapolating code performance in the enrichment range of 5-10 wt% have been well established. Moving into this range will require care because the physics of criticality begins to change as enrichments reach 6 wt% and beyond, where single moderated assemblies can go critical and criticality of weakly moderated or unmoderated systems becomes possible.

Enrichments above 5 wt% will require redesigning and re-licensing some fuel fabrication and handling equipment and fuel transportation packages. The possibility of recriticality during accidents, in particular in severe accident core melt sequences should also be addressed as this could alter the progression of such accidents.

Compliance of design – requested information: To satisfy paragraph 3.147 of IAEA SSG-52 [16], a new reactor core analysis should be performed and documented, when fuel enrichment is changed.

Relation of requirement with safety-related behaviour: Criterion 4 is related to the I3_SMR issue.

Compliance of knowledge: FULFILMENT IS PROBABLE

Comment: Regarding I3_SMR issue on higher enrichment, care should be taken using enrichments between 5-10 wt%. Namely, the physics of criticality begins to change as

enrichments reach 6 wt% and beyond. Also, the possibility of recriticality during accidents should be addressed. Also, there is rising international interest in increasing fuel burnup limits and fuel cycle length may require fuel enrichment above 5% and high reactivity/high suppression core designs [51]. Activities to verify that extended enrichment fuel is well understood and that existing design limits are still valid for fuel of 5-8% enrichment have been proposed by OECD/NEA WGFS [51]. If enrichment exceeds 8%, design changes may be potentially needed, or new design limits should be set.

Justification: D5.2 [6] states that according to the Periodic Report, based on the fuel cycle calculations, a proper reactivity reserve can only be achieved with higher uranium enrichments, and the usage of MOX fuel besides UO₂ is essential for longer cycles. D5.2 [6] also states: “*The necessity of the application of higher enrichments may be reduced significantly only if either the target profile can be altered (linear heat rates can be increased on the higher stages) or if from a neutron absorbing point of view, more compatible structural materials (or coatings like zirconium silicide for example) would be applied.*”

Criterion 5: CRUD deposition [10]

Description: Various CRUD⁶ levels are being used by vendors, according to the design models and/or the fuel designs themselves. Firm (safety) limits on CRUD deposition are not defined, although the amount of CRUD deposited and its composition can be significant to the corrosion performance of the cladding.

Compliance of design – requested information: To satisfy paragraphs 3.9, 3.23 and 3.81 of IAEA SSG-52 [16] on fuel and core design, information on safety consideration associated with coolant regarding ensuring that the chemical composition of the coolant is compatible with the materials that are present in the primary circuit (e.g. to avoid crud formation on fuel rods, and to minimize corrosion and the generation of radioactive) need to be provided. Also, information on local effect should be provided like crud induced power shifts, degradation of the heat transfer from fuel rods due to crud, the chemical effects of the coolant and the moderator on the core structures, which include corrosion, hydriding, stress corrosion and crud buildup.

Relation of requirement with safety-related behaviour: Criterion 5 is not related to any challenges, issues and most significant phenomena.

Compliance of knowledge: NOT JUDGED

Comment: The information provided in D5.2 [6] is insufficient to judge the compliance. Complete or sufficient information is not available for fuel (safety) criterion CRUD deposition[10].

Justification: D5.2 [6] does not provide information on CRUD levels.

Criterion 6: Stress/strain/fatigue [10]

Description: The first “1% strain” criterion applies to the long-term strain that occurs after gap closure induced by outer overpressure (creep down). The 1% strain criterion relates to the tangential (circumferential) strain only, and the 2.5% strain criterion relates to the combined tangential and axial strain.

⁶ CRUD – chalk river unidentified deposits – corrosion products deposited on the fuel rod cladding, first identified at Chalk River Laboratories in Canada.

The strain limit is used for Condition I (normal operation) and Condition II (faults of moderate frequency) events. The process includes thermal expansion, but the swelling process dominates. This pellet-cladding mechanical interaction (or PCMI) phenomenon is caused by a combination of cladding creep, fuel swelling (which reaches a maximum at end of fuel life), fuel rod internal pressure, and fuel pellet thermal expansion. The margins from these limits to actual failure stresses and strains are defined from the fuel vendor's database for a particular fuel, cladding, and burn-up range.

The second 1% strain criterion has been used as a PCMI criterion for operational over-power transients (Condition II events).

For PCI prevention, operational rules are established, like limiting the absolute value of power increase and the power increase velocity.

Compliance of design – requested information: To satisfy paragraphs 3.47-3.49, 3.56-3.57 and 3.59-3.64 of IAEA SSG-52 [16] on mechanical effects in fuel rods, considerations of mechanical safety in the design, and fuel pellet-cladding interaction information should be provided regarding stress, strain and fatigue.

Relation of requirement with safety-related behaviour: Criterion 6 is not related to any challenges, issues and most significant phenomena. However, it is related to WP3 phenomenon no. 18 (see Table 12).

Compliance of knowledge: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 21 (P_WP2_21) has RR = 0.62 ($IL_w = 0.67$ and $KL_w = 0.28$), which is the 11th highest value. It is a medium important phenomenon (16th rank), however the knowledge level is partial (19th rank). As P_WP2_21 phenomenon is judged as a medium important phenomenon and knowledge is partially known, the R&D needs were judged as medium level 2 priority, (M2, see Table 11).

Justification: D5.2 [6] provides information on strain fatigue P_WP2 PIRT analysis phenomenon no. 21:

- Section 2.3.3.1 WP2 - PIRT analysis of safety-related knowledge gaps:
"The 23 identified phenomena and their short explanation are the following:
...
21. Strain Fatigue - Fuel rod failure due to fatigue or corrosion fatigue;"
- Annex 2 – PIRT analysis results, WP2:
 - Short description of the item or notes: *"Fuel rod failure due to fatigue or corrosion fatigue."*
 - Rationale of PR by the proposer of the item: *"The design criterion for cladding strain fatigue is that the cumulative fatigue usage factor be less than 0.9 when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles."*
 - Rationale of KR by the proposer of the item: *"Please specify!"*

Criterion 7: Oxidation and hydriding [10]

Description: Cladding corrosion, including oxidation degrades material properties, most importantly affecting the effective cladding-to-coolant heat transfer, which results in an increase in fuel temperature and stored energy. Hydrogen absorption by the cladding and subsequent

formation of hydrides may lead to embrittlement of the cladding. These phenomena become increasingly important at higher exposures for some existing alloys. However, the significance of this issue is much reduced for modern alloys with improved corrosion and hydrogen pickup performance. Hydrogen may be added to the coolant to decrease the amount of oxygen present, which is formed by radiolysis, which consequently decreases the rate of zirconium oxidation (this is a remedy for PWRs only). However, if the hydrogen concentration is increased too much, cladding hydriding and subsequent embrittlement could be increased. In several countries the design limits of an average cladding oxide thickness at end of fuel life of 100 micron, and of an average hydride concentration of 500-600 ppm, have effectively become approved safety criteria via the approval of fuel vendor design methodologies. As corrosion of some of the traditional zirconium-based alloys is probably one of the leading parameters that limit the lifetime of nuclear fuel, there is a rationale for reviewing the adequacy of the current applicable limits on maximum local oxidation and hydriding levels in the cladding, especially in view of the performance of highly burnt fuel.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. To satisfy recommendations in paragraphs 3.5, 3.51-3.53 and 3.81 of IAEA SSG-52 [16], information needs to be provided on cladding materials selection considering low oxidation and low hydriding in high temperature conditions. Information how hydrogen pick-up correlations are determined based on cladding corrosion during normal operation for each cladding type so that appropriate fuel design limits for accidents can be expressed as a function of the pre-transient hydrogen content of the cladding. Information is also needed regarding fuel rods and assemblies compatibility with the coolant environment during all operational states, including shutdown and refuelling. Information on the chemical effects of the coolant and the moderator on the core structures and components, which include corrosion, hydriding, stress corrosion and crud buildup, should also be provided.

Relation of requirement with safety-related behaviour: Criterion 7 is related to WP2 PIRT phenomenon no. 15 (see Table 11), which affects the fuel rod cladding.

Knowledge judgement of phenomenon: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 15 (P_WP2_15) has RR = 0.46 ($IL_w = 0.57$ and $KL_w = 0.38$), which is the 19th highest value. WP2_15 phenomenon is medium important phenomenon (21th rank), the knowledge level is partial (11th rank). As it is judged medium important phenomenon and knowledge is partially known, the R&D needs were judged as beneficial (B, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion cladding oxidation and hydriding [10].

Justification: D5.2 [6] provide information on hydriding WP2 PIRT analysis phenomenon no. 15:

- Section 2.3.3.1 WP2 - PIRT analysis of safety related knowledge gaps: "15. **Hydriding - Internal hydriding due to fabrication. External hydriding due to waterside corrosion;**"
- Annex 2 – PIRT analysis results, WP2:
 - Short description of the item or notes: "*Internal hydriding due to to fabrication. External hydriding due to waterside corrosion*"
 - Rationale of PR by the proposer of the item: "*Cladding failure mechanism this is however early-in-life failure mechanism.*"
 - Rationale of KR by the proposer of the item: "*Please specify!*"

Criterion 8: Rod internal gas pressure [10]

Description: Fission gas release (FGR) and fuel rod internal pressure are critical for nuclear fuel behaviour, influencing thermal-mechanical limits. FGR depends on the fuel's microstructure and chemistry, its development with time and fuel temperature. High burn-up (>40-60 GWd/t) rapidly increases FGR. Increased FGR can lead to high internal pressures and deteriorating of the thermal conductivity and heat transfer between fuel pellets and cladding. This induces feedback since an increased fuel temperature enhances the fission gas release. The high internal rod pressures can have an important effect on fuel cladding under transients and postulated accidents behaviour. Regulatory criteria for internal gas pressure—absolute (below RCS pressure) or relative ("no lift-off") are used in various countries. In the "no lift-off" criterion the rod internal pressure may exceed the RCS pressure, but is limited so that the instantaneous cladding creep-out rate due to an internal rod pressure greater than the RCS pressure is not expected to exceed the instantaneous fuel swelling rate, i.e. the fuel to cladding gap does not open.

These criteria should not be fundamentally affected by design changes, although methods to demonstrate compliance will be affected.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraph 3.4, of IAEA SSG-52 [16] provides recommendation that fuel pellet materials should be optimized regarding fission gas retention. Regarding the design of the fuel rods, it is needed to address fuel material (including burnable absorbers) with respect to fission gas release kinetics and fuel rod performance irradiation effects on fuel rod behaviour (e.g. fission gas release).

On the other hand, NEA document for rod internal gas pressure [10] states that fission gas release and resulting fuel rod internal pressure is an important aspect of fuel behaviour.

Relation of requirement with safety-related behaviour: Criterion 8 is not related to any challenges, issues and most significant phenomena.

Knowledge judgement: NOT JUDGED

Comment: The information provided in D5.2 [6] is insufficient to judge the compliance.

Complete or sufficient information is not available for fuel (safety) criterion rod internal gas pressure [10].

Justification: D5.2 [6] does not provide any information regarding to rod internal gas pressure.

Criterion 9: Thermal mechanical loads and PCMI [10]

Description: PCMI refers to the stress and/or strain on the cladding from an expanding pellet, especially during a transient. Pellet expansion results mainly from thermal expansion and gaseous swelling, and if the stress is large enough, it can result in cladding failure. PCMI differs from the related SCC-PCI phenomenon (see Criterion 10) in so far as the latter refers to power ramps (with sufficiently high power levels, sufficiently high ramp rates) where the stress is held for a relatively long period of time and corrosion is necessary for cracking to take place.

The avoidance of mechanical fracture of the cladding during transients due to pellet-cladding mechanical interaction, which is the basic safety criterion, is partially covered by the limit on uniform cladding (plastic and elastic) transient strain of 1%. However, PCMI-induced failures can

occur at local strain levels well below 1% – particularly for brittle cladding at high burn-up (if the cladding exhibits a highly concentrated outer hydride rim).

Compliance of design – requested information: Paragraph 3.68 of IAEA SSG-52 [16] recommends to prevent fuel cladding failure caused by PCMI, possibly assisted by SCC, appropriate operating limits on power changes and power-ramp rates of change should be determined such that the power-ramp failure thresholds are not exceeded. Information demonstrating this recommendation will be needed for future SCW-SMR conceptual design.

IAEA SSG-52 [16], Annex 1: "*3.59 Cladding creepdown and fuel pellet thermal expansion and gaseous swelling will lead to a strain driven pellet–cladding mechanical interaction in all applicable plant states. The failure mode via pellet–cladding mechanical interaction is by ductility exhaustion of the cladding.*"

IAEA SSG-52 [16], Annex 1, 3.40: "*In the assessment of the peak fuel temperatures for operational states, the following burnup dependent phenomena need to be addressed: changes in fuel thermal conductivity and diffusivity and in the thermal **conductance of the pellet–cladding gap**, fuel densification, fuel swelling, accumulation of fission products in the fuel pellets, fission gas release in the free volumes of the fuel rods and any other changes in the pellet microstructure. Owing to irradiation effects, the fuel melting temperature varies as a function of fuel burnup and, thus, needs to be determined using representative irradiated fuel samples.*"

IAEA SSG-52 [16], Annex 1, 3.63: "*The power-ramp failure threshold is the lower bound of the power-ramp induced fuel failures within a burnup range called the ‘critical burnup range’. For fuel burnup values below the critical burnup range, the pellet–cladding gap remains open, so that the power change would have to be larger to reach the same level of stress in the cladding compared to the situation in which the pellet–cladding gap is closed. For fuel burnup values above the critical burnup range, experience shows that the material compound in the pellet–cladding interface generated by irradiation is such that the stress concentration on the inner surface of the cladding is reduced, making stress corrosion cracking in the cladding unlikely. Since the critical burnup range depends on the gap closure kinetics of the pellet–cladding gap, it is, therefore, dependent on the specific material properties of the cladding type and fuel rod design.*"

Relation of requirement with safety-related behaviour: Criterion 9 is not related to any challenge, issue or phenomenon (note: this is judged from WP2 phenomenon no. 4 (see Table 11), where reference "Progress on Pellet–Cladding Interaction and Stress Corrosion Cracking IAEA TEC-DOC ISBN 978-92-0-116421-62" has in title also SCC, which is chemical effect. PCMI specifically refers to the mechanical interaction between the fuel pellet and the cladding.

Knowledge judgement: NOT JUDGED

Comment: The information provided in D5.2 [6] is insufficient to judge the compliance. WP2 phenomenon no. 4 (P_WP2_4) has RR = 0.75 (IL_w = 0.81 and KL_w = 0.28), which is the 3rd highest value. It is high important phenomenon (7th rank), the knowledge level is partial (19th rank). As P_WP2_4 phenomenon is judged high important phenomenon, the R&D needs were judged as high level 2 priority (H2, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion pellet-cladding and thermal-mechanical loads [10].

Justification: D5.2 [6] provide no information on PCMI.

Criterion 10: Pellet cladding interaction (PCI)/stress corrosion cracking (SCC) [10]

Description: Some PCI failures may be associated with SCC in the cladding material and depend upon local power ramps during reactor startup or manoeuvring and during AOOs. Both the stress from the power increase and the corrosion level at the pellet-cladding gap are necessary conditions for SCC-PCI. The PCI limits/rules typically contain a maximum ramp rate for reactor power increase (W/cm/hr), a maximum “single step” power increase (W/cm), and a threshold (in W/cm) above which such power increase limitations apply and a minimum time-period after which the fuel may be considered (pre)conditioned to larger power ramps.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraph 3.48, of IAEA SSG-52 [16] provide recommendation on SCC induced by pellet–cladding interaction in the presence of corrosive fission products which should be prevented (such information needs to be provided for future SCW-SMR conceptual design).

Information should show that no fuel cladding failure takes place due to pellet–cladding mechanical interaction in normal operation and AOOs - see the following supplementary information of IAEA SSG-52 [16], Annex I in Table I-1 for PCI:

"Stress corrosion cracking in the fuel cladding occurs when the stresses on the inner surface of the cladding (as a result of pellet–cladding interaction) reach a certain limit in a corrosive environment. After a power reduction, the thermal contraction of the fuel pellets causes reopening of the pellet–cladding gap (or the gaps between the pellet fragments). If operation at reduced power is maintained long enough (i.e. extended reduced power operation), the fuel cladding will creep down and close the gaps again. The fuel rod is then considered ‘re-conditioned’ at this lower power level. When the reactor core returns to full power at a later time, tensile stresses will appear in the cladding. These residual stresses will increase the susceptibility to stress corrosion cracking driven by the pellet–cladding interaction under the corrosive fission product environment in the fuel rod."

Relation of requirement with safety-related behaviour: Criterion 10 is related to P3_WP2 (WP2 phenomenon no. 4 in Table 11), which affects the fuel rod cladding and to WP3 phenomenon no. 21 (see Table 12), which affects safety analyses.

Knowledge judgement of phenomenon:

P3_WP2 (also P_WP2_4, phenomenon no. 4 in Table 11): FULFILMENT PROBABLE

Comment: WP2 phenomenon no. 4 (P_WP2_4, see Table 11) has RR = 0.75 (IL_w = 0.81 and KL_w = 0.28), which is the 3rd highest value. It is high important phenomenon (7th rank), the knowledge level is partial (19th rank). As it is judged high important phenomenon, the R&D needs were judged as high level 2 priority (H2, see Table 11).

WP3 phenomenon no. 21: FULFILLED

Comment: WP3 phenomenon no. 21 (P_WP3_21) has RR = 0.32 (IL_w = 0.78 and KL_w = 0.67), which is the 19th highest value. It is high important phenomenon, slightly above the limit 0.75 for medium (16th rank), the knowledge level is known (2nd rank). As P_WP3_21 is judged high (close to medium) phenomenon and it is known, the R&D needs were judged as beneficial (B, see Table 12).

Justification: D5.2 [6] provide information on PCI WP2 PIRT analysis phenomenon no. 4 (P_WP2_4) and WP3 PIRT analysis phenomenon no. 21 (P_WP3_21):

- Section 2.3.3.1 WP2 - PIRT analysis of safety-related knowledge gaps: "**4. Pellet cladding interaction** - *This interaction could produce hot-spots and harm the material resistance;*"
- Annex 2 – PIRT analysis results, WP2, phenomenon no. 4:
 - Short description of the item or notes: "*This interaction could produce hot-spots and harm the material resistance*"
 - Rationale of PR by the proposer of the item: "*May result in cladding deterioration.*"
 - Rationale of KR by the proposer of the item: "*Pellet - stainless steel cladding interaction in gen I/II LWRs is known, but is uncertain at SCWR temperatures*"
- Section 2.3.3.2 WP3 - PIRT analysis of safety-related knowledge gaps:
"*The list of the 21 relevant thermal-hydraulics related safety issues (the detailed list with short explanation of the issues and with literature references can be found in Annex 3):*
....
Pellet/cladding interaction (conductance)."
- Annex 3 – PIRT analysis results, WP3, phenomenon no. 21:
 - Short description of the item or notes: "*Please, specify*"
 - Rationale of the item: "*Please, specify*"
 - Figure of merit: "*Fuel cladding temperature (T_c)*"

Criterion 11: Fuel melting [10]

Description: Traditional practice in the design of LWR fuel has assumed that failure will occur if centreline melting takes place. This analysis is performed for the maximum linear heat generation rate throughout the core, including all hot spots and hot channel factors, and it normally accounts for the effects of burn-up and fuel composition (e.g. Pu or Gd content) on the melting point. Section 4.2 of U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan uses term "Overheating of Fuel Pellets". For both normal operation and anticipated transients, centreline melting is not permitted.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraph 3.40 of IAEA SSG-52 [16] recommends that in operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent fuel melting, when appropriate provisions and uncertainties are considered. For design basis accidents and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties - see the following supplementary information of IAEA SSG-52 [16], Annex I in Table I-1 for PCI: "*In the assessment of the peak fuel temperatures for operational states, the following burnup dependent phenomena need to be addressed: changes in fuel thermal conductivity and diffusivity and in the thermal conductance of the pellet-cladding gap, fuel densification, fuel swelling, accumulation of fission products in the fuel pellets, fission gas release in the free volumes of the fuel rods and any other changes in the pellet microstructure. Owing to irradiation effects, the fuel melting temperature varies as a function of fuel burnup and, thus, needs to be determined using representative irradiated fuel samples.*"

Information demonstrating this recommendation will be needed for future SCW-SMR conceptual design.

Relation of requirement with safety-related behaviour: Criterion 11 is not related to any challenges, issues and most significant phenomena. However, it is related to WP2 no. 18 phenomenon (see Table 11), which affects the fuel rod.

Knowledge judgement of phenomenon:

WP2 phenomenon no. 18: FULFILLED

Comment: WP2 phenomenon no. 18 (P_WP2_18) has $RR = 0.51$ ($IL_w = 0.71$ and $KL_w = 0.44$), which is the 17th highest value. P_WP2_18 is a medium important phenomenon (12th rank), the knowledge level is known (5th rank). As P_WP2_18 is judged medium important phenomenon and it is partially known, the R&D needs were judged as low level 3 priority (L3, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion fuel melting [10].

Justification: D5.2 [6] provide information on overheating of fuel pellets WP2 PIRT analysis phenomenon no. 18 (P_WP2_18):

- Section 2.3.3.1 WP2 - PIRT analysis of safety-related knowledge gaps: "*18. Overheating of Fuel Pellets - fuel failure due to overheating;*"
- Annex 2 – PIRT analysis results, WP2:
 - Short description of the item or notes: "*fuel failure due to overheating*"
 - Rationale of PR by the proposer of the item: "*The design criterion is that no fuel centreline melting is allowed for normal operation and AOOs*"
 - Rationale of KR by the proposer of the item: "*Please specify!*"

Criterion 12: Linear heat generation rate (LHGR) limits [10]

Linear heat generation rate (LHGR) is a measure of the heat output per unit length of a fuel rod. LHGR limits are the most limiting of (1) thermal-hydraulic, (2) loss-of-coolant, and (3) thermal mechanical limits discussed in Criterion 9 and Criterion 10.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraphs 3.23 and 3.27 of IAEA SSG-52 [16] provide recommendation on LHGR limits. Information needs to be provided that the core power is controlled globally and locally using the means of reactivity control in such a way that the peak linear heat generation rate of each fuel rod does not exceed the specified limits anywhere in the core. Also, information is needed for the maximum linear heat generation rate, which is one of the specific thermohydraulic design limits.

Relation of requirement with safety-related behaviour: Criterion 12 is related to I5_WP4 issue.

Knowledge judgement of issue:

I5_WP4: FULFILMENT IS IMPROBABLE

Comment: The reactor core refuelling strategy (I5_WP4 issue) is intricately related to the LHGR, which directly influences core power distribution and overall reactor safety. In general, knowledge of LHGR is sufficient. However, LHGR is a function of core height and specific LHGR should be established as recommended in 3.27 of IAEA SSG-52 [16]. At present, studies are done for pre-conceptual core design and it is expected that design changes will be needed. D5.2 [6] provides the following information on linear heat rate: "*Another issue to be solved is the determination of refuelling strategy because of the requirements for linear heat rate profile. As a consequence of large differences in enrichment and fuel assembly (FA) power, the burn-up of the FAs can differ*"

largely as well." See also Table 9, I5_WP4. No information has been provided if this issue will be solved in the frame of this project.

Criterion 13: RIA cladding failure [10]

Description: In many countries, the cladding failure limit for reactivity initiated accident (RIA) is based on the Nuclear Regulatory Commission (NRC) standard review plan, which suggests a maximum radially averaged fuel enthalpy of 170 cal/g for BWRs and the departure from nucleate boiling (DNB) criterion for PWRs. It should be noted that DNB criterion is not applicable to SCW-SMR reactor. Also, based on RIA experiments at CABRI and Nuclear Safety Research Reactor (NSRR) with fuel rods at a burn-up of approximately 50 GWd/t or higher, an assessment of the adequacy of this limit appeared desirable.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraphs 3.42 and 3.75 of IAEA SSG-52 [16] provide recommendations on RIA cladding failure. Information should be provided on the evaluation of cladding deformation to determine the potential for cladding failure (e.g. burst or rupture) and any resulting release of fission products from the fuel. In the event that fuel cladding failures during an RIA cannot be prevented, the dispersal of molten fuel particles should not challenge the ability to cool the core.

Relation of requirement with safety-related behaviour: Criterion 13 is not related to any challenges, issues and most significant phenomena. However, it is related to WP2 PIRT phenomenon no. 19 (see Table 11), which affects the fuel rod cladding.

Knowledge judgement of phenomenon:

WP2 phenomenon no. 19: FULFILMENT PROBABLE

Comment: WP2 phenomenon no. 19 (P_WP2_19) has RR = 0.57 ($IL_w = 0.71$ and $KL_w = 0.38$), which is the 14th highest value. It is medium important phenomenon (12th rank), the knowledge level is known (11th rank). As P_WP2_19 is judged medium important phenomenon and it is partially known, the R&D needs were judged medium level 2 priority (M2, see Table 11).

Justification: D5.2 [6] provides information on cladding rupture, which is WP2 PIRT analysis phenomenon no. 19 (P_WP2_19):

- Section 2.3.3.1 WP2 - PIRT analysis of safety-related knowledge gaps: "19. **Cladding rupture** - *Cladding rupture due to fast heating rate,*"
- Annex 2 – PIRT analysis results, WP2, phenomenon no. 19:
 - Short description of the item or notes: "*Cladding rupture due to fast heating rates*"
 - Rationale of PR by the proposer of the item: "*A cladding rupture temperature correlation must be used in the LOCA emergency core cooling system (ECCS) analysis*"
 - Rationale of KR by the proposer of the item: "*Please specify!*"

D5.2 [6] does not provide any information on RIA cladding failures.

Criterion 14: Fuel fragmentation and fuel dispersal [10]

Description: Fuel fragmentation refers to situations where fuel cladding breaks into pieces and fuel dispersal to situations where fuel particles escape from the cladding following a rupture. Fuel fragmentation is traditionally considered to exist only in conjunction with highly energetic events such as the reactivity-initiated accidents and can also occur during the loss-of-coolant accident (LOCA). In Europe, around 220 cal/g is used for fresh fuel and 200 cal/g is used for irradiated

fuel. An international industry working group led by the Electric Power Research Institute (EPRI), suggested a value of around 240 cal/g for fresh and low burn-up fuel.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraphs 3.35, 3.72 and 3.73 of IAEA SSG-52 [16] provide recommendations on fuel fragmentation and fuel dispersal. Information should be provided that the dispersal of fuel fragments in the coolant is prevented for accident conditions involving the ballooning and rupture of the cladding.

For LWRs information on the effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be provided. Also, information on the possible effects of the dispersal of fuel particles on doses to workers and on core coolability is needed. Finally, it must be shown that in the event of flow blockage or other consequences of fuel dispersal and fuel coolant interaction as a result of fuel cladding failure (e.g. in a reactivity-initiated accident) the ability to cool the core is not endangered.

Relation of requirement with safety-related behaviour: Criterion 14 is not related to any challenges, issues and most significant phenomena. However, it is related to WP2 PIRT phenomenon no. 19 (see Table 11), which affects the fuel rod cladding.

Knowledge judgement of phenomenon:

WP2 phenomenon no. 19: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 19 (P_WP2_19) has RR = 0.57 ($IL_w = 0.71$ and $KL_w = 0.38$), which is the 14th highest value. It is a medium important phenomenon (12th rank), the knowledge level is partially known (11th rank). As P_WP2_19 is judged a medium important phenomenon and it is partially known, the R&D needs were judged medium level 2 priority (M2, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion fuel fragmentation, cladding embrittlement [10].

Justification: D5.2 [6] provides information on cladding rupture, which is WP2 PIRT analysis phenomenon no. 19:

- Section 2.3.3.1 WP2 - PIRT analysis of safety-related knowledge gaps: "19. **Cladding rupture - Cladding rupture due to fast heating rate;**"
- Annex 2 – PIRT analysis results, WP2, phenomenon no. 19:
 - Short description of the item or notes: "*Cladding rupture due to fast heating rates*"
 - Rationale of PR by the proposer of the item: "*A cladding rupture temperature correlation must be used in the LOCA emergency core cooling system (ECCS) analysis*"
 - Rationale of KR by the proposer of the item: "*Please specify!*"

D5.2 [6] does not provide any information on fuel fragmentation and fuel dispersal.

Criterion 15: Non-LOCA cladding embrittlement/temperature [10]

Description: Certain non-LOCA accidents are analysed to estimate radiological doses to the public and to demonstrate that coolability of the core is maintained. For accidents like the PWR locked rotor accident, DNB is used to indicate cladding failure for dose calculations, and the peak cladding temperature of 1480 °C is sometimes used to demonstrate coolability (this limit was

taken from early data estimates of the fuel failure boundary for LOCA conditions (1480 °C and 17% of cladding thickness oxidised by metal-water reaction).

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraph 3.73 of IAEA SSG-52 [16] provide recommendations on peak cladding temperature design limit. Information needs to be provided, that this limit is defined (and justified) in such a way that cladding oxidation will not cause excessive cladding embrittlement or accelerate uncontrollably during accident conditions.

Relation of requirement with safety-related behaviour: Criterion 15 is related to C5_HPLWR challenge (related to high peak cladding temperature), and WP3 PIRT analysis phenomenon no. 14.

Knowledge judgement of challenge:

C5_HPLWR: FULFILMENT IS IMPROBABLE

Comment: C5_HPLWR is related to further analyses of peak cladding temperature. In the HPLWR report [19] it is stated: "*The peak cladding temperature of this design is obviously exceeding the target of 630°C, as mentioned in Chapter 2.1, as the evaporator and first superheater coolant temperatures are already exceeding this limit at BOC and the first superheater peak coolant temperature is exceeding it even by far at EOC. Typically, we need to account for about 20°C to 30°C peak cladding surface temperature in excess of the peak coolant temperature, as predicted by Monti [21] for fresh fuel.*" At the time of preparing D5.2 [6], the SCW-SMR pre-conceptual design has been studied and it is expected that further design changes will be needed. Finally, safety analyses for non-LOCA should be performed after conceptual design of SCW-SMR to demonstrate that the Criterion 15 is fulfilled.

Justification: D5.2 [6] does not provide any additional information on non-LOCA cladding embrittlement or cladding temperature, except under C5_HPLWR challenge and indirectly in Table 2 of D5.2 [6]: "*Table 2: Plant states and preliminary set target parameters for design basis conditions [10]*" (see Table 20). For SCWR of Japan design [5] maximum cladding surface temperature (MCST) has been used as criterion for core design. The MCST was restricted in order to avoid oxidation corrosion of the cladding and ensure the fuel integrity. For core design much information is provided in D5.2 [6], but this is not related to cladding temperature limit during non-LOCA accidents.

Knowledge judgement of phenomenon:

WP3 phenomenon no. 14: FULFILMENT IS PROBABLE

Comment: WP3 phenomenon no. 14 (P_WP3_14) has $RR = 0.66$ ($IL_w = 1.0$ and $KL_w = 0.47$), which is the 8th highest value. It is high important phenomenon (1th rank), the knowledge level is known (10th rank). As P_WP3_14 is judged high important phenomenon, but knowledge is known, the R&D needs were judged as low level 2 priority (L2, see Table 12).

Justification: D5.2 [6] provide information on non-LOCA cladding temperature in WP3 PIRT analysis phenomenon no. 14:

- Section 2.3.3.2 WP3 - PIRT analysis of safety related knowledge gaps:
"*The list of the 21 relevant thermal-hydraulics related safety issues (the detailed list with short explanation of the issues and with literature references can be found in Annex 3):*
....
Allowable maximum cladding temperature during normal operation (WP2-WP3) in the SCW-SMR,"

- Annex 3 – PIRT analysis results, WP3, phenomenon no. 14:
 - Short description of the item or notes: "*Based on previous coupled TH-Reactor physics calculations, the maximum cladding temperature might reach unacceptably high values from cladding material point of view (WP2) in normal steady-state operation.*"
 - Rationale of the item: "*Alternative core configurations are under investigation with TH-Reactor physics models to reduce the maximum cladding temperature. Continuous negotiation between WP2-WP3 is needed to find an optimum solution.*"
 - Figure of merit: "*Fuel cladding temperature (T_c)*"

Criterion 16: LOCA cladding embrittlement [10]

Description: For zircaloy tubes, it was found that cladding would not become embrittled enough to fragment if the peak cladding temperature remained below 1200 °C and the total oxidation did not exceed 17% of the cladding thickness prior to transient based on the Baker-Just oxidation correlation.

Compliance of design – requested information: Information provided in D5.2 [6] is insufficient to judge the compliance. Paragraph 3.73 of IAEA SSG-52 [16] provide recommendations on peak cladding temperature design limit. Information needs to be provided, that this limit is defined (and justified) in such a way that cladding oxidation will not cause excessive cladding embrittlement or accelerate uncontrollably during accident conditions.

Relation of requirement with safety-related behaviour: Criterion 16 is related to P2_WP2, which is PIRT analysis WP2 no. 10 phenomenon (P_WP2_10).

Knowledge judgement of most significant phenomenon:

P2_WP2 (WP2 phenomenon no. 10 in Table 11): FULFILMENT IS PROBABLE

Comment: P2_WP2 is called "*Resistance of cladding materials under LOCA*". Knowledge judgment assumes that the SCW-SMR candidate materials, proposed for cladding, will be acceptable. If not, design change will be needed, leading to judgment FULFILMENT IMPROBABLE. Also, in that case new criterion should be proposed for SCW-SMR candidate material proposed for cladding.

WP2 phenomenon no. 10 (P_WP2_10) has RR = 0.94 ($IL_w = 0.90$ and $KL_w = 0.19$), which is the 2th highest value. It is high important phenomenon (1th rank), the knowledge level is very limited knowledge (22th rank).

As P_WP2_10 is judged high important phenomenon and it is very limited knowledge, the R&D needs were judged high level 1 priority (H1, see Table 11).

Justification: D5.2 [6] provide information on resistance of cladding materials under LOCA conditions SCWR, which is WP2 PIRT analysis phenomenon no. 10:

- Section 2.3.3.1 WP2 - PIRT analysis of safety related knowledge gaps: "**10. Resistance of cladding materials under LOCA conditions in SCWRs** - *The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases;*"
- Section 2.3.3.1 WP2 - PIRT analysis of safety related knowledge gaps: "*It is worth to mention here that the identified phenomena are valid for different plant conditions, some*

of them apply only for LOCA condition (as it is mentioned in name of item No 10.), most of them apply for all conditions."

- Annex 2 – PIRT analysis results, WP2, phenomenon no. 10:
 - Short description of the item or notes: "*The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases*"
 - Rationale of PR by the proposer of the item: "*Large scale LOCA and subsequent cooling (quenching) effects may be detrimental, in large scale, to the integrity of the irradiated and aged cladding.*"
 - Rationale of KR by the proposer of the item: "*Studied on present LWR materials, but not so much on SS or Alloy 800H and even less on materials exposed for a long time in SCW + n-irradiation.*"

Justification: D5.2 [6] does not provide any information on LOCA cladding embrittlement.

Criterion 17: Blowdown/seismic/transportation loads [10]

Description: Most countries follow the safety criteria as per NUREG-0800, SRP 4.2, Appendix A which require core coolability and control rod insertability to be assured under the combined seismic and LOCA loads.

Compliance of design – requested information: Paragraphs 3.37 and 3.77 of IAEA SSG-52 [16] provide recommendations on blowdown/seismic/transportation loads (note that blowdown is the initial phase of LOCA event).

Information is requested to demonstrate that fuel rods and fuel assemblies are designed to withstand handling loads during transport, storage, installation and refuelling operations. Finally, information is also needed on reactor core structures and components structural integrity across all plant states, enduring damage mechanisms like mechanical or flow-induced vibrations, fatigue, debris effects, thermal, hydraulic, and mechanical loads (e.g., LOCAs and seismic events), and chemical and irradiation effects, including radiation-induced growth.

NUREG-0800, SRP4.2, App. A [39] provides criteria used by the U.S. NRC staff responsible for reviewing fuel system design. In background it is explained that earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly and that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. Input for fuel assembly structural analysis are from primary coolant system and reactor internals structural analysis. Input from LOCA response analysis includes motions of core structures and transient pressure differences. For seismic analysis, nonlinear responses use primary system motions for the safe-shutdown earthquake (SSE). Requested information for review are also analytical methods used in performing structural response analyses and uncertainty allowances.

Relation of requirement with safety-related behaviour: Criterion 17 is not related to any challenges, issues and most significant phenomena.

Knowledge judgement of most significant phenomenon: NOT JUDGED

As Criterion 17 is not related to any challenges, issues and most significant phenomena, no judgement could be done. In general, complete or sufficient information is available for fuel (safety) criterion blowdown/seismic/transportation loads, as in [10] the following is stated: "*Safety*

criteria in this area are not directly affected by the new design changes. Considering the fact that compliance with criteria is demonstrated analytically, methods used to analyse the seismic/LOCA event should be well verified and validated."

Justification: D5.2 [6] does not provide any information on structural analyses of the seismic/LOCA event.

Criterion 18: Assembly hold-down force [10]

Description: LWR fuel assemblies are equipped with hold-down springs in the top piece. They have to provide sufficient forces to prevent the fuel assembly lift-off due to hydraulic loads during normal operation and AOOs [10]. Safety criteria are usually defined following NUREG-0800, SRP 4.2, App. A - vertical lift-off forces must not unseat the lower fuel assembly tie plate from the fuel support structure.

Compliance of design – requested information: Paragraph 3.55 of IAEA SSG-52 [16] provides recommendations on assembly hold-down force. Hydraulic effects in fuel assemblies (primarily in the thermohydraulic design of the fuel assembly) need to provide information on addressing fuel assembly lift-off. Annex II of IAEA SSG-52 [16] (as explained in paragraph 3.38) describes important items that need to be addressed within the fuel rod design, fuel assembly, reactivity control assembly, neutron source assembly and hydraulic plug assembly. Section II-1 of IAEA SSG-52 [16] provide recommendation on hold-down systems. Information is requested that fuel assembly hold-down systems for PWRs are designed to withstand the hydrodynamic loads.

Relation of requirement with safety-related behaviour: Criterion 18 is not related to any challenges, issues and most significant phenomena.

Knowledge judgement of most significant phenomenon: NOT JUDGED

In general, complete or sufficient information is available for fuel (safety) criterion assembly hold-down force [10]. It is explained that guide tube growth is correlated to the fast neutron fluence and hydrogen pickup. Thus, to ensure acceptable guide tube corrosion and hydrogen pickup, guide tube design and material must be selected adequately.

Justification: D5.2 [6] does not provide any information on fuel assembly hold-down systems.

Criterion 19: Fretting wear [10]

Description: The fretting wear phenomenon depends on fuel design of course but in reactor grid-to-rod fretting wear will depend on various parameters, including the cross flows signature, which is plant dependent and difficult to quantify. As a consequence, compliance with the fretting wear limit (typically 10% of the cladding thickness) is checked posteriori, through post-irradiation examination.

In [10] the following it is also stated: "*Spacer grid structural tests, control rod structural and performance tests, fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping), fuel assembly hydraulic flow tests and endurance tests (lift forces, control rod wear, vibration, fuel rod fretting) are the necessary tests to determine if a specific fuel assembly design is sensitive (or not) to the fretting wear phenomena. To simulate the effect of irradiation those tests should account for spacer spring relaxation.*"

Compliance of design – requested information: Paragraphs 3.39, 3.55-3.57, 3.166 and 4.5 of IAEA SSG-52 [16] provide recommendations on fretting wear. Additionally, Annex II of IAEA SSG-52 [16] (as explained in paragraph 3.38) describes important items that need to be addressed within the design of the fuel rod, fuel assembly, reactivity control assembly, neutron source assembly and hydraulic plug assembly.

The design should ensure that fuel rods and fuel assemblies are reliable throughout their lifetime. Information is also requested regarding prevention of grid-to-rod fretting (for LWRs).

Hydraulic effects in fuel assemblies (primarily in the thermohydraulic design of the fuel assembly) need to provide information on addressing grid-to-rod fretting. Information need to be provided that fuel assembly is designed to withstand mechanical stresses as a result of vibration and fretting wear of fuel rods (grid-to-rod fretting for LWRs), that for all applicable plant states the design of fuel rods and fuel assemblies the mechanical safety aspect of vibration and fretting damage is considered (damage should not affect the overall performance of the fuel assembly and its support structure). Fretting wear may have an impact on the post-irradiation behaviour of the fuel rods and the fuel assemblies.

Finally, design qualification requires that the performance of the fuel assemblies should be compared against design criteria identified for phenomena such as fretting wear, oxidation, hydriding, crud buildup and bowing of fuel assemblies.

Section II-1 of IAEA SSG-52 [16] also provide recommendations on design aspects of fuel rods, including the following aspects related to fretting wear:

- for fuel rod cladding design aspect is fuel rod vibration and wear (i.e. grid-to-rod fretting wear for LWRs);
- reactivity control assembly design aspect is the effect of fretting wear on cladding resistance;
- for hydraulic plug assembly the design aspect is the effect of fretting wear on guide tube resistance.

Relation of requirement with safety-related behaviour: Criterion 19 is not related to any challenges, issues and most significant phenomena. However, it is fretting wear WP2 PIRT analysis phenomenon no. 22 (see Table 11).

Knowledge judgement of phenomenon: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 22 (P_WP2_22) has RR = 0.41 ($IL_w = 0.56$ and $KL_w = 0.43$), which is the 21th highest value. It is medium (lower third) important phenomenon (23th rank), the knowledge level is partially known (6th rank). As P_WP2_22 is judged medium important phenomenon and it is partially known, the R&D needs were not needed (NN, see Table 11).

In general, complete or sufficient information is available for fuel (safety) criterion fretting wear [10]. It is explained that guide tube growth is correlated to the fast neutron fluence and hydrogen pickup. Thus, to ensure acceptable guide tube corrosion and hydrogen pickup, guide tube design and material has to be selected adequately.

Justification: D5.2 [6] provide information on fretting wear, which is WP2 PIRT analysis phenomenon no. 22 (P_WP2_22):

- Section 2.3.3.1 WP2 - PIRT analysis of safety related knowledge gaps: "22. **Fretting Wear** - *It can occur on the fuel rod cladding surfaces in contact with the spacer grids.*"

- Annex 2 – PIRT analysis results, WP2, phenomenon no. 22:
 - Short description of the item or notes: *"It can occur on the fuel rod cladding surfaces in contact with the spacer grids"*
 - Rationale of PR by the proposer of the item: *"The design criterion is to provide sufficient support to limit fuel rod vibration and cladding fretting wear"*
 - Rationale of KR by the proposer of the item: *"Please specify!"*

Criterion 20: Coolant activity [10]

Description: In most countries, limits are specified in the plant's technical specifications on the concentration of I-131 (sometimes also of Cs-137) in the primary coolant. The limits are typically around $2 \cdot 10^9$ Bq/t.

Compliance of design – requested information: Paragraph 3.39 of IAEA SSG-52 [16] provide recommendation on appropriate parameters to be selected for core monitoring. This selection will depend on the reactor type. An example of parameters is also reactor coolant activity - see the following supplementary information of IAEA SSG-52 [16], Annex I in Table I-1 for coolant: *"Reactor coolant activity is measured by a device belonging to the primary coolant makeup and water cleaning system (for details, see IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [I-2])."*

Relation of requirement with safety-related behaviour: Criterion 20 is not related to any challenges, issues and most significant phenomena.

Knowledge judgement: NOT JUDGED

In general, complete or sufficient information is available for fuel (safety) criterion coolant activity [10]. No change of the limit(s) for coolant activity is expected in conjunction with new design changes [10].

Justification: D5.2 [6] does not provide any information on coolant activity.

Criterion 21: Fuel gap activity [10]

Description: Fuel gap activity is of interest for accident scenarios that may result in cladding failures but that do not involve melting of the fuel. It determines the potential release of fission products to the primary circuit. Fixed values of release to the gap, like up to 10% of the rod inventory for noble gases and 1-6% for halogens and alkali metals, are assumed in safety analyses. Fission product release to the gap is found to increase at high burn-up and may require the modification of assumptions about gap activity that are used in safety analyses.

Compliance of design – requested information: Paragraph 3.162 of IAEA SSG-52 [16] provides recommendation on defective fuel. Fuel rod failures can affect ease of access, work scheduling and worker dose for plant operations personnel. Operation of a reactor core with defective fuel rods should stay within the radiochemical requirements as defined by the limit on coolant radionuclide activity included in the technical specifications for the plant. To clarify terminology and provide additional background, for defective fuel the following supplementary information is provided in Annex I, Table I-1 of IAEA SSG-52 [16]:

"The iodine spiking phenomenon after plant transients has received particular attention in safety evaluations. For particular pre-accident conditions, the occurrence of this phenomenon may increase the radiological consequences of the postulated accident. One approach in safety evaluation is to specify a limit to the amount of iodine activity allowed in the reactor coolant after

plant transients. The behaviour of leaking fuel rods in design basis accidents (e.g. loss of coolant accidents, reactivity initiated accidents and steam generator tube rupture) may be specific and may need to be assessed individually. For example, margins for a loss of coolant accident might not be affected by the presence of leaking fuel because conservative assumptions are specified as requirements for evaluating radiological consequences. Design limits for reactivity initiated accidents might not be affected by the presence of a limited number of leaking fuel rods, although it is recognized that a leaking fuel rod has a lower capability to withstand reactivity initiated accident loadings and, consequently, has a higher probability to cause limited fuel coolant interaction."

Relation of requirement with safety-related behaviour: Criterion 21 is not related to any challenges, issues and most significant phenomena.

Knowledge judgement: NOT JUDGED

Complete or sufficient information is not available for fuel (safety) criterion gap activity [10].

Justification: D5.2 [6] does not provide any information on fuel gap activity or safety analyses to determine the potential release of fission products to the primary circuit.

Criterion 22: Source term [10]

Description: There are no safety criteria directly associated with source terms. Various assumptions are made for the analysis of accident scenarios.

Compliance of design – requested information: Paragraphs 2.16-2.18 of IAEA SSG-2 [17] cover aspects of the analysis of releases of radioactive material up to and including the determination of the source term for releases to the environment for AOOs and accident conditions. According to IAEA glossary [40] source terms is the "*the amount and isotopic composition of radioactive material released (or postulated to be released) from a facility*". To evaluate the source term, it is necessary to identify the sources of radiation, to determine the inventories of radionuclides that are produced and to know the mechanisms by which radioactive material can travel from the source through the nuclear power plant and be released to the environment. The source term has to be evaluated (using the appropriate simulation codes) for operational states and accident conditions.

Aspects associated with the determination of the source term are introduced in several paragraphs in IAEA SSG-2 [17] to remind readers of the applicability of the general rules to this specific application (3.26, 3.35, 6.13, 7.4, 7.5, 7.44).

Relation of requirement with safety-related behaviour: Criterion 22 is related to I2_SMR issue.

Knowledge judgement: NOT JUDGED

Comment: I2_SMR issue deals with small core inventory. Complete or sufficient information is not available for fuel (safety) criterion radioactive source term and high burn-up [10]. According to [10] it is considered unlikely that new design changes or high burn-up will have a significant effect on source terms or core melt progression.

Justification: Source terms are based on measured releases from irradiated fuel, tested under accident conditions, in combination with assumptions or analyses of the effects of retention or enhancement during the course of an accident sequence [10].

D5.2 [6] provide the following information with respect to special safety-related features of SMRs: "*the smaller core size and the resulting low thermal power leads to reduction of source term, i.e. smaller possible releases in accident conditions;*". This is then listed as SMR special issue (see also I2_SMR of Table 6): "*small core inventory → small decay heat (new safety systems – possibility of air cooling for residual heat removal function) and small source terms (lower release possible to environment, resulting in decreasing or elimination of emergency preparedness zones);*"

It is logical that small core inventory will lead to lower source terms. No further information in this respect is provided in D5.2 [6].

Criterion 23: Burn-up [10]

Description: Licensed burn-up limits depend on the fuel type and fuel vendor. Licensed limits may refer to local burn-up levels and/or rod average burn-up levels and/or assembly average burn-up levels. For example, a generic limit for maximum fuel assembly average burn-up of 52 GWd/t exists in France for UO₂ and MOX fuel.

Report [10] identifies burn-up-dependent design criteria as well as key design parameters or performance measures used to demonstrate compliance with the criteria. Research showed that most current design limits could be retained when supported with data at the targeted burn-up levels. Criteria relating to the response of fuel to RIAs or to LOCAs require a more complex evaluation.

Compliance of design – requested information: For example, paragraphs 3.70, 3.71, 3.73, 3.101, 3.113, 3.123, and 4.9 of IAEA SSG-52 [16] provide recommendations related to burnup. Information on assessment and justification of fuel discharge burnup limits should be provided (limits depend on the performance of the fuel rods and fuel assembly, and on the fuel management approach). Information should be provided also for the following items:

- For DBAs and design extension conditions (DEC) without significant fuel degradation, in the case the fuel failure mechanisms and fuel failure limits are burnup dependent, irradiation effects on cladding and fuel properties should be considered in the experimental studies and should be incorporated into the analyses.
- The reactivity initiated accident failure limit is expected to depend on the fuel burnup and cladding material.
- The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions, including fuel burnup.
- In the design of reactivity control devices, due account of the effects of irradiation, such as burnup, shall be taken.
- Limits and set points should consider the impacts of fuel burnup.
- Measurements of the local power at different positions in the core should also consider the impact of the spatial power distribution changes due to core burnup effects.
- In-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors should be used to justify the specified maximum burnup limit for a new design.

Relation of requirement with safety-related behaviour: Criterion 23 is related to C5_HPLWR issue and WP4-1 phenomena no. 7 (P_WP4-1_7) and no. 9 (P_WP4-1_9).

Knowledge judgement:

C5_HPLWR issue: FULFILMENT IS IMPROBABLE

Comment: C5_HPLWR challenge deals with further analyses required for high peak cladding temperature, low fuel burn-up and high hot channel factors. Complete or sufficient information is not available for fuel (safety) criterion high burn-up [10]. Regarding the research the following is stated: "*However, the working group also considers that there is a need for further research to (a) experimentally verify the validity of safety criteria for high burn-up, in particular for burn-up levels beyond those currently licensed, and (b) further develop and benchmark the analytical models used in the safety design studies to comply with the high burn-up safety criteria.*"

WP4-1, phenomenon no. 7 (P_WP4-1_7): NOT JUDGED

Comment: P_WP4-1_7 phenomenon is called "*Material composition, also including burn-up*". This is a computation methodology parameter for Monte Carlo calculation, which was not judged the most significant and no information on knowledge and research needs was provided in D5.2 [6].

WP4-1, phenomenon no. 9 (P_WP4-1_9): NOT JUDGED

Comment: P_WP4-1_9 phenomenon is called "*Operational history for burn-up calculations*". This is a computation methodology parameter for Monte Carlo calculation, which was not judged the most significant and no information on knowledge and research needs was provided in D5.2 [6].

Justification: D5.2 [6] provide some information on burn-up, related to analyses and modelling, without information on knowledge and research needs:

- Section 3 Safety features of SCW-SMR, Safety-related features derived from HPLWR technology – challenges: "*high peak cladding temperature, low fuel burn-up and high hot channel factors require further analyses;*" (see C5_HPLWR in Table 4)
- Annex 4 – PIRT analysis results, WP4, phenomenon no. 7:
 - Descriptive name of reactor physical phenomena: "*Material composition, also including burn-up*".
 - Short description of the item or notes: "*Appropriate modeling of material composition is necessary*"
 - Reference for the item: "*n.a.*"
 - Operating status: "*Normal operation*"
- Annex 4 – PIRT analysis results, WP4, phenomenon no. 9:
 - Descriptive name of reactor physical phenomena: "*Operational history for burn-up calculations*".
 - Short description of the item or notes: "*For accurate burn-up calculations, the operational history of the reactor is necessary.*"
 - Reference for the item: "*n.a.*"
 - Operating status: "*Normal operation*"

5 Conformance of future SCW-SMR technology to Gen IV goals

Generation IV nuclear energy systems goals have three purposes [20]:

- *"First, they serve as the basis for developing criteria to assess and compare the systems in the technology roadmap.*
- *Second, they are challenging and stimulate the search for innovative nuclear energy systems—both fuel cycles and reactor technologies.*
- *Third, they will serve to motivate and guide the R&D on Generation IV systems as collaborative efforts get underway."*

Eight goals were established, of which three are safety and reliability goals [20]:

- *"Safety and Reliability–1 Generation IV nuclear energy systems operations will excel in safety and reliability.*
- *Safety and Reliability–2 Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*
- *Safety and Reliability–3 Generation IV nuclear energy systems will eliminate the need for offsite emergency response."*

These three safety and reliability (SR) goals [20] no. 1, 2 and 3 are labelled and named as SR1 Operational Safety and Reliability, SR2 Core Damage and SR3 Offsite Emergency Response, respectively. They focus on safe and reliable operation, improved accident management and minimization of consequences, investment protection, and essentially eliminating the technical need for off-site emergency response. The technology roadmap for Gen IV nuclear energy systems briefly discusses them:

"Safety and reliability are essential priorities in the development and operation of nuclear energy systems. Nuclear energy systems must be designed so that during normal operation or anticipated transients safety margins are adequate, accidents are prevented, and off-normal situations do not deteriorate into severe accidents. At the same time, competitiveness requires a very high level of reliability and performance. There has been a definite trend over the years to improve the safety and reliability of nuclear power plants, reduce the frequency and degree of off-site radioactive releases, and reduce the possibility of significant plant damage. Looking ahead, Generation IV systems will face new challenges to their reliability at higher temperatures and other anticipated conditions. Generation IV systems have goals to achieve high levels of safety and reliability through further improvements. The three safety and reliability goals continue the past trend and seek simplified designs that are safe and further reduce the potential for severe accidents and minimize their consequences. The achievement of these ambitious goals cannot rely only upon technical improvements, but will also require systematic consideration of human performance as a major contributor to the plant availability, reliability, inspectability, and maintainability."

When looking the above design goals on adequate safety margins, prevention of accidents, that off-normal situations do not deteriorate into severe accidents, and very high level of reliability and performance, it can be judged that in general could be fulfilled by IAEA SSR-2/1 (Rev. 1) [1] specific safety requirements: Requirement 42 for safety analyses requires to ensure adequate safety margins in demonstrating compliance with acceptance criteria for all plant states; Requirement 20 deals with prevention of severe accidents by strengthening the design basis for the plant; and Requirement 23 deals with reliability of items important to safety, commensurate with their safety significance.

The highest level of requirements considered in this report are three major GIF safety and reliability goals identified in D5.1 [4]. As little detailed design information is available about future SCW-SMR conceptual design, it is difficult to examine the level of support for the conformance to safety goals. The information provided by D5.2 [6] in Section 2.2.1.1 Literature review of SCWR reactor designs, under part "High Performance Light Water Reactor - Design and Analyses" was: *"Concerning the development of the safety systems, the requirements known from the GIF initiative and EUR requirements have been used as general guidelines. The highest level of requirements considered were the three major GIF safety goals: (1) the high level of safety and reliability; (2) the low probability of reactor core damage; and (3) the elimination of the need for off-site emergency response. The concept of design in depth (DiD) is applied for all plant states. The preliminary targets for different design basic conditions can be seen in Table 2."*

Preliminary targets for different design basic conditions of HPLWR in Table 2 of [19] are also shown in Table 20 of this report. As information provided in D5.2 [6] is very general and specific for HPLWR and very little detailed information exists for future SCW-SMR conceptual design, the applicability of these GIF goals (described also in D5.1 [4]) to future SCW-SMR conceptual design is discussed rather to make judgment of design compliance to safety requirements and criteria.

In the following the three safety and reliability goals [20] are discussed.

5.1 SR1 Operational Safety and Reliability

According to HPLWR design and analyses document [19] the first safety and reliability (SR1) goal *"aims at increasing operational safety by reducing: the number of events, equipment problems, human performance issues that can initiate accidents or cause them into more severe accident. It also aims at achieving increased nuclear energy systems reliability that will benefit their economics. Appropriate requirements and robust designs are needed to advance such operational objectives and to support the demonstration of safety that enhance public confidence. [...]"*

Generation IV nuclear energy systems must continue to promote the highest levels of safety and reliability by adopting established principles and best practices developed by the industry and regulators to enhance public confidence, and by employing future technological advances. The continued and judicious pursuit of excellence in safety and reliability is important to improving economics."

Discussion on applicability to future SCW-SMR conceptual design: Based on above information from HPLWR design and analyses document [19] it is judged that SR1 goal is applicable also to future SCW-SMR conceptual design.

5.2 SR2 Core Damage

Following D5.1 [4], second safety and reliability (SR2) goal looks at significantly reducing both the probabilistic and deterministic aspects of DEC-A scenarios whereas the third safety and reliability goal looks at preventing DEC-B scenarios. Reformulating SR2 goal:

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

SR2 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.

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.

Discussion on applicability to future SCW-SMR conceptual design: Based on the information in D5.1 [4] report, "*no conclusion on core damage values (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.*", it is judged that SR2 goal is applicable to future SCW-SMR conceptual design.

5.3 SR3 Offsite Emergency Response

As for the third safety and reliability (SR3) goal reformulated objective:

DEC-B scenarios shall be eliminated.

This safety principle is even stronger for the new plants as reflected in [21], etc., where only so-called Large and/or Early Release Frequency (LoERF) categories have to be eliminated. Instead, 4th Gen. NPPs shall eliminate all *significant* radioactive releases. 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) D5.1 [4] report states, 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. 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.

As for (2), D5.1 [4] report states, that the second way of achieving inherent safety linked to DEC-B is that **DEC-A scenarios are practically eliminated**.

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.

Discussion on applicability to future SCW-SMR conceptual design: For SR3 it is stated in HPLWR [17] that defence in depth is one of the important principles in all safety concepts of current reactors. It was required to be applied also for the HPLWR [17]: "*Accordingly, safe operation of the power plant shall be ensured by the following measures:*

Normal operation (DBC1) controlled by operating systems:

- *Conservative design, reliability, availability*

- *Proven technology, quality assurance*

Operational occurrences (DBC2, >10⁻²/year) controlled by control and limitation features:

- *Surveillance, diagnostics*
- *Inherent safety, nuclear stability*

Design basis accidents (DBC3/4, >10⁻⁵/year) controlled by safety systems

- *Redundancy, train separation*
- *Protection against internal and external hazards*
- *Qualification against accident conditions*
- *Automation (<30 min)*
- *Autarchy*

Design Extension 1 (DEC1): Multiple failure scenarios

- *Diversified systems*
- *Design against external event loads*

Design Extension 2 (DEC2): Severe Accidents

- *Mitigative features*
- *Prevention of energetic consequences which could lead to large early containment failure"*

In design of HPLWR [17] it was found that as long as the cladding material and its strength and corrosion resistance have not yet been defined, acceptance criteria for the maximum fuel and cladding temperature can hardly be quantified. The preliminary targets for different design basis conditions can be seen in Table 20 (compare also to OECD/NEA Criteria 12, 14, 15, and 16 for maximum linear heat generation rate, fuel fragmentation and fuel dispersal, non-LOCA maximum cladding temperature and LOCA maximum cladding temperature, respectively).

Table 20. Plant states and preliminary set target parameters for design basis conditions for HPLWR [19]

Reactor state or event	Limited parameter	Limit	Reason for limit
Normal operation DBC1	Maximum cladding temperature	630 °C	Cladding creep or corrosion Fission gas release, pressure inside fuel rod
	Maximum linear heat rate	39 kW/m	
Operational occurrence, DBC2	Maximum cladding temperature	850 °C	Cladding buckling collapse Fuel melting
	Maximum linear heat rate	2800 °C	
Design basis accidents, DBC3/4	Maximum cladding temperature	1200 °C	Cladding embrittlement due to oxidation Fuel fragmentation and dispersion
	Maximum radially averaged pellet enthalpy	963 kJ/kg	

Based on the above it is judged that SR3 goal is applicable to future SCW-SMR conceptual design.

6 Conformance of future SCW-SMR technology to WENRA safety objectives for new NPPs

6.1 Grounding of the WENRA safety objectives on IAEA SF-1 principles

In March 2008, the Reactor Harmonization Working Group (RHWG) of WENRA made a proposal for a study on new reactors [43]. The study includes identification and review of the existing relevant documentation on new reactors (IAEA and NEA documents, national regulations and other relevant documents) and from this documentation select a justified set of safety objectives, safety principles and specific considerations which are relevant for new reactors. The safety objectives for new reactors have been defined on the basis of a systematic investigation of the Fundamental Safety Principles (SF-1 document issued 2006 by the IAEA [26]). The following IAEA SF-1 fundamental safety principles have been found to be especially relevant for improvement of safety of new reactors:

IAEA SF-1 Principle 5 "Optimization of protection" [26]: in line with this principle it was proposed [43] (See objectives O1 to O4 and O6):

"the safety of new reactors will have to be improved as far as reasonably achievable starting from the design stage, with due consideration given to insights gained from:

- *experience feedback from existing reactors;*
- *deterministic and probabilistic safety assessments;*
- *state-of-the art technologies, analysis methodologies and techniques;*
- *results of safety research."*

IAEA SF-1 Principle 8 "Prevention of accidents" [26]: in implementing this principle it was proposed [43] (See objectives O1 to O5 and O7):

"the defence-in-depth concept remains the key safety approach for new reactors. Therefore, for new reactors, strengthening of the implementation of the concept has to be aimed for:

- *reinforcement of each level of the defence in depth concept,*
- *improvement of the independence between the levels of defence in depth."*

IAEA SF-1 Principle 6 "Limitation of risks to individuals" [26] and Principle 7 "Protection of present and future generations" [26]: in line with these two principles it was proposed [43] (See objectives O2, O3 and O6):

"the radiological and non-radiological impact of normal and abnormal operation, potential accidents and decommissioning activities will have to be reduced at the design stage."

IAEA SF-1 Principle 3 "Leadership and management for safety": in line with this principle it was proposed [43] (See objectives O5 and O7):

"due consideration has to be given to safety management from an early stage coherently with security requirements."

As it can be seen, especially relevant for improvement of safety of new reactors were IAEA SF-1 [26] principles no. 3 and no. 5 to 8, which were used to ground the proposed seven WENRA safety objectives O1 to O7 for new reactors [43].

In D5.1 [4] it is stated that WENRA objectives and key positions gathered in [41] apply to all existing and new plants, and hence are classified as fully applicable to ECC-SMART (note: also IAEA SF-1 principles were classified as fully applicable). WENRA safety objectives for new NPPs are upper level principles that should be applicable to all types of reactors. It was expected that

the safety objectives would be applicable to SMRs, too. However, by WENRA Reactor Harmonisation Working Group (RHWG) it was considered beneficial to study them from the point of view of SMRs to confirm the applicability and to identify potential questions that would benefit from further study or guidance [42]. Considering the wide variety of SMR designs currently being developed, it is not easy to identify safety features that would be common to all designs. Instead, the RHWG considered generic design features and features of different deployment schemes of the new concepts that may affect the applicability of the Safety Objectives. Because little detailed design information was available about many of the innovative designs that were being considered or proposed, it was difficult to examine the level of support for the claims of enhanced safety. Therefore, the purpose of report [42] was not to evaluate the proposed concepts or the validity of the safety claims but to evaluate the applicability of the WENRA Safety Objectives to SMRs.

6.2 SMR features

Claimed features and potential challenges of SMR concepts as judged by WENRA RHWG [42] (the reader may compare below claimed features and potential challenges of SMRs concepts to WNA special safety-related features of SMRs identified in D5.2 [6], which are listed in Table 6):

- *"low power resulting in reduced decay heat and smaller activity inventory per reactor module;*
- *novel measures to enhance safety, such as "inherently safe" fuel, coolant material with enhanced safety features, natural circulation as a main means of heat removal, wide use of passive safety systems and practical elimination of situations that can lead to early or large releases;*
- *partially different initiating events (e.g. absence of some events like LB-LOCA in integrated designs, but also new potential events like module to module interactions);*
- *long grace periods for operator actions;*
- *challenges for periodic inspections of components in integrated designs where all the main primary component are incorporated inside one single vessel;*
- *unconventional number of physical barriers between fission products and environment;*
- *use of novel fuels (e.g. molten salt, ceramic);*
- *operating concept – high degree of automation and reduced role of operators, remote control, unmanned units, several reactors operated by one operator team;*
- *use of common SSCs between several reactor cores;*
- *unconventional siting: underground, sea-bed based, remote locations lacking basic infrastructure, off-grid locations;*
- *factory-fuelled reactor cores;*
- *focus of initial testing shifted from site to factory; and*
- *emergence of new companies both as developers and as utilities".*

6.3 Applicability of safety objectives to future SCW-SMR conceptual design

In the following the applicability of WENRA safety objectives to future SCW-SMR conceptual design are judged. Some features of SMRs that differ from the present-day reactors have been given in [42] as examples, even if they do not impact the applicability of the objectives. In the following discussion on features of SMRs that differ from the present-day reactors is judged with respect to applicability of objectives to future SCW-SMR conceptual design.

01. Normal operation, abnormal events and prevention of accidents [41]

- *reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation.*
- *reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events.*

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: *"The number of active systems may be reduced (resulting in reduced number of component failures) and materials less prone to failures might be utilised. Some concepts require very little operator invention which helps to reduce the probability of human errors."*

Judgement for SCW-SMR conceptual design: According to D5.2 [6] the SCWR concepts have adopted many passive safety systems to complement active safety systems to enhance the safety performance of the whole reactor concept compared with the current fleet (mostly LWRs) of nuclear reactors. For future SCW-SMR conceptual design it is expected to be further simplified.

- Description [42]: *"Different operational aspects may, at least in the first projects, pose a challenge. New aspects may be, for example:*
 - *higher degree of automation in the plant control and reduced number of operating staff;*
 - *the role of the operating staff may differ from what is traditional in large NPPs;*
 - *one operating team may operate several reactors (potentially from a remote location);*
 - *use of new technologies in plant control and monitoring as well as in condition monitoring (artificial intelligence, diagnostics, robotics...);*
 - *interactions between several coupled reactor modules;*
 - *potential feedback of co-generation/process heat industrial application."*
- Judgement for SCW-SMR conceptual design: In general, the challenges may be relevant, but it is too early to judge the future SCW-SMR conceptual design as operational aspects are not in the scope of ECC-SMART project.

O2. Accidents without core melt [41]

- *ensuring that accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation).*
- *reducing, as far as reasonably achievable,*
 - *the core damage frequency taking into account all types of credible hazards and failures and credible combinations of events;*
 - *the releases of radioactive material from all sources.*
- *providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.*

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: *"However, for those SMR concepts where molten is the normal state of the fuel, the term "core melt" is not meaningful but for example a fuel leakage or failure of*

the heat removal pathways could still cause a release. The idea of the Safety Objective is valid, but the terminology "core melt" needs to be refined depending on the SMR concept."

- Judgement for SCW-SMR conceptual design: For SCW-SMR the terminology "core melt" is valid.

O3. Accidents with core melt [41]

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

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: *"As for O2, the term "core melt" is not fitting for all SMR concepts. However, the Safety Objective should be interpreted to mean "accidents which would lead to large or early releases". Therefore, O3 addresses also possible other scenarios that may lead to large or early release than core melt (e.g. leakage of liquid fuel from a molten salt reactor)."*
- Judgement for SCW-SMR conceptual design: For SCW-SMR the terminology "core melt" is valid as regarding this feature it does not differ from the present-day reactors.

O4. Independence between all levels of defence-in-depth [41]

- *"enhancing the effectiveness of the independence between all levels of defence-in-depth, in particular through diversity provisions (in addition to the strengthening of each of these levels separately as addressed in the previous three objectives), to provide as far as reasonably achievable an overall reinforcement of defence-in-depth."*

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: *"The independence between levels of defence-in-depth, to the extent reasonably practicable, is a key element of ensuring the effectiveness of the defence-in-depth concept and that is applicable independent of the technology used. As noticed in the introduction of this report, the term SMR covers a wide range of designs and there is no universally agreed definition for this term. Considering the wide variety of SMR designs currently being developed [2], it is not easy to identify safety features that are common to all designs. Therefore, discussions on the application of defence-in-depth (DiD), and in particular of the concept of independence between all levels of DiD (Safety Objective O4) to SMRs should be based on particular SMR designs or at least design types."*

- Judgement for SCW-SMR conceptual design: According to D5.2 [6] for the HPLWR, the design of the safety systems basically adheres to the DiD safety principle. Also, in D5.2 [6] it is stated: "*The situation of supercritical water-cooled SMR is somewhat special: it belongs to Gen IV technologies, but – as we could see the similarities with BWR reactors and because of the application of light water as moderator / coolant – it can be considered as advanced light water reactor as well. For example, IAEA-TECDOC-1785 (Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident) [12] summarizes the specific safety features of LWR SMRs and their design features for all defence-in-depth levels.*" Similarly, the application of defence-in-depth is expected for future SCW-SMR conceptual design.

O5. Safety and security interfaces [41]

- "*ensuring that safety measures and security measures are designed and implemented in an integrated manner. Synergies between safety and security enhancements should be sought.*"

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: "*Several SMRs have features enhancing security (e.g. compact integrated design with smaller number of systems needing physical protection and with fewer access points, difficult access due to e.g. underground location, long grace periods and less need for operator actions to reduce the likelihood of the main control room being targeted). On the other hand, some aspects may bring new challenges (e.g. remote operation, having unmanned stations possibly in remote locations or, on the other hand, close to densely populated areas, transportation of modules with loaded core). However, the new features do not affect the applicability of the Safety Objective, they rather confirm the importance of considering both safety and security aspects in an integrated manner.*"
- Judgement for SCW-SMR conceptual design: According to D5.1 [4] just safety has been discussed. Enhanced security by underground location, new challenges like and remote operation and transportation of modules with loaded core seems not feasible for future SCW-SMR conceptual design.

O6. Radiation protection and waste management [41]

- "*reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities:*
 - *individual and collective doses for workers;*
 - *radioactive discharges to the environment;*
 - *quantity and activity of radioactive waste.*"

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: "*Several SMRs may have features that differ from the present-day reactors, but they do not affect the applicability of the Safety Objective. Many SMR concepts feature a compact design with small footprint and minimized building volume. This may result in lesser available space for radiation shielding and may require access routes and working areas closer to radiation sources than in present-day reactors.*"

On the other hand, the need to access the nuclear island during operation might be minimized, there might be a reduced number of components needing maintenance and there might be less activation (of structural materials) by design. SMRs thus have both beneficial and detrimental features regarding radiation protection of workers."

- Judgement for SCW-SMR conceptual design: D5.2 [6] does not provide any information on radiation protection or radioactive waste. In general, the challenges may be relevant, but it is too early to judge the future SCW-SMR conceptual design as operational aspects are not in the scope of ECC-SMART project.

O7. Leadership and management for safety [41]

- *"ensuring effective management for safety from the design stage. This implies that the licensee:*
 - *establishes effective leadership and management for safety over the entire new plant project and has sufficient in house technical and financial resources to fulfil its prime responsibility in safety;*
 - *ensures that all other organizations involved in siting, design, construction, commissioning, operation and decommissioning of new plants demonstrate awareness among the staff of the nuclear safety issues associated with their work and their role in ensuring safety."*

RHWG judgement: This objective is applicable to SMRs.

Discussion of features in [42] that differ from the present-day reactors:

- Description [42]: *"Many SMRs are intended to support other purposes in addition to electricity production. The reactors may be utilised for example for district heating, for small scale electricity production, to produce process heat for industry or for desalination. As a consequence, the companies interested in SMRs may be very different from the traditional users of nuclear energy, which typically have been power companies with electricity production as a core business. The new companies may have very little experience on use of nuclear energy and a wish to outsource as many tasks as possible. ...
Whatever the organisational arrangements are, the Safety Objective is valid. However, in application of requirements, a graded approach should be used."*
- Judgement for SCW-SMR conceptual design: D5.2 [6] does not provide any information on leadership and management for safety. However, IAEA SF-1 [26] Principle 3 Leadership and management for safety and the requirements on leadership and management for safety are also set also in IAEA SSR-2/1 (rev. 1) Requirements 1-3, which must be satisfied. Therefore, WENRA O7 objective is applicable to future SCW-SMR conceptual design.

7 Pre-licensing study results for future SCW-SMR technology

In the following the requirements and criteria related to safety-related behaviour of future SCW-SMR conceptual design are summarized.

According to D5.1 [4] there are four key sources of information concerning international legislation or guidance applicable to nuclear reactors: IAEA, OECD/NEA-sponsored GIF, European Union legal acts and international fora at European level. All these were considered except European Nuclear Safety Directive (Council Directive 2014/87/Euratom [44]), for which in D5.1 [4] it is explained that 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 (Rev. 1) [1], under the so-called “practically eliminated conditions” (see paragraphs 2.11, 4.3, 5.31 and 6.5 of IAEA SSR-2/1 (Rev. 1) [1]). 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.

In this pre-licensing study the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 are related to the challenges, issues and gaps in knowledge regarding the safety-related behaviour of SCW-SMR as identified in WP5.2 and to the available level of detail. This has to be done separately for the following international legislation or guidance:

- IAEA relevant standards (SF-1 [26], SSR-2/1 (Rev. 1) [1])
- OECD/NEA fuel safety criteria (23 criteria) [10]
- OECD/NEA GIF safety and reliability goals (judgement of applicability to SCW-SMR) [20]
- WENRA safety objectives for new NPPs (judgement of applicability to SCW-SMR) [41]

7.1 Pre-licensing study results for IAEA standards

As explained in Section 3.1, compliance to IAEA SF-1 [26] objective and principles has not been judged due to insufficient detailed information on future SCW-SMR conceptual design. Also, the SF-1 safety objective and safety principles form the basis for deriving the IAEA SSR-2/1 (Rev. 1) [1] safety function requirements that must be met for the nuclear power plant, as well as the safety design criteria. This means that when future SCW-SMR conceptual design will be assessed against safety requirements established in IAEA safety standards, the compliance to IAEA SF-1 [26] objective and principles could be judged.

7.1.1 Compliance of design – requested information

Table 21 summarizes the pre-licensing study results with the list of the recommendations in the IAEA specific safety guides SSG-2 [17], SSG-52 [16], SSG-53 [37] and SSG-56 [36], based on which requested information needed to judge the compliance of future SCW-SMR conceptual design to the selected safety criteria and requirements of IAEA SSR-2/1 (Rev. 1) [1] was determined. As explained above, the selected requirements of IAEA SSR-2/1 (Rev. 1) [1] are Requirements 42-58 (for more detailed information see Sections 3.2.3 through 3.2.6).

In general, recommendations of IAEA SSG-2 (Rev. 1) [17] are applicable for Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1]. In this report Chapter 5 recommendations are used, as SCW-SMR conceptual design is not in the scope of the project and therefore also deterministic analyses of PIEs could not be performed due to not yet developed future SCW-SMR conceptual design (note

that in the frame of ECC-SMART project some pre-conceptual studies of a SCW-SMR have been performed using computer codes, see D3.3 [7]): regarding the use of computer codes, for verification paragraphs 5.14-5.20 are applicable, for validation paragraphs 5.21-5.39 are applicable, for qualification of input data paragraph 5.40 is applicable and for documentation of computer codes paragraphs 5.41-5.43 are applicable.

For Requirements 43-58 of IAEA SSR-2/1 (Rev. 1) [1], related to the design of reactor core and associated features, reactor coolant systems, and containment structure and containment system, all relevant paragraphs with recommendations are specified. It should be noted that SSGs besides recommendations for design of specific systems provide also recommendations to fulfil requirements for the management of safety, principal technical requirements and general design requirements, established in Sections 3, 4 and 5 of IAEA SSR-2/1 (Rev. 1) [1], respectively (see Table 13, Requirements 1-42).

SSG-52 [16] provides recommendations on the design of the reactor core to meet the requirements established in IAEA SSR-2/1 (Rev. 1) [1]. General considerations for safe core design based on requirements for the management of safety, principal technical requirements and general design requirements are described in Section 2. Specific considerations for design of reactor core and associated features to fulfil Requirements 43-46 are given in Section 3, while Section 4 provides recommendations on the qualification and testing of the SSCs of the reactor core (not in the scope of this report).

SSG-56 [36] on the design of the reactor coolant system and associated systems for nuclear power plants provides recommendations on how to meet the requirements of IAEA SSR-2/1 (Rev. 1) [1]. It is organized in a bit different manner than SSG-52 [16]. For this pre-licensing study relevant Sections are 3, 4, 5 and 7. Section 3 provides generic design recommendations for the reactor coolant system and associated systems. Section 4 provides design recommendations for the different heat transfer chains and generic considerations for the ultimate heat sink. Section 5 provides supplementary design recommendations that are specific to reactor coolant systems for PWR, BWR and PHWR technologies. Finally, Section 7 provide supplementary design recommendations that are specific to associated systems for BWR.

SSG-53 [37] on the design of the reactor containment and associated systems for nuclear power plants provides recommendations on how to meet the requirements of IAEA SSR-2/1 (Rev. 1) [1]. It describes the safety functions relating to the containment and its associated systems and addresses the main requirements of SSR-2/1 (Rev. 1) [1] that need to be considered in Section 2. Recommendations on the design basis of the containment structure and its components and associated systems are provided in Section 3. Specific recommendations for the design of the containment and its associated systems are provided in Section 4. Finally, Section 5 covers tests and inspections and provides recommendations for commissioning tests and for in-service tests and inspections (not in the scope of this report).

Table 21. Recommendations of IAEA specific safety guides with description how to judge the compliance of future SCW-SMR conceptual design to the selected safety criteria and requirements of IAEA SSR-2/1 (Rev. 1)

Systems/item	SSR-2/1 (Rev. 1)	Requirement (paragraphs)	SSG no.	Paragraphs
Safety analysis	Req. 42	Safety analysis of the plant design (5.71–5.76)	SSG-2	5.14-5.43
Reactor core and associated features	Req. 43	Performance of fuel elements and assemblies (6.1–6.3)	SSG-52	2.1-2.27, 3.4-3.76
	Req. 44	Structural capability of the reactor core	SSG-52	3.77-3.88
	Req. 45	Control of the reactor core (6.4–6.6)	SSG-52	3.89-3.102
	Req. 46	Reactor shutdown (6.7–6.12)	SSG-52	3.103-3.127
Reactor coolant systems	Req. 47	Design of reactor coolant systems (6.13–6.16)	SSG-56	3.2-3.143, 5.4-5.16
	Req. 48	Overpressure protection of the reactor coolant pressure boundary	SSG-56	3.117-3.120, 5.29-5.41
	Req. 49	Inventory of reactor coolant	SSG-56	5.18-5.19
	Req. 50	Cleanup of reactor coolant (6.17)	SSG-56	7.2-7.4
	Req. 51	Removal of residual heat from the reactor core	SSG-56	7.6-7.16
	Req. 52	Emergency cooling of the reactor core (6.18–6.19)	SSG-56	7.17-7.26
Containment structure and containment system	Req. 53	Heat transfer to an ultimate heat sink (6.19A–6.19B)	SSG-56	4.22-4.27
	Req. 54	Containment system for the reactor	SSG-53	2.2-2.3, 3.1-3.90
	Req. 55	Control of radioactive releases from the containment (6.20–6.21)	SSG-53	2.4, 4.90-4.130
	Req. 56	Isolation of the containment (6.22–6.24)	SSG-53	4.154-4.166
	Req. 57	Access to the containment (6.25–6.26)	SSG-53	4.174-4.180
	Req. 58	Control of containment conditions (6.27–6.30)	SSG-53	4.131-4.150

7.1.2 Relation of IAEA SSR-2/1 (Rev. 1) standard requirements and criteria with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR – knowledge judgement

Results for relation of IAEA SSR-2/1 (Rev. 1) [1] standard requirements and criteria with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR are shown in Table 22. Judgement is done regarding the knowledge, indicating the research needs.

Table 22. Relation of relevant IAEA SSR-2/1 (Rev. 1) requirements and criteria with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR

Requirement/criterion	Challenge/issue/significant phenomenon	Knowledge judgement
Requirement 42	P1_WP4	FULFILMENT IS PROBABLE
Requirement 43	C1_SCWR	FULFILMENT IS PROBABLE
	C2_SCWR	FULFILMENT IS PROBABLE
	C3_SCWR	FULFILMENT IS PROBABLE
	C5_SCWR	FULFILMENT IS PROBABLE
	C6_SCWR	FULFILMENT IS PROBABLE
Requirement 44	C4_HPLWR	FULFILMENT IS IMPROBABLE
Requirement 45	C1_HPLWR	FULFILMENT IS PROBABLE
	C2_HPLWR	FULFILMENT IS PROBABLE
	C3_HPLWR	FULFILMENT IS PROBABLE
	C2_BWR	FULFILMENT IS PROBABLE
Requirement 46	N.A.	N.A.
Requirement 47	I2_WP3	FULFILMENT IS IMPROBABLE
Requirement 48	N.A.	N.A.
Requirement 49	N.A.	N.A.
Requirement 50	N.A.	N.A.
Requirement 51	N.A.	N.A.
Requirement 52	N.A.	N.A.
Requirement 53	N.A.	N.A.
Requirement 54	C4_BWR	FULFILLED
Requirement 55	I2_SMR	FULFILLED
Requirement 56	N.A.	N.A.
Requirement 57	N.A.	N.A.
Requirement 58	I4_WP2	FULFILMENT IS PROBABLE

In the following comments given in Section 3.2 for each judgement regarding the knowledge, indicating the research needs, are summarized for selected IAEA SSR-2/1 (Rev. 1) [1] requirements and criteria related to challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR.

Requirement 42 partly related to P1_WP4 (see Table 9): FULFILMENT IS PROBABLE

Comment: Partial relation apply to the information on WP4 PIRT analysis of safety related gaps for normal operation/all conditions and rod ejection accidents (REA) D5.2 [6]. It is judged that lacking materials or material composition, boundary conditions and other data is expected to be available after completed SCW-SMR conceptual design.

Requirement 43 related to C1_SCWR (see Table 3): FULFILMENT IS PROBABLE

Comment: novel manufacturing processes carry the risk of a first-of-its-kind design and may introduce new technical challenges. However, significant R&D activities have been and are currently underway to qualify novel manufacturing processes (in particular additive manufacturing of safety-classified components) with issuing of associated code cases and nuclear design code evolutions. Thus, FULFILMENT is highly likely.

Requirement 43 related to C2 SCWR (see Table 3): FULFILMENT IS PROBABLE

Comment: The aggressive chemical effects of SCW will require experimental testing and demonstration.

Requirement 43 related to C3 SCWR (see Table 3): FULFILMENT IS PROBABLE

Comment: SCW as a coolant in highly radiative environment requires significantly high level of experimental testing and demonstration before adopting.

Requirement 43 related to C5 SCWR (see Table 3): FULFILMENT IS PROBABLE

Comment: The most significant knowledge gap related to the fuel technology of the SCWR is the change in material properties of cladding material as a function of the irradiation damage - this knowledge gap should be closed during the development of SCW-SMR concept.

Requirement 43 related to C6 SCWR (see Table 3): FULFILMENT IS PROBABLE

Comment: Maximum "diametral strain" estimated for the pressure tube of the Canadian SCWR concept after 75 years of full power operation would require validation. In-core irradiation experiments are required at SCWR conditions to validate the presented estimation.

Requirement 44 related to C4 HPLWR (see Table 4): FULFILMENT IS IMPROBABLE

Comment: The studies in the frame of HPLWR indicated areas of design optimization for stress and deformation analyses of the RPV, the major reactor internals and of the assembly boxes [19]. Also, the following it is stated [19]: "*A great challenge has been to design the internals of the pressure vessel such that they can freely expand under the increased temperature differences, but to seal each component against the others such that cold feedwater cannot penetrate into the hot steam. It is still an open question, how close these sealing systems can be built, and how durable they will be stay under long term operation.*"

From studies of HPLWR [20] it can be concluded that design changes may also be needed of future SCW-SMR conceptual design, before structural capability of the reactor core will be demonstrated.

Requirement 45 related C1 HPLWR (see Table 4): FULFILMENT IS PROBABLE

Comment: As written in D5.2 [6] this is feature of HPLWR. For HPLWR reactor analysis of the core power distribution iterative analysis is required [17]: "*Like with boiling water reactors, the core power distribution is significantly influenced by the coolant density distribution which is responsible for neutron moderation, together with the moderator water inside the water boxes and between the assembly boxes. The coolant density, in turn, is decreasing by the fissile power so that both, the coolant and moderator heat up and the core power distribution must be analysed iteratively to yield a consistent, coupled solution.*"

Requirement 45 related C2 HPLWR (see Table 4): FULFILMENT IS PROBABLE

Comment: As written in D5.2 [6] in Section 2.2.1.1 Literature review of SCWR reactor designs, High Performance Light Water Reactor - Design and Analyses:

"Another important issue is the problem of coolant and moderator flow stability, also well-known from BWR reactors. The preliminary analysis performed for HPLWR indicates possible Xenon oscillation instabilities that must be considered during the further design."

A Technology Roadmap for Gen IV nuclear energy systems [20] states that important SCW technology gaps are in the areas of SCWR safety, including power-flow stability during operation.

In accordance with [20] an SCWR safety research activity is recommended for power-flow stability assessments.

Requirement 45 related C3 HPLWR (see Table 4): FULFILMENT IS PROBABLE

Comment: The preliminary analysis performed for HPLWR indicates possible Xenon oscillation instabilities that must be considered during the further design (see D5.2 [6]). To understand the behaviour of possible Xenon oscillation instabilities a full core model is required [17] "*In order to fully understand the behaviour of the HPLWR 3-pass core against xenon oscillations, a full-core model applying fast computational methods will be required.*"

Requirement 45 related C2 BWR (see

Table 5): FULFILMENT IS PROBABLE

Comment: D5.2 [6] in Section 2.2.2 'Safety features of BWR reactors' provides the following information:

"As a consequence of the in-core boiling, the moderator density and reactor power – and also the neutron flux – is much lower in the upper part of the core, resulting in a larger core volume for given thermal power (compared to PWRs). The reactor vessel volume is much larger than in case of pressurized water reactors due to the steam separation facilities and the internal jet pumps."

Regarding knowledge in large changes in neutron flux, the paper [45] concludes the following: "*Instabilities that may occur during the BWR operation constitute a widely known problem in the scientific community addressed for more than thirty years. A great deal of literature is available including data and models. The analysis of the phenomena involved requires a multidisciplinary approach comprising various areas like transient thermal-hydraulics, neutron kinetics, fuel behavior including in-core fuel management, instrumentation, plant control and monitoring, and detailed knowledge of plant features. The use of large thermal-hydraulic system codes should be promoted in this area, provided 3-D neutron kinetics modeling and suitable numerics and specific user guidelines are implemented.*"

Requirement 47 related I2 WP3 (see Table 8): FULFILMENT IS IMPROBABLE

Comment: D5.2 [6] provide the following information: "*There are no accepted formulation of design and safety concept. The detailed thermal hydraulic analysis cannot start until their availability. This issue has been regarded also partially solved (see below).*"

Based on the "Work Package Periodic Report M18" of WP3, the main achievements of WP3 are so far:

1. *The SCW-SMR design concept has been developed based on the concept of HPLWR;"*.

However, when checking SCW-SMR concept in [31], information suggests that mainly core design has been proposed: "*Based on the concept of the High Performance Light Water Reactor (HPLWR), an innovative concept of a small modular reactor is presented, which shall have smaller peak cladding temperatures than the HPLWR due to better coolant mixing, and which shall enable a passive residual heat removal by natural convection. The core is designed with horizontal fuel assemblies"*.

Requirement 54 related C4_BWR (see

Table 5): FULFILLED

Comment: C4_BWR challenge is more description than challenge. It seems that special containment arrangement with drywell and wetwell makes it more complicated.

Requirement 55 related I2_SM R (see Table 6): FULFILLED

Comment: I2_SM R is more special safety feature rather than issue. Namely, in the D5.2 [6] the following statement is done:

"The World Nuclear Association (WNA) summarizes the special safety-related features of SMRs as follows:

- ...
- *the smaller core size and the resulting low thermal power leads to reduction of source term, i.e. smaller possible releases in accident conditions;"*)

Requirement 58 related I4_WP2 (see Table 7): FULFILMENT IS PROBABLE

Comment: In the report D5.2 [6] the following information is provided: *"Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature – The radiolysis processes in SCW are not well-known. Moreover, there is not much information on the effect of pressure (p) [MPa] and temperature (T) [K] in the electrochemical behaviour of SCW. This "material issue" is under investigation as well during the project by the WP2 partners".*

At the time being it is difficult to judge that design changes of the containment are needed, as this is not yet in the scope of the project. Nevertheless, it should be kept in mind that radiolysis of the water in the core and radiolysis of the water in the sump or in the suppression pool should be taken into account in the identification of sources of combustible gases in the containment (see paragraph 4.133 of IAEA SSG-53 [37].)

Requirement 42 related to PIRT WP2 and WP3 phenomena (see Tables 11 and 12):

Comment: In the following the results of knowledge judgement for most important phenomena are summarized regarding R&D needs.

The accuracy of incorporated phenomenological models in the deterministic computer codes should be known and traceable. Each computer code should ensure (through verification and validation) that the models for important phenomena are appropriate. IAEA SSG-2 (Rev. 1) [17] proposes to assess the accuracy of individual computer codes which also include identification of the important phenomena in the supporting experimental data and expected plant behaviour. In the frame of ECC-SMART project the important phenomena were identified through PIRT analyses. The reader can refer to Tables 17 and 18, showing categorization of PIRT WP2 and PIRT WP3 phenomena (the details are shown in Tables 11 and 12 for PIRT for WP2 and WP3, respectively). The summary of the results is shown in the following:

H1 - High priority level 1 (high important phenomenon and very limited knowledge) R&D phenomena from WP2:

- P_WP2_8 (i.e. most significant phenomenon P1_WP2): Radiolysis processes,
- P_WP2_10 (i.e. most significant phenomenon P2_WP2): Resistance of cladding materials under LOCA conditions SCWR.

H2 - high priority level 2 (high important phenomenon, which is partially known) R&D phenomena from WP2:

- P_WP2_4 (i.e. most significant phenomenon P3_WP2): Pellet cladding interaction;
- P_WP2_17 (i.e. significant phenomenon P4_WP2): Overheating of the Cladding;
- P_WP2_14 (i.e. significant phenomenon P5_WP2): Irradiation Assisted Stress Corrosion Cracking (IASCC);
- P_WP2_12: Oxide release from the cladding surface by spalling;
- P_WP2_6: Changes in the mechanical properties of the materials produced by ageing and/or irradiation.

H1 - High priority level 1 (high important phenomenon and very limited knowledge) R&D phenomena from WP3:

- P_WP3_5 (i.e. most significant phenomenon P1_WP3): Heat and mass transfer along corroded and rough surfaces.

H2 - high priority level 2 (high important phenomenon, which is partially known) R&D phenomena from WP3:

- P_WP3_4 (i.e. most significant phenomenon P4_WP3): Turbulent heat and mass transfer in water under supercritical pressure conditions,
- P_WP3_6 (i.e. most significant phenomenon P3_WP3): Deterioration of heat transfer (DHT),
- P_WP3_2 (i.e. most significant phenomenon P2_WP3): Heat transfer in water under supercritical pressure conditions,
- P_WP3_7 (i.e. most significant phenomenon P5_WP3): Transition from supercritical to subcritical pressure,
- P_WP3_9: Natural circulation of water under super - or sub -critical pressure conditions,
- P_WP3_16: CHF near the critical point.

As can be seen, in PIRT for WP2 seven very important phenomena for R&D are identified (including all five most significant phenomena identified in D5.2 [6] PIRT analysis, see Table 7) and in PIRT for WP3 also seven very important phenomena for R&D are identified (including all five most significant phenomena identified in D5.2 [6] PIRT analysis, see Table 8).

7.2 Pre-licensing study results for OECD/NEA fuel safety criteria

Table 23 summarizes the results OECD/NEA fuel safety criteria [10] related to the challenges, issues and phenomena of safety-related behaviour of SCW-SMR (for more details refer to Section 4).

Table 23. Relation of relevant OECD/NEA fuel safety criteria with challenges (C), issues (I) and phenomena (P) of safety-related behaviour of SCW-SMR

Criterion	Challenge/issue/phenomenon	Knowledge judgement
Criterion 1	P_WP3_18	FULFILMENT IS PROBABLE
Criterion 2	I2_WP4	FULFILMENT IS PROBABLE
Criterion 3	N.A.	NOT JUDGED
Criterion 4	I3_SMR	FULFILMENT IS PROBABLE
Criterion 5	N.A.	NOT JUDGED
Criterion 6	P_WP2_21	FULFILMENT IS PROBABLE
Criterion 7	P_WP2_15	FULFILMENT IS PROBABLE
Criterion 8	N.A.	NOT JUDGED
Criterion 9	P_WP2_4	NOT JUDGED
Criterion 10	P3_WP2 (i.e. P_WP2_4)	FULFILMENT PROBABLE
	P_WP3_21	FULFILLED
Criterion 11	P_WP2_18	FULFILLED
Criterion 12	I5_WP4	FULFILMENT IS IMPROBABLE
Criterion 13	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 14	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 15	C5_HPLWR	FULFILMENT IS IMPROBABLE
	P_WP3_14	FULFILMENT PROBABLE
Criterion 16	P2_WP2 (i.e. P_WP2_10)	FULFILMENT IS PROBABLE
Criterion 17	N.A.	NOT JUDGED
Criterion 18	N.A.	NOT JUDGED
Criterion 19	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 20	N.A.	NOT JUDGED
Criterion 21	N.A.	NOT JUDGED
Criterion 22	I2_SMR	NOT JUDGED
Criterion 23	C5_HPLWR	FULFILMENT IS IMPROBABLE
	P_WP4-1_7	NOT JUDGED
	P_WP4-1_9	NOT JUDGED

In the following comments given in Section 4 for each judgement regarding the knowledge, indicating the research needs, are summarized for OECD/NEA fuel safety criteria [10] related to challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR.

Criterion 1 related to P_WP3_18: FULFILMENT IS PROBABLE

Comment: WP3 phenomenon no. 18 has $RR = 0.69$ ($IL_w = 0.83$ and $KL_w = 0.33$), which is the 7th highest value. It is not so important phenomenon (14th rank), however the knowledge level is rather low (3rd rank). As it is not very important phenomenon, the research needs don't have the highest priority.

Criterion 2 related to I1_WP4: FULFILMENT IS PROBABLE

Comment: Regarding I1_WP4 issue on negative temperature reactivity coefficients the document on fuel safety criteria [10] states that although the reactivity coefficients may be affected, the effects of new fuel design changes are not considered to affect the corresponding safety criteria themselves.

Criterion 4 related to I3_SMR: FULFILMENT IS PROBABLE

Comment: Regarding I3_SMR issue on higher enrichment, care should be taken using enrichments between 5-10 wt%. Namely, the physics of criticality begins to change as enrichments reach 6 wt% and beyond. Also, the possibility of recriticality during accidents should be addressed. Also, there is rising international interest in increasing fuel burnup limits and fuel cycle length may require fuel enrichment above 5% and high reactivity/high suppression core designs [51]. Activities to verify that extended enrichment fuel is well understood and that existing design limits are still valid for fuel of 5-8% enrichment, have been proposed by OECD/NEA WGFS [51]. If enrichment will exceed 8%, design changes may be potentially needed or new design limits should be set.

Criterion 6 related to P_WP2_21: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 21 (WP2_21) has $RR = 0.62$ ($IL_w = 0.67$ and $KL_w = 0.28$), which is the 11th highest value. It is medium important phenomenon (16th rank), however the knowledge level is partial (19th rank). As WP2_21 phenomenon is judged medium important phenomenon and knowledge is partially known, the R&D needs were judged as medium level 2 priority, (M2, see Table 11).

Criterion 7 related to P_WP2_15: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 15 (WP2_15) has $RR = 0.46$ ($IL_w = 0.57$ and $KL_w = 0.38$), which is the 19th highest value. WP2_15 phenomenon is medium important phenomenon (21th rank), the knowledge level is partial (11th rank). As it is judged medium important phenomenon and knowledge is partially known, the R&D needs were judged as beneficial (B, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion cladding oxidation and hydriding [10].

Criterion 10 related to P3_WP2 (i.e. P_WP2_4): FULFILMENT PROBABLE

Comment: WP2 phenomenon no. 4 (P_WP2_4, see Table 11) has $RR = 0.75$ ($IL_w = 0.81$ and $KL_w = 0.28$), which is the 3rd highest value. It is high important phenomenon (7th rank), the knowledge level is partial (19th rank). As P_WP2_4 is judged high important phenomenon, the R&D needs were judged as high level 2 priority (H2, see Table 11).

Criterion 10 related to P_WP3_21: FULFILLED

Comment: WP3 phenomenon no. 21 (P_WP3_21) has $RR = 0.32$ ($IL_w = 0.78$ and $KL_w = 0.67$), which is the 19th highest value. It is high important phenomenon, slightly above the limit 0.75 for medium (16th rank), the knowledge level is known (2nd rank). As P_WP3_21 is judged high (lower third) important phenomenon and it is known, the R&D needs were judged as beneficial (B, see Table 12).

Criterion 11 related to P_WP2_18: FULFILLED

Comment: WP2 phenomenon no. 18 (P_WP2_18) has $RR = 0.51$ ($IL_w = 0.71$ and $KL_w = 0.44$), which is the 17th highest value. P_WP2_18 is medium important phenomenon (12th rank), the knowledge level is known (5th rank). As P_WP2_18 is judged medium important phenomenon and it is partially known, the R&D needs were judged as low level 3 priority (L3, see Table 11).

Criterion 12 related to I5_WP4: FULFILMENT IS IMPROBABLE

Comment: Reactor core refuelling strategy (I5_WP4 issue) is intricately related to the LHGR, which directly influences core power distribution and overall reactor safety. In general knowledge of LHGR is sufficient. However, LHGR is a function of core height and specific LHGR should be

established as recommended in 3.27 of IAEA SSG-52 [16]. At present, studies are done for pre-conceptual core design and it is expected that design changes will be needed. D5.2 [6] provide the following information on linear heat rate: "*Another issue to be solved is the determination of refuelling strategy because of the requirements for linear heat rate profile. As a consequence of large differences in enrichment and fuel assembly (FA) power, the burn-up of the FAs can differ largely as well.*" See also Table 9, I5_WP4. No information has been provided if I5_WP4 issue will be solved in the frame of this project.

Criterion 13 related to P_WP2_19: FULFILMENT PROBABLE

Comment: WP2 phenomenon no. 19 (P_WP2_19) has RR = 0.57 ($IL_w = 0.71$ and $KL_w = 0.38$), which is the 14th highest value. It is medium important phenomenon (12th rank), the knowledge level is partially known (11th rank). As WP2_19 is judged medium important phenomenon and it is partially known, the R&D needs were judged medium level 2 priority (M2, see Table 11).

Criterion 14 related to P_WP2_19: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 19 (P_WP2_19) has RR = 0.57 ($IL_w = 0.71$ and $KL_w = 0.38$), which is the 14th highest value. It is medium important phenomenon (12th rank), the knowledge level is partially known (11th rank). As P_WP2_19 is judged medium important phenomenon and it is partially known, the R&D needs were judged medium level 2 priority (M2, see Table 11).

Complete or sufficient information is not available for fuel (safety) criterion fuel fragmentation, cladding embrittlement [10].

Criterion 15 related to C5_HPLWR: FULFILMENT IS IMPROBABLE

Comment: C5_HPLWR is related to further analyses of peak cladding temperature. In the HPLWR report [19] it is stated: "*The peak cladding temperature of this design is obviously exceeding the target of 630°C, as mentioned in Chapter 2.1, as the evaporator and first superheater coolant temperatures are already exceeding this limit at BOC and the first superheater peak coolant temperature is exceeding it even by far at EOC. Typically, we need to account for about 20°C to 30°C peak cladding surface temperature in excess of the peak coolant temperature, as predicted by Monti [21] for fresh fuel.*" At the time of preparing D5.2 [6], the SCW-SMR pre-conceptual design has been studied and it is expected that further design changes will be needed. Finally, safety analyses for non-LOCA should be performed after conceptual design of SCW-SMR to demonstrate that the Criterion 15 is fulfilled.

Criterion 15 related to P_WP3_14: FULFILMENT IS PROBABLE

Comment: WP3 phenomenon no. 14 has RR = 0.66 ($IL_w = 1.0$ and $KL_w = 0.47$), which is the 8th highest value. It is high important phenomenon (1th rank), the knowledge level is between partial and known (10th rank). As it is judged high important phenomenon and knowledge is partial to known, the research needs have the medium priority.

Criterion 16 related to P2_WP2 (i.e. P_WP2_10): FULFILMENT IS PROBABLE

Comment: P2_WP2 is called "*Resistance of cladding materials under LOCA*". Knowledge judgment assumes that the SCW-SMR candidate materials, proposed for cladding, will be acceptable. If not, design change will be needed, leading to judgment FULFILMENT IS IMPROBABLE. Also, in that case new criterion should be proposed for SCW-SMR candidate material proposed for cladding.

WP2 phenomenon no. 10 (P_WP2_10) has RR = 0.94 ($IL_w = 0.90$ and $KL_w = 0.19$), which is the 2th highest value. It is high important phenomenon (1th rank), the knowledge level is very limited

knowledge (22th rank). As P_WP2_10 is judged high important phenomenon and it is very limited knowledge, the R&D needs were judged high level 1 priority (H1, see Table 11).

Criterion 19 related to P_WP2_22: FULFILMENT IS PROBABLE

Comment: WP2 phenomenon no. 22 (P_WP2_22) has $RR = 0.41$ ($IL_w = 0.56$ and $KL_w = 0.43$), which is the 21th highest value. It is medium (lower third) important phenomenon (23th rank), the knowledge level is partially known (6th rank). As P_WP2_22 is judged medium important phenomenon and it is partially known, the R&D needs were not needed (NN, see Table 11).

In general, complete or sufficient information is available for fuel (safety) criterion fretting wear [10]. It is explained that guide tube growth is correlated to the fast neutron fluence and hydrogen pickup. Thus, to ensure acceptable guide tube corrosion and hydrogen pickup, guide tube design and material has to be selected adequately.

Criterion 22 related to I2_SMR: NOT JUDGED

Comment: I2_SMR issue deals with small core inventory. Complete or sufficient information is not available for fuel (safety) criterion radioactive source term and high burn-up [10]. According to [10] it is considered unlikely that new design changes or high burn-up will have a significant effect on source terms or core melt progression.

Criterion 23 related to C5_HPLWR: FULFILMENT IS IMPROBABLE

C5_HPLWR challenge deals with further analyses required for high peak cladding temperature, low fuel burn-up and high hot channel factors. Complete or sufficient information is not available for fuel (safety) criterion high burn-up [10]. Regarding the research the following is stated: "*However, the working group also considers that there is a need for further research to (a) experimentally verify the validity of safety criteria for high burn-up, in particular for burn-up levels beyond those currently licensed, and (b) further develop and benchmark the analytical models used in the safety design studies to comply with the high burn-up safety criteria.*"

Criterion 23 related to P_WP4-1_7: NOT JUDGED

Comment: P_WP4-1_7 phenomenon is called "*Material composition, also including burn-up*". This is computation methodology parameter for Monte Carlo calculation, which was not judged, is not the most significant phenomenon and no information on knowledge and research needs was provided in D5.2 [6].

Criterion 23 related to P_WP4-1_9: NOT JUDGED

Comment: P_WP4-1_9 phenomenon is called "*Operational history for burn-up calculations*". This is computation methodology parameter for Monte Carlo calculation, which was not judged, is not the most significant phenomenon and no information on knowledge and research needs was provided in D5.2 [6].

7.3 Pre-licensing study results for applicability of Gen IV goals

The highest level of requirements considered in this report are three major GIF safety and reliability goals identified in D5.1 [4]. As little detailed design information is available about future SCW-SMR conceptual design, it is difficult to examine the level of support for the conformance to safety goals. Therefore, the applicability to future SCW-SMR conceptual design has been discussed.

SR1 goal - Operational Safety and Reliability

Discussion on applicability to future SCW-SMR conceptual design (see Section 5.1): Based on information from HPLWR design and analyses document [19] it is judged that SR1 goal is applicable also to future SCW-SMR conceptual design.

SR2 goal - Core Damage

Discussion on applicability to future SCW-SMR conceptual design (see Section 5.2): Based on the information in D5.1 [4] report, "*no conclusion on core damage values (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.*", it is judged that SR2 goal is applicable to future SCW-SMR conceptual design.

SR3 goal - Offsite Emergency Response

Discussion on applicability to future SCW-SMR conceptual design (see Section 5.3): For SR3 is stated in HPLWR [17] that defence in depth is one of the important principles in all safety concepts of current reactors. It was required to be applied also for the HPLWR [17], where measures shall be encused for all design basis conditions (DBC 1 to 4) and design extension conditions (DEC 1 to 2) – see Section 5.3.

7.4 Pre-licensing study results for applicability of WENRA safety objectives for SMRs

In the following discussion on features of SMRs that differ from the present-day reactors [42] is judged with respect to applicability of objectives to future SCW-SMR conceptual design.

O1. Normal operation, abnormal events and prevention of accidents

Description of discussion on features of SMRs that differ from the present-day reactors [42]: "*The number of active systems may be reduced (resulting in reduced number of component failures) and materials less prone to failures might be utilised. Some concepts require very little operator invention which helps to reduce the probability of human errors.*"

Judgement for SCW-SMR conceptual design: According to D5.2 [6] the SCWR concepts have adopted many passive safety systems to complement active safety systems to enhance the safety performance of the whole reactor concept compared with the current fleet (mostly LWRs) of nuclear reactors. For future SCW-SMR conceptual design it is expected to be further simplified.

Description of discussion on features of SMRs that differ from the present-day reactors [42]: "*Different operational aspects may, at least in the first projects, pose a challenge. New aspects may be, for example:*

- *higher degree of automation in the plant control and reduced number of operating staff;*
- *the role of the operating staff may differ from what is traditional in large NPPs;*
- *one operating team may operate several reactors (potentially from a remote location);*
- *use of new technologies in plant control and monitoring as well as in condition monitoring (artificial intelligence, diagnostics, robotics...);*
- *interactions between several coupled reactor modules;*
- *potential feedback of co-generation/process heat industrial application.*"

Judgement for SCW-SMR conceptual design: In general, the challenges may be relevant, but it is too early to judge the future SCW-SMR conceptual design as operational aspects are not in the scope of ECC-SMART project.

O2. Accidents without core melt

Description of discussion on features of SMRs that differ from the present-day reactors [42]: *"However, for those SMR concepts where molten is the normal state of the fuel, the term "core melt" is not meaningful but for example a fuel leakage or failure of the heat removal pathways could still cause a release. The idea of the Safety Objective is valid, but the terminology "core melt" needs to be refined depending on the SMR concept."*

Judgement for SCW-SMR conceptual design: For SCW-SMR the terminology "core melt" is valid.

O3. Accidents with core melt

Description of discussion on features of SMRs that differ from the present-day reactors [42]: *"As for O2, the term "core melt" is not fitting for all SMR concepts. However, the Safety Objective should be interpreted to mean "accidents which would lead to large or early releases". Therefore, O3 addresses also possible other scenarios that may lead to large or early release than core melt (e.g. leakage of liquid fuel from a molten salt reactor)."*

Judgement for SCW-SMR conceptual design: For SCW-SMR the terminology "core melt" is valid as with respect to this feature future SCW-SMR conceptual design does not differ from the present-day reactors.

O4. Independence between all levels of defence-in-depth

Description of discussion on features of SMRs that differ from the present-day reactors [42]: *"The independence between levels of defence-in-depth, to the extent reasonably practicable, is a key element of ensuring the effectiveness of the defence-in-depth concept and that is applicable independent of the technology used."*

As noticed in the introduction of this report, the term SMR covers a wide range of designs and there is no universally agreed definition for this term. Considering the wide variety of SMR designs currently being developed [2], it is not easy to identify safety features that are common to all designs. Therefore, discussions on the application of defence-in-depth (DiD), and in particular of the concept of independence between all levels of DiD (Safety Objective O4) to SMRs should be based on particular SMR designs or at least design types."

Judgement for SCW-SMR conceptual design: According to D5.2 [6] for the HPLWR, the design of the safety systems basically adheres to the DiD safety principle. Also, it is stated [6]: *"The situation of supercritical water-cooled SMR is somewhat special: it belongs to Gen IV technologies, but – as we could see the similarities with BWR reactors and because of the application of light water as moderator / coolant – it can be considered as advanced light water reactor as well. For example, IAEA-TECDOC-1785 (Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident) [12] summarizes the specific safety features of LWR SMRs and their design features for all defence-in-depth levels."* Similarly, the application of defence-in-depth is expected for future SCW-SMR conceptual design.

O5. Safety and security interfaces

Description of discussion on features of SMRs that differ from the present-day reactors [42]: *"Several SMRs have features enhancing security (e.g. compact integrated design with smaller number of systems needing physical protection and with fewer access points, difficult access due to e.g. underground location, long grace periods and less need for operator actions to reduce the likelihood of the main control room being targeted). On the other hand, some aspects may bring new challenges (e.g. remote operation, having unmanned stations possibly in remote locations or, on the other hand, close to densely populated areas, transportation of modules with loaded*

core). However, the new features do not affect the applicability of the Safety Objective, they rather confirm the importance of considering both safety and security aspects in an integrated manner." Judgement for SCW-SMR conceptual design: According to D5.1 [4] just safety has been discussed. Enhanced security by underground location, new challenges like and remote operation and transportation of modules with loaded core seems not feasible for future SCW-SMR conceptual design.

O6. Radiation protection and waste management

Description of discussion on features of SMRs that differ from the present-day reactors [42]:

"Several SMRs may have features that differ from the present-day reactors, but they do not affect the applicability of the Safety Objective.

Many SMR concepts feature a compact design with small footprint and minimized building volume. This may result in lesser available space for radiation shielding and may require access routes and working areas closer to radiation sources than in present-day reactors.

On the other hand, the need to access the nuclear island during operation might be minimized, there might be a reduced number of components needing maintenance and there might be less activation (of structural materials) by design. SMRs thus have both beneficial and detrimental features regarding radiation protection of workers."

Judgement for SCW-SMR conceptual design: D5.2 [6] does not provide any information on radiation protection or radioactive waste. In general, the challenges may be relevant, but it is too early to judge the future SCW-SMR conceptual design as operational aspects are not in the scope of ECC-SMART project.

O7. Leadership and management for safety

Description of discussion on features of SMRs that differ from the present-day reactors [42]:

"Many SMRs are intended to support other purposes in addition to electricity production. The reactors may be utilised for example for district heating, for small scale electricity production, to produce process heat for industry or for desalination. As a consequence, the companies interested in SMRs may be very different from the traditional users of nuclear energy, which typically have been power companies with electricity production as a core business. The new companies may have very little experience on use of nuclear energy and a wish to outsource as many tasks as possible.

...

Whatever the organisational arrangements are, the Safety Objective is valid. However, in application of requirements, a graded approach should be used."

Judgement for SCW-SMR conceptual design: D5.2 [6] does not provide any information on leadership and management for safety. However, IAEA SF-1 [26] Principle 3 Leadership and management for safety and the requirements on leadership and management for safety are also set in IAEA SSR-2/1 (rev. 1) Requirements 1-3, which must be satisfied. Therefore, the WENRA O7 objective applies to future SCW-SMR conceptual design.

8 Conclusions

The requirements and criteria from the relevant documents have been related to the challenges, issues and phenomena gaps in knowledge regarding the safety-related behaviour of future SCW-SMR conceptual design. The requirements considered were design Requirements 42 to 58 of IAEA SSR-2/1 (Rev. 1) [1]. It was demonstrated that these requirements apply to future SCW-SMR conceptual design, where their associated requirements in paragraphs are in a large extent applicable – only one paragraph needs re-evaluation and two new paragraphs were suggested to be added.

The relation of relevant IAEA SSR-2/1 (Rev. 1) [1] requirements and criteria with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR showed that in the majority of cases (11 out of 15) the fulfilment of knowledge gaps is probable provided that further research is needed. In two cases the present pre-conceptual design requires design changes. For Requirement 42 of IAEA SSR-2/1 (Rev. 1) [1] on deterministic safety analyses the results of knowledge judgement for WP2 and WP3 identified phenomena regarding R&D needs are shown.

Relation of relevant OECD/NEA fuel safety criteria [10] with challenges, issues and phenomena of safety-related behaviour of SCW-SMR showed that for 12 out of 26 criteria the fulfilment of knowledge gaps is probable provided that further research is needed. In two cases the present pre-conceptual design requires design changes or is even not judged (10 out of 26) due to not available information.

The highest level of requirements considered in this report are three major GIF safety and reliability goals identified in D5.1 [4]. As little detailed design information is available about future SCW-SMR conceptual design, it is difficult to examine the level of support for the conformance to safety goals. Therefore, the applicability to future SCW-SMR conceptual design has been discussed.

Similarly, the applicability of WENRA safety objectives to future SCW-SMR conceptual design has been considered. Discussion on features of SMRs that differ from the present-day reactors [42] has been judged to relation with the applicability of objectives to future SCW-SMR conceptual design.

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