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Joint European Canadian Chinese development of Small Modular Reactor Technology

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Abstract:

This deliverable outlines all issues related to the demonstration of safety in each of the development phase of the new reactor technology with a specific focus on the Supercritical Water-Cooled Small Modular Reactor (SCW-SMR). The document provides definitions of development phases, their relations to the scope of safety demonstration, definitions of safety requirements and their safety criteria, required Codes and Standards for a designing, the experimental program needed in an evaluation of material properties to prove their suitability for application in SCW-SMR, applicability of analytical tools for safety demonstrations, and also the legislative status for possibility of SCW-SMR licensing.

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Executive Summary

The main objective of the ECC-SMART project in WP5.4 is the preparation of guidelines for safety demonstration in the individual phases of the new reactor design development. This document is the main outcome of the activities and summarizes the identified needs. It means that the individual chapters reflect the activities which need to be performed to provide safety demonstration, i.e. to identify the phase of development, to review the state of the design of the reactor technology to be analysed, to define appropriate safety requirements and safety criteria for a demonstration of a fulfilment of the safety requirements, review of the codes and standards as the design must follow them and also analytical tools needed for the analytical demonstrations. Finally, the legislation basis has a key importance as the licensing of nuclear facilities is the national responsibility in the European Union, so the preparation of legislation in four of the ECC-SMART partner countries is discussed from the SCW-SMR licensing perspective.

Chapter 2, titled “Development Stages of SCW-SMR", provides a detailed overview of the development process of supercritical water-cooled small modular reactors within the project’s framework. The chapter examines key milestones, design advancements, and technological innovations. It highlights the progress from initial conceptual studies to the refinement of reactor components and operational parameters. The content is based on an in-depth analysis of relevant literature and research findings, ensuring alignment with the broader goals of the ECC-SMART project. Chapter 2 also outlines the practical implications of these advancements, particularly enhancing reactor safety and efficiency. Additionally, it discusses the potential of SCW-SMR technology to contribute to the transition toward sustainable and low-carbon energy systems. Finally, the chapter reflects on the role of ECC-SMART in positioning this technology as a viable solution for future energy needs.

Chapter 3 describes the application of defence in depth (DiD) principle for the SCW-SMR. After a general description of DiD objectives and levels, the possible plant states for novel reactors are introduced. DiD levels for Small Modular Reactors are shown based on the IAEA document INSAG-28 [INSAG-28], called Application of the Principle of Defence in Depth in Nuclear Safety to Small Modular Reactors. Based on the literature review and the INSAG-28 document mentioned above, the suggested DiD levels and plant states for SCW-SMR have been defined. Because of the limitations of the pre-conceptual design phase, only preliminary recommendations can be formulated in this state. Plant states include Design Basis Condition states (DBC1-4) and Design Extension Conditions (DEC1-2) for complex failure without and with core melt, respectively. In Chapter 3, the association of suggested safety functions and plant states for SCW-SMR is introduced, together with a short description of the suggested safety systems of the design.

Chapter 4 provides guidelines for the safety requirements, safety criteria and methods for safety demonstration, needed for future demonstration of SCW-SMRs. In the frame of safety requirements, the preparation for pre-licensing, which is typically done simultaneously with conceptual design, is described first. Then an example of NUWARD preparation for pre-licensing and pre-conceptual and conceptual design is given, and GIF safety approach for design. Lastly, guidelines and instructions regarding the applicability of Gen IV goals and WENRA safety objectives of SMR are given, and recommendations regarding IAEA standards for design. In the frame of safety criteria their role is described in relation to safety requirements and major criteria for the three barriers are described (i.e. fuel safety criteria, primary circuit criteria, and containment criteria). In the frame of methods for safety demonstration, summaries are provided of the GIF goals in developing SMR reactor with particular attention on SCW SMR, of legislation (from IAEA

down to the local legislation), and of the concept of practical elimination. At the end guidelines for the definition of a preliminary safety report is given.

Chapter 5 focuses on further needs, mainly experimental ones, for the development of the SCW-SMR. The first part is related to the needs in the area of material research. The other describes the applicability of existing codes and standards for various areas and providers (US ASME and French AFCEN). The final part of Chapter 5 summarizes the applicability of the existing computer programs for mainly safety analysis for the SCW-SMR. Also, it describes the needs for their further development, because most of these computer codes were developed for LWR applications.

The description of the legislative status in relation to the SCW-SMR Licensing is discussed in Chapter 6. Generally, the legislation in the EU countries is prepared strongly for LWR Licensing. Some modifications in several countries are under preparation or will come into force soon, but they will cover issues of LWR SMR, not for Gen IV technologies like SCW-SMR. As an example, the situation in four countries co-creating the ECC-SMART project consortium is presented. In contradiction, the independent subchapter contains a description of the current progress in the USA, because the NRC prepared a proposal of updated 10 C.F.R. Part 53, which is focused on Licensing of advanced technologies. However, the plan is that Part 53 will come into force in 2027, now the phase of commenting is ongoing as the first necessary step of the preparation of a new legislation.

The conclusion summarizes the aim of the document and demonstrates six main steps in the safety demonstration to fill in any of the new technology developments. It covers the identification of the development process phase and the required scope of the safety demonstrations, definition or update of the safety requirements of the design under development, definition or update of safety criteria related to the appropriate safety requirements, review of the state of the art to identify needs of further experimental support or analytical tool development or validation, performing of the experimental program, and performing appropriate analytical program to demonstrate fulfillments of safety requirements.

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List of acronyms and abbreviations

3-D	Three Dimensional
ADS	Automatic Depressurization System
ADVANCE Act	Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act (in USA)
AI	Artificial Intelligence
AOO	Anticipated Operational Occurrences
AP600	600 MWe (Westinghouse), two-loop advanced passive plant
ASME	American Society of Mechanical Engineers
BOP	Balance of Plant
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CC-NUC	Code Cases: Nuclear Component
CEIDEN	Nuclear Fission Research and Development Technology Platform
CFD	Computational Fluid Dynamics
CIV	Containment Isolation Valve
CNRA	Committee on Nuclear Regulatory Activity (Committee of NEA)
CNSC	Canadian Nuclear Safety Commission
CRDM	Control Rod Drive Mechanism
CSG	Compact Plate Steam Generator
CSN	Consejo de Seguridad Nuclear (Spain)
CSNI	Committee on Safety of Nuclear Installations (Committee of NEA)
DBA	Design Basis Accident
DBC	Design Basis Conditions
DDLF	Digital Devices of Limited Functionality
DEC-A	Design Extension Conditions, level A (without core melting)
DEC-B	Design Extension Conditions, level B (with core melting)
DHT	Deterioration of Heat Transfer
DiD	Defence in Depth
DNBR	Departure from Nucleate Boiling Ration
DOPF	Design Options and Provisions File
DOS	Safety Option File
DPA	Displacement Per Atom
DSA	Deterministic Safety Analyses
EAC	Environmental Assisted Cracking
EBSD	Electron Back Scatter Diffraction
ECCS	Emergency Core Cooling System
ECC-SMART	(Joint) European Canadian Chinese Development of Small Modular Reactor Technology
EDX	Energy Dispersive X-ray Spectroscopy
ENIQ	European Network for Inspection and Qualification
ENSREG	European Nuclear Safety Regulators Group
EOP	Emergency Operating Procedure
EPR	European Pressurized Reactor
ESF	Engineered Safety Features
EU	European Commission
EUR	European Utility Requirements

FA	Fuel Assembly
FEM	Finite Element Method
FOAK	First of a Kind
Gen III	Generation III reactor technology
Gen III+	Generation III+ reactor technology
Gen IV	Generation IV reactor technology
GIF	Generation IV International Forum
GOX	Generic Oxidation Model
GSR	General Safety Requirements
H2	Hydrogen
HAEA	Hungarian Atomic Energy Authority
HA-LEU	High-Assay Low-Enriched Uranium
HPLWR	High Performance LWR
HTGR	High-Temperature Gas-cooled Reactor
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IHT	Improved Heat Transfer
ISAM	Integrated Safety Assessment Methodology
ISI	In-Service Inspection
IVR	In-Vessel Retention
JER	Joint Early Review
LB LOCA	Large Break LOCA
LERF	Large Early Release Frequency
LEU	Low-Enriched Uranium
LFR	Lead-cooled Fast Reactor
LHGR	Linear Heat Generation Rate
LLRF	Large Late Release Frequency
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MC	Metal Containment
MCPR	Minimum Critical Power Ratio
MOX	Mixed Oxides
MSLB	Main Steam Line Break
MSR	Molten Salt Reactor
NC&S	Nuclear Codes and Standards
NDE	Non-Destructive Examination
NDT	Non-Destructive Testing
NEA	Nuclear Energy Agency (agency of OECD)
NEIMA	Nuclear Energy Innovation and Modernization Act (in USA)
NOAK	N-th of a Kind
NPP	Nuclear Power Plant
NRC	U.S. Nuclear Regulatory Commission
NSC	Nuclear Safety Codes
NSSS	Nuclear Steam Supply System
OECD	Organization of Economic Co-operation and Development
Oi	Objective Number i
OLC	Operational Limits and Conditions
OP	Operating Procedure
OPT	Objective Provision Tree

PA	Postulated Accident
PCT	Peak Cladding Temperature
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRV	Pressure Relieve Valve
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSP	Pressure Suppression Pool
PWR	Pressurized Water Reactor
QA	Quality Assurance
QM	Quality Management
QSR	Qualitative Safety Features Review
R&D	Research and Development
RCC-MRx	AFCEN Code for SFR
RCS	Reactor Coolant System
RIA	Reactivity Initiated Accident
RI-ISI	Risk Informed In-Service Inspection
RIM	Reliability and Integrity Management
RP	Recommended Practice
RPV	Reactor Pressure Vessel
RPS	Reactor Protection System
RR	Research Reactor
SA	Severe Accident
SAM	Severe Accident Management
SBLOCA	Small Break Loss of Coolant Accident
S-CSG	Safety - Compact Plate Steam Generator
SC	Supercritical
SCW	Supercritical Water
SCWR	Supercritical Water-Cooled Reactor
SCW-SMR	Supercritical Water-Cooled Small Modular Reactor
SEM	Scanning Electron Microscopy
SFP	Spent Fuel Pool
SFR	Sodium-cooled Fast Reactor
SGTR	Steam Generator Tube Rupture
SMR	Small Modular Reactor
SOi	Safety Objective Number i
SRi	Safety Requirement Number i
SS	Stainless Steel
SSC	Systems, Structures and Components
SSR	Specific Safety Requirements
SSRT	Slow Strain Rate Testing
SUJB	Czech Republic State Office for Nuclear Safety (SONS)
TH	Thermal-Hydraulics
TRLs	Technological Readiness Levels
UNIPI	University in Pisa (Italy)
USA	United States of America
VA	Video Analytics
VVER	Water-cooled Water-moderated Energetic Reactor
WNA	World Nuclear Association

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Related documents

- Deliverable D2.1 ECC-SMART Project, Test Matrix based on available materials, version 2, 04/07/2022
- Deliverable D2.4 ECC-SMART Project, Report summarising the results of corrosion tests with pre-irradiated material, 30/11/2024
- Deliverable D3.3 ECC-SMART Project, Report on the preconceptual studies on the core layout and passive safety concept of the SCW-SMR, 31/8/2024
- Deliverable D3.6 ECC-SMART Project, Report on preconceptual design requirements for ECC SCW-SMR, draft (expected submission 31/01/2025
- Deliverable D4.1 ECC-SMART Project, Neutron physics code selection results, 15/11/2021
- Deliverable D4.2 ECC-SMART Project, Analytical investigation of neutron physics parameters relevant to the safety and feasibility of the SCW-SMR, 20/03/2023
- Deliverable D4.3 ECC-SMART Project, Report summarizing the results of pre-conceptual core design calculations, 02/07/2024
- Deliverable D5.1 ECC-SMART Project, Safety criteria and requirements for the SCW-SMR concept, 31/08/2021
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- ECC-SMART Project. (2022). Review Report on Safety Criteria and Requirements for the SCW-SMR Concept. Retrieved from ECC-SMART External Newsletter: https://enen.eu/wp-content/uploads/2022/09/3_ECC-SMART_External-Newsletter_August2022_final.pdf, August 2022

1 Introduction

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

The project consists of 6 work packages (four technical and two administrative). WP1 handles project coordination, while WP6 manages dissemination and communication, promoting the project and organizing educational activities. WP2 focuses on material testing, aiming to understand the corrosion behavior of candidate materials for SCW-SMR cladding through long-term exposures and electrochemical measurements. It also assesses the corrosion of pre-irradiated materials to support their qualification and compliance with existing standards. WP3 addresses thermal hydraulics and safety, creating a database of experimental and numerical data, improving and validating CFD models, and developing heat transfer correlations. It analyzes the safety and design of the SCW-SMR concept to derive ECC design requirements. WP4 is dedicated to neutronics and reactor physics, studying design and safety-related parameters to support pre-conceptual design. It selects appropriate neutron physics codes and provides reactor physics analysis of preliminary core layouts. WP5, Synthesis & Guidelines For Safety Standards, develops safety criteria and synthesizes findings from WPs 2, 3, and 4. This WP5 also conducts a pre-licensing study and develops safety demonstration guidelines for SCW-SMR. This report focuses on basic recommendations for the safety assessment of the SCW-SMR under development in its development phases in the main output of Task 5.4 of WP5.

The main feature of the SCW-SMR is that the coolant of the primary circuit is water in a supercritical state operating at 25 MPa and with a core outlet temperature in the range of 450 °C to 500 °C [ECC-D3.3]. These conditions place increased demands on materials used in the design of the core, reactor pressure vessel (RPV) and primary circuit, so those are some of the main knowledge gaps to be filled for the success of this Gen IV design. All other parts can be expected to be designed in the same conditions as standard Gen III or Gen III+ LWR reactors – regardless of large units or SMR. This SCWR concept with a higher core exit temperature has a higher energy conversion efficiency (in the range of 44 % [HPLWR]) compared to current light water reactors with values in the range up to 35 %, which is a very important feature favouring this concept.

The development of any nuclear reactor, SMR or even a large one, must be based on a whole range of requirements that are imposed on nuclear facilities in legislation. Here we already encounter the first fundamental obstacle, which is the hierarchy of legislative regulations, because when implementing any nuclear facility, the requirements of the legislation valid in the country where the implementation takes place must primarily be met. Fig. 1.1 illustrates the hierarchy of legislative validity in the Czech Republic. At the top of this hierarchy are the Constitution of the Czech Republic (CR) and International Treaties/Conventions. Below these are laws, with the Atomic Act being a primary example in the nuclear sector, though it is not the only relevant legislation. In the case of the construction of a nuclear facility, other laws are also involved in the processes. In the Czech Republic, it is primarily Act No, 100/2001 Coll. [CREIAACT] defining the requirements for Environment Impact Assessment and also Building Act No. 83/2021 Coll. [CRBACT], the amendment of which is being now under negotiation in the Parliament of the Czech Republic (now at the end of 2024), and the new provisions should also affect the structuring of state administration for large investment projects, including the construction of nuclear facilities. At the third level, there are decrees, the aim of which is to define more technical requirements and to explain how to fulfil individual provisions given by law. An even more detailed explanation is then contained in safety instructions issued by the Czech Republic State Office for Nuclear Safety for the area of an application of the Atomic Act [CRNACT].

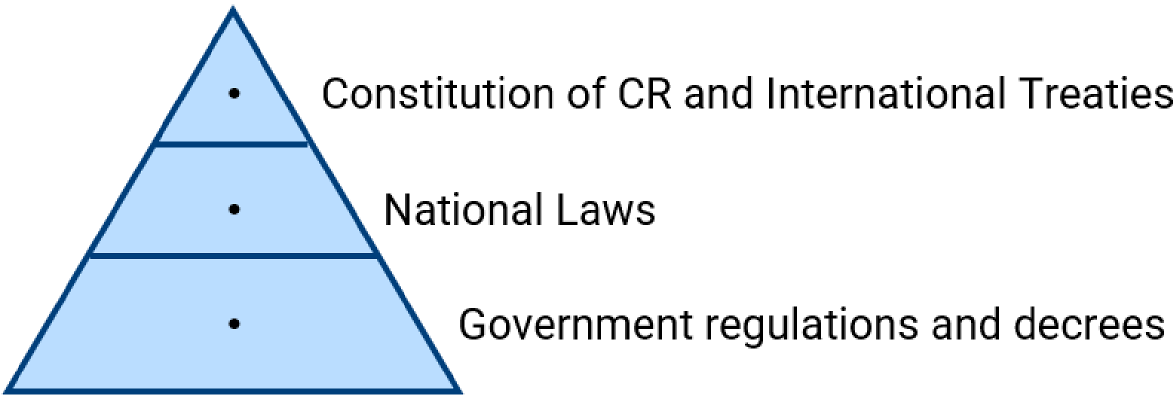

Fig. 1.1 *Legislative order in the Czech Republic*

If we look at the first level, the international conventions relevant to the use of nuclear energy for peaceful purposes are as follows (again the example of treaties/conventions signed by the Czech Republic):

- Convention on Nuclear Safety,
- Convention on Physical Protection of Nuclear Materials,
- Treaty on Non-Proliferation of Nuclear Weapons,
- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radiological Waste Management,
- Vienna Convention on Civil Liability for Nuclear Damage,
- Convention on Supplementary Compensation for Nuclear Damage,
- EURATOM Treaty.

The legislative basis above described is valid for the deployment of the nuclear facility and it shall be fulfilled in the case of the construction of the SCW-SMR in the Czech Republic.

However, the main topic to be described in this document is focused on technical requirements to be fulfilled during the process of technology development in each of the development process phases. It is obvious that the development process of advanced nuclear technology, led by the international consortium, cannot follow any specific national legislation and must be driven by requirements defined by international bodies, like IAEA, WENRA or Nuclear reactor operators (WANO). The development process must also fulfil safety targets, which must be defined in advance and can be distinguished into Deterministic and Probabilistic ones. Such deterministic targets are defined independently for all stages of the nuclear facility or its parts (operation, outage – in various phases or spent fuel pool) and any operational states of the facility (nominal operation, abnormal operational conditions or accidental conditions of DBA, DEC-A and DEC-B). The development process can be subdivided into several phases. During each phase, the developers must ensure that all safety targets are met. However, the scope of the safety assessment may vary and become broader in later phases. For the basic design, the safety assessment must cover the requirements outlined in the Safety Assessment Report.

This report is subdivided into six chapters to cover various aspects related to the development process. Chapter 1 is this introduction. Chapter 2 is focused on the descriptions of phases of the development process because there are more approaches related to distinguishing them and it is important in relation to the scope of the safety assessment. One of the approaches is related to design phases, another uses Technology Readiness Levels. Mainly that which uses TRL is more complicated because different industrial areas need their own definitions of the individual levels of the TRL. Chapter 3 describes the SCW-SMR concept of the ECC-SMART project and already applied or to be applied approaches in the concept development. Here must be pointed out mainly the Defense in Depth (DiD) approach, which has a key importance in the development process of the insuring of safety. The application of DiD is discussed at a more basic level for generic SMRs and specifically for SCW-SMR. The safety functions of the advanced reactors are also discussed in the chapter, as well as their relation to plant states. Last but not least, the concept of safety systems expected in the current SCW-SMR design used in the ECC-SMART project is briefly described showing that the current pre-concept needs significant progress to be possible to express that at least the concept of the SCW-SMR is finished. Chapter 4 is the most important as it describes requirements for safety demonstration. This chapter covers an overview of safety requirements and the basis of the international documentation (IAEA, OECD/NEA, GIF, and WENRA) and will also judge the applicability of SCW-SMR. The specific part is focused on the identification of requirements not yet applied, but to be necessarily applied in the future safety demonstrations of the SCW-SMR. The specific sub-chapter is focused on the safety criteria because any safety assessment is done against the specific safety criteria which must be fulfilled, otherwise the evaluated nuclear facility cannot be considered safe, and any safety authority would not permit such facility for construction. The independent chapter 4.3 focuses on the methods for conducting safety demonstrations, using the Czech Republic as a key example. Chapter 5 is focused on the requirements for the experimental and analytical support of the SCW-SMR development. The needs for experimental support come from two main areas – material behavior experimental program and support of new technological systems, components, and equipment. That is mainly related to the qualification of the technological parts. Data from the experimental programs are applied to the analytical tools, which are later used for safety analyses. Generally, the experimental support for the analytical tools has a key importance in the safety improvement via the reduction of the uncertainties of the analytical results. Chapter 6 is related to the licensing limitations in European countries. The responsibility of the licensing in the EU countries belongs to member states, so there is no harmonized legislation in the EU. Thus, the national legislations can significantly differ, but all of them have a common feature, which is the focus on the LWR reactors. It means that none of the European countries has its legislation prepared for the Gen IV technology licensing. Due to this common feature, the Chapter 6 includes only four examples from the legislative approaches in the Czech Republic, Slovenia, Spain and Hungary. As a comparison to the EU practice the last subchapter describes the current activities on the legislation development to allow licensing of the Gen IV designs in the USA. The last Chapter of the document summarizes the topics described and formulates individual steps of the guidelines for the safety demonstration in the various phases of the Gen IV design development, to be valid also for the SCW-SMR development activity.

2 Development Stages of SCW-SMR

2.1 Introduction of ECC-SMART SMR

The ECC-SMART project is a collaborative initiative involving European, Canadian, and Chinese partners, aimed at advancing the development of Small Modular Reactor (SMR) technology, with a specific focus on the Supercritical Water-cooled Small Modular Reactor (SCW-SMR) [ECC-SM1]. This project seeks to address the increasing demand for sustainable and safe nuclear energy solutions by leveraging the unique advantages of SCW technology, which include higher thermal efficiency and reduced operational costs [GIV08]. Additionally, the modular nature of these systems enables flexible deployment and scalability, making them well-suited for diverse energy needs. Their compact design and transportability also allow nuclear energy to reach remote areas, providing reliable and clean power to regions with limited access to traditional energy infrastructure.

The SCW-SMR design incorporates several innovative features to enhance safety, efficiency, and reliability. This reactor is designed to operate at supercritical pressures and temperatures, which significantly improve thermal efficiency compared to conventional reactors. The use of supercritical water as a coolant allows for a more compact and simplified reactor design, reducing the overall footprint and complexity of the plant.

One of the key objectives of the ECC-SMART project is to demonstrate the feasibility and safety of the SCW-SMR concept through rigorous testing and validation. This involves comprehensive safety analyses, experimental campaigns, and the development of detailed safety guidelines. The project encompasses multiple work packages, each focusing on different aspects of the reactor design and safety systems. These work packages include materials science, neutronics, thermohydraulics and safety system integration, among others.

The SCW-SMR design also emphasizes the use of advanced structural materials and innovative fuel cladding technologies to withstand the high-pressure and high-temperature environment of supercritical water. In addition to advanced materials, the project includes the study of alloys previously used in Light Water Reactors (LWRs), leveraging their proven performance and adapting them to the unique conditions of SCW systems. The reactor core is designed with optimized fuel assembly configurations to enhance neutron economy and thermal-hydraulic performance. These design elements collectively contribute to the overall safety and efficiency of the SCW-SMR, positioning it as a viable candidate for future nuclear power generation.

In addition to technical advancements, the ECC-SMART project addresses regulatory challenges by developing safety standards and guidelines that align with international best practices. By fostering collaboration among international partners, the project aims to harmonize safety requirements and facilitate the deployment of SCW-SMRs across different regions.

Ultimately, the ECC-SMART project seeks to pave the way for the deployment of SCW-SMRs, contributing to the global effort to achieve a sustainable and secure energy future. Through its innovative approach and comprehensive research, the project aims to establish SCW-SMRs as a key component of the next generation of nuclear power plants.

2.2 Safety requirements applied to ECC-SMART SMR

This chapter outlines the comprehensive safety requirements essential for the design and operation of the ECC-SMART SCW-SMR [ECC-SM2]. It covers the regulatory framework and compliance standards that the reactor must adhere to, the design basis accidents and safety margins considered during the design process, and the specific safety features and mitigation strategies implemented to ensure robust protection against potential hazards. These elements collectively ensure that the ECC-SMART SMR meets stringent safety criteria, providing a reliable and secure solution for future nuclear power generation.

2.2.1 Regulatory Framework and Compliance

The regulatory framework and compliance standards for the ECC-SMART SCW-SMR are critical to ensuring the reactor's safety and operational integrity [IAEA-24]. This subchapter provides an overview of the regulatory landscape, detailing the key regulations, guidelines, and standards that govern the design, construction, and operation of the SCW-SMR. It also highlights the compliance strategies adopted by the ECC-SMART project to meet these stringent requirements.

International Regulatory Standards

The ECC-SMART SCW-SMR must adhere to a range of international regulatory standards to ensure its safety and reliability. These standards are set by various international bodies, including:

- International Atomic Energy Agency (IAEA): Provides comprehensive safety standards and guidelines for nuclear reactors, including the General Safety Requirements (GSR) and Specific Safety Requirements (SSR) [IAEA-14].
- Generation IV International Forum: Establishes safety and performance goals for Generation IV reactors, emphasizing sustainability, safety, reliability, and economic competitiveness [GIV09].
- World Nuclear Association (WNA): Offers guidelines and best practices for the safe operation of nuclear power plants globally [WNA02].

European Regulatory Framework

Within Europe, the ECC-SMART SCW-SMR must comply with the European Union's regulatory framework, which includes:

- European Nuclear Safety Regulators Group (ENSREG): Provides oversight and coordination of nuclear safety regulations across EU member states [ENSREG1]
- Euratom Treaty: Establishes the legal framework for nuclear energy development and safety within the EU, including directives on nuclear safety, radioactive waste management, and radiation protection [EUR04]

More over, most of the authorities in the EU countries take a compliance with the EUR report requirements (in the form as certification based on the EUR Report) as very important confirmation of the correct approach to the safety of the evaluated technology.

- European Utility Requirements (EUR): The EUR is a voluntary initiative by European utilities to harmonize design requirements for new nuclear power plants, including SMRs.

It outlines technical specifications to ensure designs are safe, competitive, and licensable across Europe [EUR03]

National Regulatory Bodies

Each country participating in the ECC-SMART project has its own national supervisory authority responsible for nuclear safety oversight. For example, some national regulatory authorities include [IAEA-23]:

- Canada: Canadian Nuclear Safety Commission
- China: National Nuclear Safety Administration
- Italy: The National Inspectorate for Nuclear Safety and Radiation Protection
- Spain: The Nuclear Safety Council
- Germany: The Federal Ministry for the Environment, Nature Conservation, Nuclear Safety and Consumer Protection
- Czech Republic: State Office for Nuclear Safety
- Slovenia: Slovenian Nuclear Safety Administration

These national bodies ensure that the SCW-SMR complies with local regulations and safety standards, which may include additional requirements specific to each country.

Key Regulatory and National Requirements

The regulatory framework for the ECC-SMART project, focusing on the SCW-SMR, includes several key requirements to ensure the reactor's safety and operational integrity [IAEA-SSR-2/1]. These requirements include identifying and analyzing potential accident scenarios, known as Design Basis Accidents (DBAs), to ensure the reactor can safely withstand these events. This involves rigorous safety assessments to evaluate the reactor's response to various hypothetical accident conditions, ensuring that all potential risks are adequately mitigated. Additionally, establishing conservative safety margins is crucial to account for uncertainties in the design and operational parameters. These margins provide an additional layer of protection by ensuring that the reactor operates well within safe limits under all conditions, including unexpected events.

Emergency preparedness is another critical requirement, involving the development of comprehensive emergency response plans to protect public health and safety in the event of an accident. These plans include detailed procedures for evacuation, communication, and coordination with local authorities to ensure a swift and effective response to any emergency situation. Furthermore, ensuring that the reactor's operation minimizes environmental impact is essential. This includes the management of radioactive waste and emissions, implementing stringent controls and monitoring systems to prevent environmental contamination, and ensuring compliance with environmental regulations.

In addition to these general requirements, the ECC-SMART SCW-SMR design must comply with national regulatory requirements specific to each participating country. These requirements include obtaining the necessary licenses and permits from national regulatory bodies for the construction and operation of the reactor. This process ensures that the reactor meets all local safety and environmental standards before it can be built and operated. Detailed safety assessments are also conducted to demonstrate compliance with national safety standards.

These assessments involve thorough evaluations of the reactor's design, construction, and operational procedures to ensure they meet all regulatory requirements. Furthermore, the reactor must undergo regular inspections and audits by national regulatory bodies to ensure ongoing compliance with safety requirements. These inspections and audits help verify that the reactor continues to operate safely and in accordance with all applicable regulations.

Compliance Strategies

To achieve compliance with the diverse regulatory requirements, the ECC-SMART project employs several strategies [IAEA-GSR-P2]. One key approach is the harmonization of standards, which involves aligning the SCW-SMR design with both international and regional standards to facilitate regulatory approval across different jurisdictions. This ensures that the reactor meets the necessary safety and operational criteria globally. Additionally, the project conducts comprehensive safety analyses, including Probabilistic Safety Assessment (PSA) and Deterministic Safety Analyses (DSA), to demonstrate the reactor's safety under various operational and accident conditions. These analyses provide a robust framework for assessing potential risks and ensuring that the reactor can operate safely under a wide range of scenarios.

Stakeholder engagement is another critical strategy, involving collaboration with regulatory bodies, industry stakeholders, and the public. This engagement ensures transparency and addresses any concerns related to the SCW-SMR's safety and environmental impact. By maintaining open communication channels, the project can build trust and support among all relevant parties. Furthermore, the ECC-SMART project emphasizes continuous improvement by implementing a feedback loop that incorporates lessons learned from operational experience and ongoing research into the reactor's design and safety systems. This iterative process allows for the ongoing enhancement of safety measures and the adaptation of the reactor design to new insights and technological advancements.

Compliance with IAEA Safety Standards

The ECC-SMART SCW-SMR design adheres to the IAEA's safety standards, which include:

- IAEA Safety Fundamentals: Establishes the fundamental safety principles for nuclear installations [IAEA-SF-1].
- IAEA Safety Requirements: Provides specific requirements for the design, operation, and decommissioning of nuclear reactors [IAEA-SSR-2/1].
- IAEA Safety Guides: Offers detailed guidance on implementing the safety requirements, including best practices for safety assessment, emergency preparedness, and radiation protection [IAEA-GSG2].

By adhering to these regulatory frameworks and compliance strategies, the ECC-SMART project aims to ensure the safe and reliable operation of the SCW-SMR, contributing to the global effort to develop sustainable and secure nuclear energy solutions.

2.2.2 Design Basis Accidents and Safety Margins

Identification and Analysis of Design Basis Accidents (DBAs)

Design Basis Accidents (DBAs) are hypothetical accident scenarios that are used to evaluate the safety and robustness of the ECC-SMART SCW-SMR or any reactor in general [IAEA-SSR-2/1]. The identification and analysis of DBAs are crucial for ensuring that the reactor can withstand a wide range of potential accidents without compromising safety. The process begins with a comprehensive hazard analysis to identify all possible initiating events that could lead to an accident. These events include internal factors such as equipment failures and operator errors, as well as external factors like natural disasters and human-induced events.

Once potential DBAs are identified, they are analysed using both deterministic and probabilistic safety assessment (PSA) methods. Deterministic safety analysis involves evaluating the reactor's response to specific accident scenarios using conservative assumptions to ensure that safety margins are maintained. This type of analysis focuses on worst-case scenarios to ensure that the reactor's safety systems are capable of handling extreme conditions. Probabilistic safety assessment, on the other hand, evaluates the likelihood of different accident scenarios and their potential consequences. PSA provides a more comprehensive understanding of the risks associated with various DBAs and helps prioritize safety measures based on their probability and impact.

The analysis of DBAs also involves the use of advanced simulation tools and models to predict the reactor's behaviour under different accident conditions. These tools help identify potential vulnerabilities and areas where safety improvements are needed. The results of the DBA analysis are used to inform the design of the reactor's safety systems and to develop mitigation strategies that ensure the reactor remains within safe operational limits during and after an accident.

Establishment and Justification of Safety Margins

Safety margins are an essential component of the ECC-SMART SCW-SMR's design, providing an additional layer of protection against uncertainties in the reactor's operation and potential accident scenarios [IAEA-10]. The establishment of safety margins involves setting conservative limits on key operational parameters to ensure that the reactor operates well within safe boundaries under all conditions. These margins account for uncertainties in the design, construction, and operation of the reactor, as well as potential variations in environmental conditions and human performance.

The process of establishing safety margins begins with a thorough analysis of the reactor's design and operational parameters. This analysis identifies critical safety functions and the parameters that influence them, such as temperature, pressure, and flow rates. Conservative limits are then set for these parameters based on the results of safety analyses and engineering judgment. These limits are designed to ensure that the reactor can tolerate deviations from normal operating conditions without compromising safety.

The justification of safety margins involves demonstrating that the chosen limits are sufficient to protect against potential risks. This is done through a combination of deterministic and probabilistic safety analyses, which evaluate the reactor's response to various accident scenarios and the effectiveness of its safety systems. The results of these analyses are used to validate the safety margins and to ensure that they provide adequate protection against uncertainties.

In addition to setting conservative limits, the ECC-SMART project also emphasizes the importance of continuous monitoring and assessment of safety margins. This involves regular inspections, testing, and maintenance of the reactor's safety systems to ensure that they remain effective over time. Any deviations from the established safety margins are promptly addressed through corrective actions and design improvements.

By establishing and justifying robust safety margins, the ECC-SMART project ensures that the SCW-SMR can operate safely under a wide range of conditions, providing a reliable and secure source of nuclear energy. These safety margins are a critical component of the reactor's overall safety strategy, helping to protect against potential risks and ensuring the long-term safety and reliability of the reactor.

2.2.3 Safety Features and Mitigation Strategies

The ECC-SMART SCW-SMR incorporates a range of advanced safety features and mitigation strategies designed to ensure the reactor's safety under both normal and accident conditions [IAEA-SSR-2/1]. These features are integral to the reactor's design, providing multiple layers of defence to prevent accidents and mitigate their consequences should they occur.

Passive Safety Systems

One of the key safety features of the ECC-SMART SCW-SMR design [IAEA-16] is its reliance on passive safety systems. These systems do not require active controls or external power sources to function, making them inherently reliable even in the event of a power outage. Key passive safety systems include:

- Natural Circulation Cooling: The reactor is designed to utilize natural circulation for cooling, which relies on gravity and buoyancy forces to circulate coolant through the reactor core. This eliminates the need for mechanical pumps, reducing the risk of pump failure and ensuring continuous cooling even during power loss.
- Gravity-Driven Emergency Core Cooling System (ECCS): In the event of a Loss of Coolant Accident (LOCA), the ECCS provides rapid cooling to the reactor core by releasing coolant from elevated tanks. This system operates without the need for external power, ensuring that the core remains adequately cooled under all conditions.

Active Safety Systems

In addition to passive systems, the ECC-SMART SCW-SMR design [WNA03] is equipped with active safety systems that provide additional layers of protection. These systems are designed to detect and respond to abnormal conditions, ensuring the reactor remains within safe operational limits. Key active safety systems include:

- Reactor Protection System (RPS): The RPS continuously monitors reactor parameters and automatically initiates shutdown procedures if any parameter exceeds predefined safety limits. This system ensures rapid and reliable reactor shutdown in response to abnormal conditions.
- Containment Isolation System: This system automatically isolates the containment building in the event of a significant release of radioactive material, preventing the spread of contamination and protecting the environment and public health.

Mitigation Strategies

The ECC-SMART project has developed comprehensive mitigation strategies to address potential accident scenarios and ensure the reactor's safety [IAEA-NSG2-15]. These strategies include:

- Redundancy and Diversity: The reactor's safety systems are designed with redundancy and diversity to ensure that multiple independent systems can perform the same safety function. This reduces the likelihood of common-cause failures and enhances overall system reliability.
- Defence-in-Depth: The ECC-SMART SCW-SMR employs a defence-in-depth approach, which involves multiple layers of safety measures to protect against accidents. This includes physical barriers, safety systems, and administrative controls that work together to prevent accidents and mitigate their consequences.
- Severe Accident Management: The project has developed detailed severe accident management guidelines to address scenarios beyond the design basis. These guidelines provide procedures for managing severe accidents, including core melt scenarios, to minimize their impact and ensure the safety of the reactor and surrounding areas.

Advanced Materials and Design

The ECC-SMART SCW-SMR takes into account both well-established materials and advanced materials, alongside innovative design features, to enhance safety and performance [SPR01]. These include:

- High-Performance Cladding: The reactor core is equipped with advanced cladding materials that can withstand high temperatures and pressures, reducing the risk of cladding failure and improving overall reactor safety.
- Optimized Fuel Assembly Design: The fuel assemblies are designed to enhance neutron economy and thermal-hydraulic performance, ensuring efficient and safe reactor operation.

By integrating these advanced safety features and mitigation strategies, the ECC-SMART SCW-SMR aims to provide a robust and reliable nuclear power solution. These measures ensure that the reactor can operate safely under a wide range of conditions, contributing to the overall goal of developing sustainable and secure nuclear energy technologies.

2.3 Maturity level of ECC-SMART SMR

The maturity level of the ECC-SMART SCW-SMR can be assessed using two different methods: the design stages and the Technology Readiness Level (TRL) system. These methods provide a comprehensive understanding of the development progress and readiness of the reactor technology.

2.3.1 Design Stages

The development of the ECC-SMART SCW-SMR can be categorized into several design stages, each representing a different level of maturity and development progress [IAEA1513] and [IAEA-NRT-1-18]. These stages include:

- Pre-conceptual Design: This initial stage involves defining the basic concepts and parameters of the reactor. During this phase, preliminary safety analyses are conducted to identify potential risks and design requirements. The focus is on establishing the feasibility of the reactor concept and outlining the key design features.
- Conceptual Design: In this stage, more detailed studies are conducted to evaluate the technical and economic viability of the reactor. This includes in-depth safety analyses, material selection, and preliminary engineering designs. The feasibility design stage aims to demonstrate that the reactor concept is practical and can meet the required safety and performance standards.
- Basic Design: The basic design stage involves developing detailed design specifications and engineering drawings. This phase includes comprehensive safety assessments (Full scope of Safety Assessment Report), including both deterministic and probabilistic analyses, to ensure that the reactor design meets all regulatory requirements. The basic design provides a clear blueprint for the reactor's construction and operation. It is used in permitting of construction license.
- Detail Design: This stage produces detail drawing for production of individual parts of technology and civil construction. It is basis for updated Safety Assessment Report to be submitted as part of documentation in permitting for a nuclear facility commissioning.
- First of a Kind (FOAK): The FOAK stage represents the first commercial deployment of the reactor technology. This stage involves constructing and operating the first full-scale reactor, incorporating all the lessons learned from the previous stages. The FOAK reactor serves as a benchmark for future deployments and provides valuable operational data to further refine the design. The [IAEA1513] contains this stage, but it is not related to developing of the technology, but its deployment.

There are also alternative definitions of the project phases, as an example the approach from WNA report is included into this report [WNA01], and its structure is demonstrated at Fig. 2.1. This approach is related to the Finnish regulatory framework and distinguishes four phases:

- Phase 1: Conceptual design.
- Phase 2: Plant-level engineering design.
- Phase 3: System-level engineering design.
- Phase 4: Component-level engineering design.

These phases are described below in the independent sub-chapter 2.3.1.1, where each phase is defined in terms of both its level of engineering design and the safety and environmental assessments that the design should be capable of underpinning. The description of the phases is based on what is required for a first-of-a-kind (FOAK) reactor. For a n^{th}-of-a-kind (NOAK) reactor, the scope and level of detail required in each phase would be reduced depending on whether the NOAK reactor is being licensed or constructed in the same country as the FOAK reactor.

The ECC-SMART SCW-SMR concept is currently still in the development of the Conceptual Design stage, which is a critical phase in the development process and for the SCW-SMR not yet finalized. This stage, after its completion, involves the creation of detailed design specifications and engineering drawings, which serve as the foundation for the reactor's construction and operation. Comprehensive safety assessments are conducted during this phase, including both deterministic and probabilistic analyses, to ensure that the reactor design meets all regulatory requirements and can safely operate under various conditions.

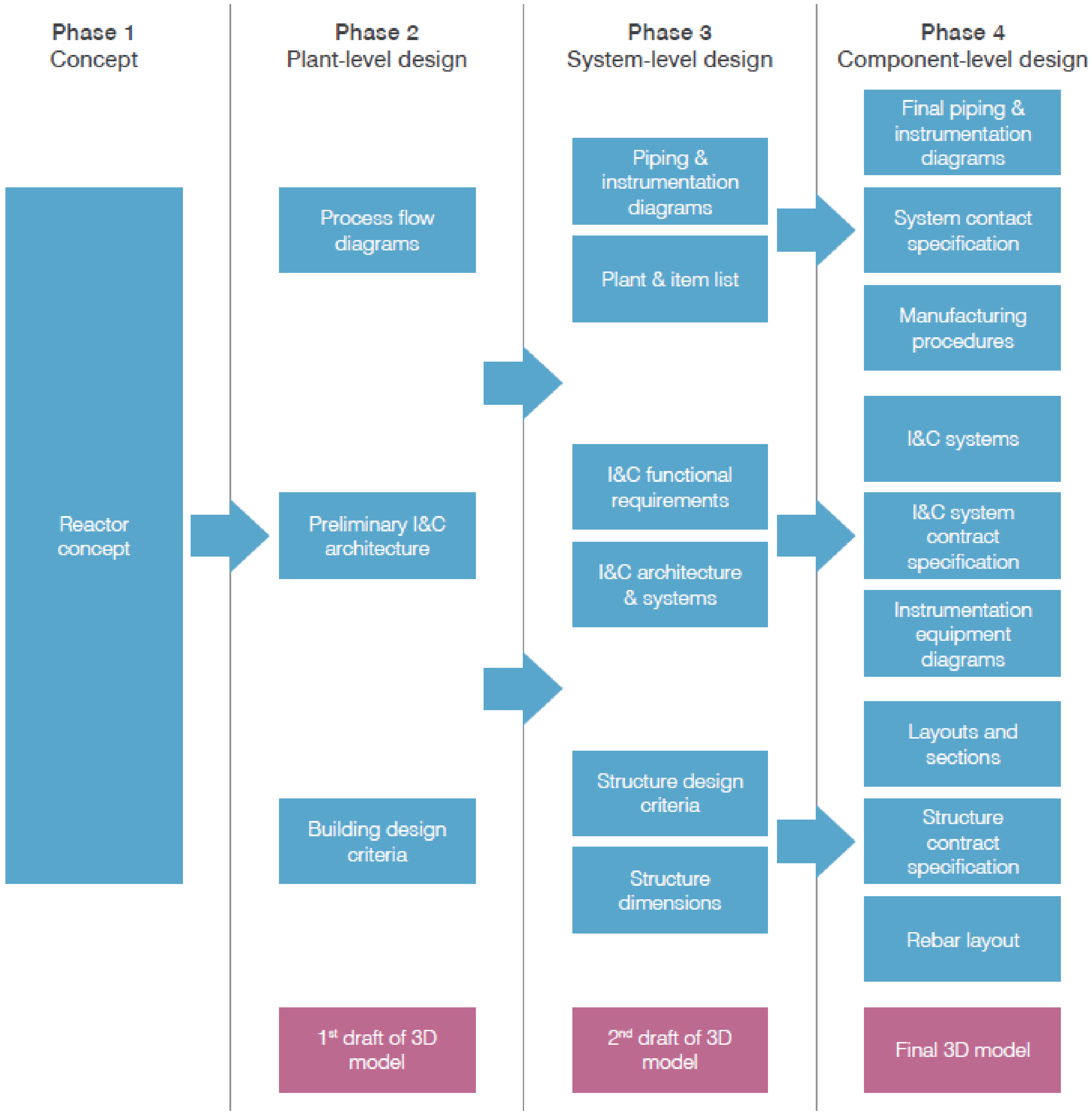

Fig. 2.1 *Schematic overview of development phases based on WNA* [WNA01]

The conceptual design stage is essential for establishing a clear and precise blueprint for the reactor. It involves rigorous evaluation of the reactor's systems and components to identify and mitigate potential risks. This phase also includes the selection of materials, optimization of fuel assembly configurations, and integration of advanced safety features. By addressing these aspects, the conceptual design ensures that the reactor is both technically feasible and economically viable.

Furthermore, this stage lays the groundwork for subsequent phases, such as the demonstration of a prototype or pilot plant. The detailed design and safety assessments conducted during the conceptual design phase provide the necessary data and insights to build and test a prototype, validating the reactor's performance and safety features in a real-world environment. This validation is crucial for identifying any design issues and making necessary adjustments before moving on to full-scale deployment.

Ultimately, the conceptual design stage is pivotal for ensuring the feasibility and safety of the ECC-SMART SCW-SMR. It provides a robust framework for the reactor's development, paving the way for its successful demonstration and eventual commercial deployment. By thoroughly addressing all design and safety aspects at this stage, the ECC-SMART project aims to develop a reliable and sustainable nuclear reactor that meets the highest standards of safety and performance.

2.3.1.1 Description of phases of WNA approach

The description of the content of individual Phases is directly cited from [WNA01].

Phase 1: Conceptual design

This is the design phase in which the design options are selected, and enhanced, critical questions are asked, solutions developed, major risks are identified, and mitigation plans put in place.

The output from this phase is generally a document, or suite of documents, outlining the design and safety principles, the key decisions taken and the rationale for those decisions.

In general, the majority of the steps in this phase should be completed prior to any engagement with the regulatory authorities, although some pre-licensing activities allow for regulatory engagement during this phase.

Phase 2: Plant-level engineering design

During this phase all key systems, structures and components (SSCs), their requirements and key design parameters should be defined. This will generally include:

- *Process flow diagrams of the systems.*
- *Preliminary instrumentation and control (I&C) architecture.*
- *Preliminary design drawings, e.g., single line diagrams.*
- *Definition of plant layout (building design criteria including basic dimensions).*
- *Preliminary specifications for safety-classified systems.*
- *Safety design.*
- *First draft of 3D model.*

The systems generally focused on in this phase include reactor core, reactor coolant system, safety systems (including auxiliary safety systems), I&C (preliminary architecture), electrical power supply, steam and power conversion systems, and civil works and structures.

The design at this stage should be sufficient to allow preliminary assessments of:

- *Plant safety against regulatory requirements.*
- *Environmental impact.*
- *Security requirements.*

The output from this phase will be a suite of documents defining the key design parameters of the safety related SSCs and safety features of the reactor design, alongside several preliminary assessments.

Phase 3: System-level engineering design

In this phase the definitions of the SSCs and their requirements and parameters are further refined, and all other plant systems are defined. During this phase the design team will grow significantly in size and capability, and the wider supply chain may be used to supplement some design capability or undertake design of SSCs under contract.

During this phase the following is normally produced:

- *Piping and instrumentation diagrams of the systems.*
- *Plant and item list.*
- *I&C functional requirements; system architecture and drawings.*
- *Structure design criteria and dimensions.*
- *Preliminary specifications for safety-related components.*
- *Second draft of 3D model.*

In addition to a more detailed description of the systems identified in Phase 2, additional systems of particular importance during this design phase are: reactor chemistry, radiological protection systems, and radioactive waste management systems.

Assessments undertaken at this phase require a greater underpinning of the design of the SSCs and their associated support systems. The design at this phase should be sufficient to allow the preliminary safety analysis report (PSAR) to be produced and the following assessments to be undertaken:

- *Design basis and design extension conditions including deterministic analysis*
- *Probabilistic safety assessment (Level 1 and 2).*
- *Assessment and justification of any new materials proposed.*
- *Human factor engineering.*
- *Internal hazards: preliminary assessment.*
- *External hazards: definitions of required loads for building design.*
- *Operational principles and requirements.*
- *Decommissioning requirements.*
- *Environmental impact assessment[1].*
- *Security requirements.*

It should be noted that this stage does not require the detailed component engineering design that is needed for the components to be manufactured, i.e. not all isometric drawings or detailed 3D models of components need to be developed at this point. The timing of the detailed design for manufacturing will be driven by the deployment schedule of the individual project.

Phase 4: Component-level engineering design

Very often for large-scale nuclear plants, the design process for the lower safety critical systems and non-safety systems, within the nuclear island, will take place during the construction phase.

[1] Environmental impact assessment may be produced and assessed separately to the safety analysis documentation.

In the context of SMR deployment, this may be less feasible as a result of the modular nature of construction and more of these systems and components will need to be designed and manufactured earlier in the process.

It is during this phase that the final detailed engineering to allow manufacture of all SSCs for the entire plant is undertaken with the aim that design modifications are minimized once construction commences.

During this phase the following is normally produced or updated:

- *Finalized detail design including manufacturing requirements and component specifications for all SSCs.*
- *Building layout specifications and drawings.*
- *Final 3D model.*

The design at this phase should be sufficient to allow the final safety assessment report (FSAR) to be produced and assessments of the following to be undertaken:

- *Design basis and design extension conditions, including deterministic analysis.*
- *Probabilistic safety assessment (Level 1 and 2)*[2]
- *Assessment and justification of any new materials proposed.*
- *Human factors engineering.*
- *Internal and external hazards.*
- *Operational principles and requirements.*
- *Decommissioning requirements.*
- *Environmental impact assessment.*
- *Security requirements.*

An overview of the different phases of design maturity is provided in Fig. 2.1.

2.3.2 Technology Readiness Level System in Space industry

The Technology Readiness Level (TRL) system is another method used to assess the maturity of the ECC-SMART SCW-SMR. The TRL system consists of nine levels, ranging from basic research (TRL 1) to full-scale deployment (TRL 9). Each level represents a different stage of development and readiness [NASA01]:

- TRL 1: Basic Principles Observed: Initial scientific research begins, and basic principles are observed. This level involves fundamental studies to understand the underlying science of the reactor technology.
- TRL 2: Technology Concept Formulated: The technology concept and application are formulated. This includes defining the reactor’s basic design and identifying potential applications.
- TRL 3: Experimental Proof of Concept: Experimental research is conducted to demonstrate the feasibility of the technology concept. This includes laboratory-scale experiments and initial safety analyses.

[2] Some countries also required Level 3 probabilistic safety assessment.

- TRL 4: Technology Validated in Lab: The technology is validated in a laboratory environment. This involves more detailed experiments and simulations to verify the reactor's performance and safety.
- TRL 5: Technology Validated in Relevant Environment: The technology is tested in a relevant environment, such as a pilot plant or prototype. This stage includes extensive testing to validate the reactor's design and safety features under realistic conditions.
- TRL 6: Technology Demonstrated in Relevant Environment: The technology is demonstrated in a relevant environment, with a focus on scaling up the reactor design and addressing any technical challenges.
- TRL 7: System Prototype Demonstration in Operational Environment: A system prototype is demonstrated in an operational environment. This stage involves building and testing a full-scale prototype to validate the reactor's performance and safety.
- TRL 8: System Complete and Qualified: The reactor system is complete and qualified through testing and demonstration. This level involves finalizing the reactor design and ensuring that it meets all regulatory and safety requirements.
- TRL 9: Actual System Proven in Operational Environment: The reactor technology is proven in an operational environment, with the first commercial deployment. This stage involves constructing and operating the first full-scale reactor, providing valuable operational data for future deployments.

The ECC-SMART SCW-SMR concept is currently at TRL 2: Technology Concept Formulated. At this stage, the focus is on developing the fundamental principles and design features of the reactor, supported by initial theoretical studies, simulations, and feasibility assessments. This includes identifying the key components, safety systems, and operational parameters required for the reactor to function effectively.

While experimental validation of the reactor's core components and safety systems has not yet been conducted, this early stage of development establishes the groundwork for subsequent testing and validation. Reaching TRL 2 signifies that the concept is well-defined and provides a clear direction for future research and development.

The next steps will involve advancing the design to TRL 3 by initiating experimental work to validate the key principles in a laboratory setting. This progression will include testing critical components, assessing safety systems, and refining the design based on experimental data. These efforts will ensure the ECC-SMART SCW-SMR's readiness for further technological development and eventual deployment as a reliable and sustainable nuclear energy solution.

By assessing the ECC-SMART SCW-SMR using both the design stages and the TRL system, we can gain a comprehensive understanding of the reactor's maturity level and readiness for deployment. These methods provide a clear roadmap for the development and commercialization of the reactor technology, ensuring that it meets all safety and performance standards.

Table 2.1 *Definition of TRLs for development of SMR* [CR-New1]

TRL 9	The project is implemented and the reactor system is operational	TRL 9: The actual/ real SMR system is operated/operates over the full range of expected conditions.	The project has been implemented in its final form and is operating at full scale under expected operating conditions.
TRL 7-8	The reactor system is commissioned/ being commissioned	TRL 8: The actual system is completed and qualified through tests and demonstrations	This SMR project technology has been shown to be operationally proven in its final form and under the expected conditions. In almost all cases, this TRL represents the end of the actual development of the project. Examples include developmental testing and evaluation of SMR project systems. Supporting information includes operating procedures that are virtually/nearly complete. The Operational Readiness Review (ORR) was successfully completed prior to the start of hot functional testing.
		TRL 7: Full-scale or prototype key equipment/systems/ technologies of the project have been successfully tested in a relevant/ appropriate environment.	This represents a major advance from TRL 6. TRL 7 requires the demonstration of an actual prototype system in a relevant environment. Examples include full-scale prototype testing during cold commissioning/ start-up. Supporting information includes full-scale test results and analysis of differences between test environments and an analysis of what the experimental results mean for the relevant operating system/ environment. The final design is virtually complete.
TRL 4-6	Demonstrator of the technology used in the reactor project	TRL 6: Engineering/ pilot scale, validation of a similar (prototype) key device/system/ technology of the project in a relevant environment	Engineering scale models or prototypes are tested in the appropriate environment. This represents a major step in the demonstrated technology readiness of the SMR project. Examples include testing a prototype system on an engineering scale. Supporting information includes engineering scale test results and analysis of differences between the engineering scale, prototype system/medium, and analysis of what the experimental results mean for the relevant operating system/environment. TRL 6 initiates a real technical evolution of the technology as an operating system. The main difference between TRL 5 and 6 is the transition from laboratory scale to engineering scale and the establishment of scaling factors to enable the design of the operating system. The prototype should be able to perform all the functions that the operating system will require. The operational environment for testing should closely match the actual operational environment.
		TRL 5: Laboratory scale, validation of a similar system in a relevant environment	The basic technological components of the SMR project are integrated in such a way that the system configuration corresponds in almost all respects to the final version of the project (basic design). Examples include testing a high-fidelity system at laboratory scale in a simulated environment with a range of simulants and real waste. Supporting information includes the results of lab-scale testing, an analysis of the differences between the lab and the relevant operating system/environment, and an analysis of what the experimental results mean for the relevant operating system/environment. The main difference between TRL 4 and 5 is the increased fidelity of the system and environment to the actual application. The tested system is almost a prototype.
		TRL 4: Validation of components and/or the system in a laboratory environment	The core technology components are integrated to ensure that the individual components work together properly in the system. Examples include ad hoc hardware integration in the lab. Supporting information includes results of integrated experiments and estimates of how experimental components and experimental test results differ from expected system performance goals. TRL 4-6 represent a bridge from scientific research to engineering. TRL 4 is the first step in determining whether the individual components will work together as a system. The laboratory system is likely to be a combination of hand-held equipment and a few special components that may require special handling, calibration or securing to work.
TRL 2-3	Research and development for verification pre-conceptual and conceptual design of the reactor	TRL 3: SMR conceptual design developed.	Active research and development of the SMR project has been initiated following the pre-conceptual project. This includes analytical and laboratory scale studies to physically validate the analytical predictions of individual elements of the technology. Examples include components that are not yet integrated or are being tested in representative experimental facilities. Supporting information includes the results of laboratory tests performed to measure the parameters of interest and comparisons with analytical predictions for critical subsystems. At TRL 3, the work has moved from the paper phase to experimental work, verifying that the project concept works as expected on experimental devices. However, the technology components used in the SMR project are not yet integrated into the overall system. Modelling and simulation can be used to complement physical experiments.
		TRL 2: Pre-conceptual project created	Once the basic principles have been verified, a pre-conceptual design of the SMR can be developed. Applications of the technologies are speculative and there may not be experimental validation or a detailed analysis to support these assumptions at this stage. Examples are still limited to analytical studies. Supporting information includes publications or other references that outline the application under consideration and provide analysis supporting the concept. The step from TRL 1 to TRL 2 moves ideas from pure to applied research. Most of the work involves analytical or paper-based studies with an emphasis on better understanding the science. The experimental work is designed to confirm the basic scientific observations made during TRL 1.
TRL 1	Basic research on the technology considered for the reactor project	TRL 1: Basic principles identified and published	This is the lowest level of technology readiness used in the SMR project. Basic research is beginning to translate into applied research and development. Examples include theoretical studies of the basic properties of a technology or experimental work that consists mainly of observations of physical phenomena. Supporting information includes published research or other references that identify the principles underlying the technologies used in the pre-conceptual design of the project.

2.3.3 TRL in Nuclear industry

Definition of TRLs always depends on the industrial sector of their applications, because for instance in the Chemical industry the testing of new technology is always at laboratory scale, half industrial scale and then applied to full industrial scale. That is the reason, that for example in the

Czech Republic the working group organized by the Ministry of Industry and Trade on Deployment of SMRs in CR decided to apply own definitions of TRLs for SMR (See Table 2.1 [CR-New1]).

2.4 Summary of the primary issues

To advance the ECC-SMART SCW-SMR from its current stage of development to higher levels of maturity, several key features need to be implemented and refined. These features are essential for ensuring that the reactor design meets all safety, performance, and regulatory requirements, enabling it to progress through the design stages and Technology Readiness Levels (TRLs). This process involves a comprehensive approach that includes enhancing safety systems, optimizing material and structural components, and conducting rigorous operational and safety analyses. Additionally, addressing regulatory and compliance challenges is crucial to ensure that the reactor meets both international and national standards. All technical parts of this chapter are based on document [PROS01] which provides valuable insights into these requirements and the necessary steps for compliance. By focusing on these areas, the ECC-SMART project aims to develop a robust and reliable reactor design that can be successfully demonstrated and deployed in real-world conditions. This holistic approach not only improves the reactor's technical capabilities but also builds confidence among stakeholders, regulatory bodies, and the public, paving the way for the widespread adoption of SCW-SMR technology.

2.4.1 SMR-specific issues

The development and deployment of Small Modular Reactors present unique challenges and opportunities compared to traditional large-scale nuclear reactors. The ECC-SMART with its supercritical water-cooled design, brings additional specific issues that need to be addressed to ensure its successful implementation.

One of the key issues in the development of the SCW-SMR is the adaptation of safety features from large reactors, while also incorporating more extensive use of passive safety systems. Large reactors typically rely on a combination of active and passive safety systems to ensure safe operation. However, for the SCW-SMR, there is a greater emphasis on passive safety features due to their inherent reliability and simplicity. Passive safety systems, such as natural circulation cooling and gravity-driven emergency core cooling, do not require external power or active controls, making them highly effective in maintaining reactor safety during power outages or other emergencies. The SCW-SMR design leverages these passive systems to enhance safety, reduce complexity, and improve overall reliability. By minimizing reliance on active components, the SCW-SMR can achieve a higher level of safety and resilience, addressing one of the critical challenges in the deployment of small modular reactors.

Another significant issue for the SCW-SMR is the small core inventory, which results in lower decay heat and smaller source terms. This characteristic presents unique opportunities for enhancing safety and reducing the environmental impact. The reduced decay heat generated by the smaller core allows for the implementation of innovative safety systems, such as air cooling for residual heat removal. Air cooling systems can effectively dissipate residual heat without relying on complex active cooling mechanisms, further enhancing the reactor's passive safety profile. Additionally, the smaller source terms associated with the SCW-SMR mean that, in the event of an accident, the potential release of radioactive materials to the environment is significantly lower. This reduction in potential release can lead to a decrease or even elimination of emergency preparedness zones, simplifying emergency planning and reducing the overall impact on surrounding communities. By leveraging these advantages, the SCW-SMR can achieve a higher level of safety and environmental protection, addressing key concerns associated with nuclear power generation.

Next SMR-related issue specific to the SCW-SMR involves differences in reactor physics parameters, such as higher fuel enrichment and the use of new fuel types. The SCW-SMR design often requires higher enrichment levels to achieve the desired neutron economy and thermal efficiency. This higher enrichment can pose challenges in terms of fuel handling, storage, and regulatory compliance. Additionally, the use of new fuel types, such as advanced cladding materials and innovative fuel compositions, introduces complexities in fuel fabrication and performance assessment. These new fuel types must be rigorously tested to ensure they can withstand the high-pressure and high-temperature conditions of supercritical water reactors. The differences in reactor physics parameters also necessitate detailed safety analyses to evaluate the impact on reactor behavior, including reactivity control, fuel burnup, and potential accident scenarios. Addressing these challenges is crucial for ensuring the safe and efficient operation of the SCW-SMR and requires ongoing research and development to optimize fuel performance and reactor safety.

2.4.2 WP2: Materials Testing

One of the critical issues in the materials used for the SCW-SMR is their corrosion behavior during long-term exposure to the high-pressure and high-temperature environment of supercritical water. Over extended periods, materials such as fuel cladding and structural components can undergo significant corrosion, which can compromise their mechanical integrity. This is particularly concerning for materials exposed to supercritical water, as the aggressive environment can accelerate corrosion rates.

One of the main challenges of SCWRs is that, due to their operating conditions, Zr alloys—widely used for fuel cladding in LWRs—are not suitable. Zr alloys suffer accelerated corrosion under supercritical water (SCW) conditions, which significantly limits their performance. Therefore, the selection of traditional materials that have demonstrated good corrosion resistance both in LWRs and in simulated SCWR operating conditions, as well as advanced materials such as AFA alloys, are considered potential candidates. To address this issue, the ECC-SMART project, in particular WP2, focuses on extended testing of selected materials (310S, 800H, AFA) to prove their corrosion resistance and integrity under these extreme conditions with potential application as fuel cladding.

In general, such materials need to be rigorously tested through long-term exposure experiments to evaluate their corrosion behavior and ensure their durability over the reactor’s operational lifespan.

Additionally, understanding the mechanisms of corrosion and the factors that influence it, such as temperature, pressure, and water chemistry, is crucial. This knowledge is crucial for the selection of structural material and ensuring the safe operation of developed nuclear technology. In addition, based on the knowledge of the materials' behaviour, some mitigation strategies can be proposed including optimized fuel campaign, water chemistry or implementing protective coatings. However, proper knowledge of materials behavor under simulated operational conditions is significant for establishing/optimizing the reguirements on structural materials, which should be used in newly developed nuclear technologies such as SCW-SMR.

While the ECC-SMART project has investigated material behavior using tubular samples to understand the potential effects of geometry and the manufacturing process, it should be noted that this does not replicate the exact manufacturing processes used for fuel cladding. This raises the question of whether the intent is to strictly reflect project-specific achievements or to also explore potential manufacturing alternatives.

The effect of radiolysis in supercritical water and the resulting changes in electrochemistry with varying pressure and temperature are critical issues for the SCW-SMR. Radiolysis, the dissociation of water molecules due to radiation, produces reactive species such as hydrogen peroxide, oxygen, and hydrogen. These species can significantly alter the chemical environment inside the reactor, impacting material corrosion and overall reactor safety.

In the SCW-SMR, the high-pressure and high-temperature conditions could exacerbate these effects, leading to more aggressive corrosion behavior. The reactive species generated by radiolysis can accelerate the oxidation and degradation of structural materials and cladding, compromising their integrity over time. Understanding the kinetics of radiolysis and the stability of the produced species under supercritical conditions is essential for predicting and mitigating these effects.

Additionally, the chemical and electrochemical properties of water change depending on pressure and temperature variations which also influence the corrosion potential of materials. At supercritical conditions, the solubility of gases and the ionic strength of the water change, affecting the electrochemical reactions at the material surfaces. These changes may lead to alterations in corrosion mechanisms, which in turn could impact the material's behavior in supercritical water.

In conclusion, addressing the material issues for the SCW-SMR is critical for ensuring the reactor's long-term safety and reliability. Key challenges include managing corrosion behavior during long-term exposure, mitigating the effects of irradiation on fuel cladding materials, optimizing manufacturing processes to prevent crack initiation, and understanding the impact of radiolysis and electrochemical changes under supercritical conditions. By selecting advanced materials with superior performance characteristics, refining manufacturing techniques, and conducting comprehensive testing and analysis, the ECC-SMART project can enhance the durability and integrity of some SCW technology components. These efforts are essential for developing a robust and resilient SCW-SMR that can operate safely and efficiently under the demanding conditions of supercritical water.

The results obtained in Work Package 2 of the ECC-SMART project have demonstrated that the 800H and 310S alloys exhibit excellent corrosion behavior under simulated reactor conditions at different pressures and temperatures and for long exposures. Additionally, the AFA alloy, based on a 310S composition and specifically developed for the project, also showed promising corrosion resistance. However, further testing is needed to enhance its microstructure and confirm its performance under extended conditions. These materials also demonstrated good performance under simulated accident conditions. Notably, the 800H alloy, after testing in steam at 1200 °C, developed structurally stable oxide layers, a critical property for accident tolerance. In contrast, the 310S alloy and the AFA did not exhibit the same level of structural stability in their oxide layers under these extreme conditions.

The effects of neutron irradiation up to 0.3 dpa were also studied, revealing that this level of radiation primarily induces the formation of black dots in the microstructure. While this microstructural change leads to notable variations in mechanical properties, it does not affect corrosion resistance in supercritical water. This work represents a valuable first step, but further studies at higher doses and longer exposure times will be necessary to fully understand the behavior under operational conditions of SCW-SMR. Significant progress has also been made in understanding the electrochemical processes involved in the supercritical environment. Temperature and pressure were identified as critical variables, with oxygen having a less pronounced but measurable effect. The results confirm previous observations that a corrosion has a maximum rate at approximately 380 °C, compared to slightly lower or higher temperatures, although the predominant effect of temperature is evident between 380 °C and 500 °C. Finally, radiolysis studies have yielded highly promising results, suggesting that it may be possible to suppress radiolysis up to 500 °C with hydrogen injection. These findings mark an important milestone in advancing the understanding and development of materials and processes for supercritical water reactors.

However, the ECC-SMART project contains several experimental material testing programs, there are other topics which require additional testing. Such program could be included in any follow up project on SCW technology. Some of them are pointed out in following paragraphs.

One of another critical issue for the SCW-SMR is the effect of irradiation on fuel cladding materials. Fuel cladding materials in nuclear reactors are exposed to intense neutron radiation, which can significantly alter their physical and mechanical properties over time. This irradiation can lead to embrittlement, swelling, and changes in thermal conductivity, all of which can compromise the integrity and performance of the fuel cladding.

Research into the effects of irradiation involves both experimental studies and computational modelling. Experimental studies typically include irradiation tests in research reactors, where samples of fuel cladding (without fuel) are exposed to neutrons similar to those in an operating reactor. These irradiated specimens are further tested in the Hot Cells. These tests help determine the changes in mechanical properties, microstructure, and corrosion resistance of the materials. The manufacturing process of tubes and rods plays a crucial role in the crack initiation behavior of candidate materials for the SCW-SMR. Techniques such as welding, extrusion, and heat treatment can introduce residual stresses and microstructural changes that significantly impact material integrity. These manufacturing-induced stresses can act as sites for crack initiation, especially under the high-pressure and high-temperature conditions of supercritical water reactors.

For the SCW-SMR, optimizing manufacturing processes is crucial to minimizing adverse effects such as residual stresses and material instability. This includes selecting suitable manufacturing techniques to enhance microstructural stability and ensure long-term performance. Advanced methods like precision welding and controlled heat treatment can play a key role in achieving these objectives. Additionally, rigorous testing and quality control are vital to ensure that components meet stringent safety and operational requirements.

2.4.3 WP3: Thermal Hydraulics and Safety of the SCW-SMR

Thermohydraulics plays a crucial role in the design and operation of the SCW-SMR, as it directly impacts the reactor’s efficiency, safety, and overall performance. This subchapter will explore the key thermohydraulic issues identified in Work Package 3 (WP3) of the ECC-SMART project, focusing on the behavior of supercritical water under high-pressure and high-temperature conditions. Understanding the complex interactions between heat transfer, fluid dynamics, and reactor components is essential for optimizing the reactor’s thermal-hydraulic performance and ensuring its safe operation. This section will address the challenges and solutions related to coolant flow dynamics, heat transfer mechanisms, and the impact of thermohydraulic phenomena on reactor safety and efficiency.

One of the primary challenges in the thermohydraulic analysis of the SCW-SMR is accurately modelling turbulent heat transfer under supercritical conditions. Supercritical water exhibits unique thermal and fluid dynamic properties that differ significantly from those of subcritical fluids. These properties include drastic changes in density, specific heat, and thermal conductivity near the pseudo-critical point, which complicate the prediction of heat transfer behavior.

To address these challenges, advanced models and methods are required to accurately simulate turbulent heat transfer in supercritical water. These models must account for the complex interactions between fluid flow and heat transfer, including the effects of buoyancy, variable properties, and turbulence. Computational Fluid Dynamics (CFD) simulations play a crucial role in this regard, providing detailed insights into the flow and thermal fields within the reactor.

Additionally, empirical correlations are essential for predicting heat transfer coefficients under supercritical conditions. These correlations are typically derived from experimental data and must be validated against a wide range of operating conditions to ensure their accuracy and reliability. Developing robust correlations for supercritical heat transfer is critical for designing efficient and safe cooling systems for the SCW-SMR.

Another significant issue is the modelling of Deterioration of Heat Transfer (DHT), a phenomenon that can occur under certain conditions in supercritical water reactors. DHT is characterized by a sudden decrease in heat transfer efficiency, leading to localized overheating and potential damage to reactor components. Accurate prediction and mitigation of DHT are essential for maintaining the reactor’s thermal-hydraulic performance and safety.

To model DHT, it is necessary to understand the underlying mechanisms that cause this phenomenon, such as changes in flow regime, turbulence intensity, and thermal stratification. Advanced CFD models and experimental studies are required to investigate these mechanisms and develop strategies to prevent or mitigate DHT. This includes optimizing the reactor’s operating conditions, such as flow rates and heat flux distributions, to minimize the risk of DHT.

A significant knowledge gap exists in the formulation of the design and safety concept for the SCW-SMR, particularly concerning the thermohydraulic aspects. The unique properties of supercritical water, such as its variable density and thermal conductivity near the pseudo-critical point, introduce complexities that are not fully understood. These complexities affect the design of the reactor’s cooling system, the prediction of heat transfer behavior, and the management of thermal-hydraulic stability. To bridge this knowledge gap, extensive research and development are required to develop accurate models and simulations that can predict the behavior of supercritical water under various operating conditions. This includes understanding the interactions between fluid dynamics, heat transfer, and reactor materials, as well as the impact of these interactions on reactor safety and performance.

Furthermore, the safety concept for the SCW-SMR must address the potential risks associated with supercritical water conditions, such as the DHT and the challenges in maintaining effective cooling during transient and accident scenarios. Developing a robust safety concept involves identifying and mitigating these risks through advanced safety systems, such as passive cooling mechanisms and ECCS. It also requires comprehensive safety assessments, including both deterministic and probabilistic analyses, to evaluate the reactor's response to various accident scenarios.

Understanding and addressing the most significant thermohydraulic phenomena within the SCW-SMR project is crucial for optimizing reactor performance and safety. Heat and mass transfer along corroded and rough surfaces significantly impact the thermal and hydraulic characteristics of reactor components, affecting heat transfer efficiency and fluid flow dynamics. Corroded surfaces increase resistance to heat transfer, while rough surfaces enhance turbulence, potentially improving heat transfer but also increasing pressure drop. Experimental studies and advanced modelling are essential to quantify these effects and ensure the reliability of the cooling system.

Heat transfer in water under supercritical pressure conditions is another complex phenomenon central to the SCW-SMR's operation. Supercritical water exhibits unique thermal properties, such as drastic changes in density and specific heat near the pseudo-critical point, which significantly affect heat transfer performance. Accurate prediction of heat transfer coefficients in this regime is crucial for designing efficient cooling systems, requiring advanced computational fluid dynamics (CFD) models and empirical correlations.

Deterioration of Heat Transfer (DHT) is characterized by a sudden decrease in heat transfer efficiency, leading to localized overheating and potential damage to reactor components. Understanding the mechanisms that cause DHT is essential for developing strategies to prevent or mitigate its occurrence, involving detailed experimental investigations and predictive models.

Turbulent heat and mass transfer in water under supercritical pressure conditions enhances mixing and heat transfer rates but complicates flow behavior prediction. Advanced CFD simulations are essential for capturing detailed flow and thermal fields within the reactor.

The transition from supercritical to subcritical pressure involves significant changes in fluid properties, impacting heat transfer and fluid flow dynamics. Managing this transition is crucial for maintaining reactor stability and preventing thermal-hydraulic instabilities, requiring detailed modelling and experimental studies.

In conclusion, the thermohydraulic analysis of the SCW-SMR is pivotal for ensuring its safe and efficient operation. Addressing key phenomena such as heat and mass transfer along corroded and rough surfaces, heat transfer under supercritical pressure conditions, and the DHT is essential. Additionally, understanding turbulent heat and mass transfer and managing the transition from supercritical to subcritical pressure are critical for maintaining reactor stability and performance. By leveraging advanced modelling, simulation tools, and experimental studies, the ECC-SMART project aims to analyse the reactor's thermal-hydraulic design, enhance its cooling systems, and ensure robust safety measures. These efforts are crucial for the successful deployment and operation of the SCW-SMR, contributing to the advancement of sustainable and reliable nuclear energy technologies. In conclusion, the thermohydraulic analysis of the SCW-SMR has been a central focus of the ECC-SMART project, aimed at ensuring its safe and efficient operation. Significant progress has been made in addressing key phenomena, including heat and mass transfer along corroded and rough surfaces, heat transfer under supercritical pressure conditions, and the Departure from Heat Transfer (DHT). Advances have also been achieved in understanding turbulent heat and mass transfer and managing the transition from supercritical to subcritical pressure, both of which are critical for maintaining reactor stability and performance.

Through the use of advanced modeling and simulation tools, as well as targeted experimental studies, the project has successfully optimized several aspects of the reactor's thermal-hydraulic design. This includes improved predictions of critical heat transfer behaviors and enhanced cooling system performance. While some experimental validation remains limited due to project constraints, the insights gained provide a solid foundation for future research and development.

Overall, ECC-SMART has made significant strides in advancing the understanding of thermal-hydraulic phenomena for SCW-SMRs and has contributed valuable knowledge to support the safe and sustainable deployment of this technology. These achievements underscore the project's role in pushing the boundaries of modern nuclear energy solutions.

2.4.4 WP4: Neutron Physics of the SCW-SMR

Neutronics plays a fundamental role in the design and operation of the SCW-SMR, influencing critical aspects such as reactor stability, fuel efficiency, and safety. This subchapter will delve into the key neutronic issues identified in WP4 of the ECC-SMART project. It will explore the challenges associated with neutron flux distribution, reactivity control, fuel burnup, and the impact of advanced fuel types on reactor performance. Understanding these neutronic phenomena is essential for optimizing the reactor core design, ensuring effective reactivity management, and enhancing overall reactor safety.

One of the critical neutronic issues for the SCW-SMR is the behavior of temperature reactivity coefficients. In the SCW-SMR, all temperature reactivity coefficients are designed to be negative, which is a desirable safety feature. Negative temperature reactivity coefficients ensure that as the temperature of the reactor increases, the reactivity decreases, providing an inherent feedback mechanism that helps stabilize the reactor and prevent runaway reactions. However, the behavior of these coefficients can vary significantly at different stages of coolant flow.

During normal operation, the coolant flow through the reactor core experiences various stages, including subcooled, pseudo-critical, and supercritical conditions. Each of these stages affects the neutron flux distribution and the moderation of neutrons differently. For instance, in the subcooled region, the coolant density is higher, leading to more effective neutron moderation and a different reactivity response compared to the supercritical region, where the coolant density is lower.

These variations in coolant flow stages can lead to differences in the temperature reactivity coefficients, impacting the overall reactivity control and stability of the reactor. Understanding and accurately modeling these differences is crucial for ensuring the safe and efficient operation of the SCW-SMR. Advanced neutronic simulations and experimental validations are necessary to characterize the temperature reactivity coefficients across all stages of coolant flow, enabling the ECC-SMART project to optimize the reactor design and enhance its safety features.

Ensuring an adequate reactivity reserve is a critical issue for the SCW-SMR, particularly when considering the use of High-Assay Low-Enriched Uranium (HA-LEU) or Mixed Oxide (MOX) fuel. The reactivity reserve is essential for maintaining the reactor's ability to sustain a controlled nuclear reaction over its operational cycle. HA-LEU and MOX fuels offer potential benefits in terms of fuel efficiency and waste reduction, but they also introduce uncertainties in reactivity management. These uncertainties stem from differences in neutron flux distribution, fuel burnup rates, and the behavior of fission products compared to traditional low-enriched uranium (LEU) fuel.

The use of HA-LEU or MOX fuel requires detailed neutronic analyses to accurately predict the reactor's behavior over time. These analyses must account for the unique characteristics of these fuels, such as higher initial reactivity and different isotopic compositions. Additionally, the impact of fuel composition on safety margins, control rod effectiveness, and overall reactor kinetics must be thoroughly evaluated. Advanced modelling and simulation tools are essential for addressing these uncertainties, enabling the ECC-SMART project to optimize fuel management strategies and ensure a reliable reactivity reserve throughout the reactor's lifecycle. By understanding and mitigating these uncertainties, the project can enhance the safety and efficiency of the SCW-SMR, leveraging the advantages of advanced fuel types while maintaining robust control over reactor operations.

Shaping the power profile within the SCW-SMR is a complex issue that requires careful consideration of the reactor's fuel assembly (FA) design. Achieving an optimal power distribution is crucial for maximizing fuel utilization, maintaining thermal efficiency, and ensuring the reactor's safety. In the context of the SCW-SMR, this often necessitates the use of a large number of different fuel assemblies with varying enrichments and configurations. The diversity in fuel assemblies helps to flatten the power profile, reducing peak power densities and minimizing the risk of hot spots that could lead to fuel damage.

The need for a variety of fuel assemblies introduces several challenges. Firstly, it complicates the fuel management strategy, requiring precise planning and coordination to ensure that each assembly contributes effectively to the desired power profile. Additionally, the manufacturing and quality control processes must be robust enough to produce fuel assemblies with consistent performance characteristics. This diversity also impacts the reactor's operational flexibility, as the reactivity and burnup characteristics of each assembly must be carefully monitored and managed throughout the fuel cycle.

Advanced modelling and simulation tools are essential for designing and optimizing the power profile of the SCW-SMR. These tools can simulate the behavior of different fuel assemblies under various operating conditions, allowing engineers to predict and adjust the power distribution within the reactor core.

Optimizing moderation within the SCW-SMR is a critical issue that directly impacts the reactor's efficiency and safety. Effective moderation is essential for maintaining the desired neutron flux and achieving optimal fuel utilization. In the context of the SCW-SMR, this often involves adjusting the FA design or lattice pitch (sometimes referred to as the distance between fuel assemblies) to achieve better moderation. By increasing the lattice pitch, a greater volume of moderator (supercritical water) can interact with the neutrons, enhancing the moderation effect.

Another strategy involves lowering the moderator temperature, which can improve the moderation efficiency. Supercritical water at lower temperatures has a higher density, which increases its moderating capability. However, maintaining lower moderator temperatures in a supercritical water environment presents significant engineering challenges, requiring precise control of the reactor's thermal-hydraulic conditions.

Alternatively, design modifications to the fuel assemblies themselves can be considered. This might include altering the geometry of the fuel rods or incorporating advanced materials that enhance neutron moderation. These design changes must be carefully evaluated to ensure they do not adversely affect the reactor's thermal-hydraulic performance or safety margins.

Developing an effective refuelling strategy is a crucial aspect of the SCW-SMR's operation, directly impacting its efficiency, safety, and economic viability. The refuelling strategy must ensure that the reactor maintains a consistent and optimal power output while minimizing downtime and operational disruptions. For the SCW-SMR, this involves carefully planning the timing and sequence of fuel assembly replacements to maintain a balanced and stable core reactivity.

One approach to refuelling is the use of a batch refuelling strategy, where a portion of the fuel assemblies is replaced at regular intervals. This method allows for continuous operation with periodic shutdowns for refuelling, ensuring that the reactor remains within safe operational limits. The batch refuelling strategy must be optimized to balance the burnup of the fuel, ensuring that the remaining fuel assemblies continue to operate efficiently.

Another strategy is the use of a continuous or on-line refuelling system, which allows for the replacement of fuel assemblies without shutting down the reactor. This approach can significantly enhance the reactor's operational flexibility and availability, reducing downtime and improving overall efficiency. However, implementing an on-line refuelling system in a supercritical water reactor presents significant technical challenges, requiring advanced handling and insertion mechanisms to ensure safety and reliability.

The choice of refuelling strategy also depends on the reactor's core design and the characteristics of the fuel assemblies. Advanced modelling and simulation tools are essential for evaluating different refuelling strategies, allowing engineers to predict the impact on core reactivity, power distribution, and fuel utilization.

In conclusion, addressing the neutronic issues identified in WP4 is essential for optimizing the performance and safety of the SCW-SMR. Key challenges include the understanding of temperature reactivity coefficients, managing uncertainties related to reactivity reserves with advanced fuels, shaping the power profile with diverse fuel assemblies, optimizing moderation through design modifications, and developing effective refuelling strategies. By leveraging advanced modeling and simulation tools, conducting rigorous experimental studies, and implementing innovative design solutions, the ECC-SMART project can enhance the thermal-hydraulic performance of the SCW-SMR, ensuring its reliable and efficient operation under supercritical conditions.

2.4.5 Legislation issues

Navigating the complex landscape of legislation is crucial for the successful development and deployment of the SCW-SMR. This subchapter will explore the key legislative issues identified in the ECC-SMART project, focusing on the regulatory frameworks, licensing processes, and compliance requirements that govern nuclear reactor design and operation based on [MAZZ01]. Understanding and adhering to these legislative requirements is essential for ensuring the reactor meets all safety, environmental, and operational standards. This section will address the challenges associated with harmonizing international and national regulations, obtaining necessary permits and licenses, and engaging with stakeholders to build public trust and acceptance. By addressing these legislative issues, the ECC-SMART project aims to pave the way for the safe and efficient deployment of the SCW-SMR, contributing to the advancement of sustainable nuclear energy technologies.

One of the primary legislative challenges facing the development of the SCW-SMR is that European legislation is not specifically prepared to address the unique features of SMR technologies. Current regulatory frameworks and safety standards are predominantly designed for large power units, such as the European Pressurized Reactor (EPR) and the VVER-1000. These regulations focus on the complexities and risks associated with large-scale nuclear reactors, which differ significantly from those of SMRs.

SMRs, including the SCW-SMR, offer distinct advantages such as modularity, enhanced safety features, and reduced environmental impact. However, the existing legislative framework does not fully accommodate these benefits, often imposing requirements that are more suited to large reactors. This misalignment can lead to unnecessary regulatory hurdles and increased costs for SMR projects, potentially hindering their development and deployment.

To address this issue, it is essential to adapt and update European legislation to reflect the specific characteristics and safety profiles of SMRs. This includes developing tailored safety standards, licensing processes, and compliance requirements that recognize the inherent safety features and operational flexibility of SMRs. Engaging with regulatory bodies, policymakers, and industry stakeholders is crucial for driving these legislative changes and ensuring that the regulatory environment supports the advancement of SMR technologies. By aligning legislation with the unique features of SMRs, the ECC-SMART project can facilitate the efficient and safe deployment of the SCW-SMR, contributing to the diversification and sustainability of Europe's nuclear energy landscape.

Another significant legislative issue for the SCW-SMR is the lack of specific nuclear standards to cover supercritical applications. Current nuclear standards are primarily designed for conventional reactors and do not adequately address the unique challenges posed by supercritical water reactors, which operate under high pressure, high temperature, and intense neutron irradiation conditions.

The SCW-SMR operates at pressures and temperatures significantly higher than those of traditional reactors, which introduces new safety and material integrity concerns. High-pressure conditions require robust containment systems and advanced materials that can withstand the extreme environment without compromising safety. Similarly, high temperatures necessitate the use of materials with excellent thermal stability and resistance to thermal fatigue.

Moreover, the combination of high pressure, high temperature, and neutron irradiation presents a unique set of challenges that are not fully covered by existing standards. Neutron irradiation can cause material embrittlement, swelling, and other degradation mechanisms that must be carefully managed to ensure the long-term reliability of reactor components. The lack of specific standards for these conditions means that the SCW-SMR must rely on a combination of existing standards and additional safety margins, which can complicate the design and regulatory approval process.

To address these gaps, it is essential to develop new nuclear standards that specifically address the requirements of supercritical water reactors. This includes establishing guidelines for material selection, design criteria for high-pressure and high-temperature components, and safety assessments that account for the combined effects of neutron irradiation. Collaboration with international regulatory bodies, research institutions, and industry stakeholders is crucial for developing these standards and ensuring they are based on the latest scientific and technical knowledge.

The implementation of SMRs in the European nuclear power plant portfolio should begin with the adoption of the graded approach and practical elimination concepts. These concepts are essential for tailoring regulatory requirements to the specific characteristics and safety profiles of SMRs, ensuring that they are both effective and proportionate.

The graded approach involves applying different levels of regulatory scrutiny and safety measures based on the potential risks and complexities of the reactor design. For SMRs, which typically have enhanced safety features and lower risk profiles compared to large reactors, this approach allows for more flexible and efficient regulatory processes. It ensures that the regulatory requirements are commensurate with the actual risks, avoiding unnecessary burdens while maintaining high safety standards.

The practical elimination concept focuses on identifying and eliminating potential accident scenarios that could lead to significant radioactive releases. By designing SMRs with inherent safety features and robust mitigation measures, the likelihood of such scenarios can be practically eliminated. This concept is crucial for gaining public and regulatory confidence in the safety of SMRs.

In the Czech Republic, for example, the decree 329/2017 [CR-D329] covers the graded approach, emphasizing the need for a conservative application. This decree provides a framework for applying the graded approach to nuclear safety, ensuring that all potential risks are adequately addressed while allowing for flexibility in regulatory requirements.

In conclusion, addressing the legislative issues is crucial for the successful development and deployment of the SCW-SMR within the European nuclear landscape. The current European legislation, primarily designed for large reactors, does not fully accommodate the unique features and advantages of SMR technologies, necessitating updates to regulatory frameworks. Additionally, there are significant gaps in nuclear standards for supercritical applications, particularly concerning high pressure, high temperature, and neutron irradiation conditions. The adoption of the graded approach and practical elimination concepts, along with clear definitions and guidelines for passive systems, is essential for integrating SMRs into the European NPP portfolio. By resolving these legislative issues, the ECC-SMART project can create a supportive regulatory environment that enables the efficient and safe deployment of the SCW-SMR, enhancing its technical capabilities and building confidence among stakeholders, regulatory bodies, and the public.

3 SCW-SMR Concept

The principle of Defence in Depth (DiD) has been applied for the development and design of advanced nuclear reactor designs and for the evaluation and improvement of existing reactors for decades. However, the application of the existing DiD considerations for novel SMR designs is a challenge for designers because of the gradually different safety features of these projects. Novel designs can result in different classification of DiD levels – even with the exclusion of core melt or large release events – or in new approaches for engineering barriers.

This chapter describes the safety features of SCW-SMR at each level of (DiD). As there is no available conceptual design for the SCW-SMR at this time, mainly general considerations are listed below, based on our present knowledge and the results of WP2-4. After the general description of the DiD levels, the special aspects of the DiD principle for SMR reactors and especially for the SCW-SMR are introduced. The main goal of this task is to link together the levels of DiD with the safety functions.

The operational and accident states of the SCW-SMR and their main features are described, in order to get a clear idea, which operational modes and accident conditions the reactor has to face.

After listing a possible set of safety functions for the SCW-SMR, the association of these safety functions and the suggested plant states is presented. This gives a basis for development of detailed requirements for the given safety functions.

The short description of systems related to operation and safety already developed/used for safety demonstrations in ECC-SMART project are described as well.

3.1 Defence in Depth

According to the Glossary of the IAEA [IAEA-Glossary], the defence in depth (DiD) is:

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

The principle of DiD has been applied for the development and design of advanced nuclear reactor designs and for the evaluation and improvement of existing reactors for decades. The method of having multiple physical or engineered barriers between the radioactive sources and the environment dates back to the very first applications of nuclear energy. Later, this system of multiple barriers has been supplemented with the methods and safety provisions applied for the protection of the barriers. Recently, usually four physical barriers (the fuel matrix, the fuel cladding, the primary coolant boundary and the containment) and five levels of DiD are used in large nuclear power plants.

According to the definition above [IAEA-Glossary], the principle of defence in depth includes not only the technological and safety systems and provisions, but also the procedures and other administrative solutions for the maintenance of safety and for the mitigation of accident consequences.

However, the application of the existing DiD considerations for novel SMR designs is a challenge for designers because of the gradually different safety features of these projects. The advanced design can result in different number of barriers or in new methods and safety provisions at different levels. Some SMR concepts are able to exclude core melting or off-site radiological consequences, changing the traditional structure of defence in depth.

According to the [IAEA-Glossary], the objectives of defence in depth are:

(a) To compensate for human induced events and component failures;

(b) To maintain the effectiveness of the barriers by averting damage to the facility and to the barriers themselves;

(c) To protect workers, members of the public and the environment from harm in accident conditions in the event that these barriers are not fully effective.

The Fundamental Safety Principles [IAEA-SF-1] states that: "*Defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people or to the environment. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. When properly implemented, defence in depth ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability. The independent effectiveness of the different levels of defence is a necessary element of defence in depth*."

The International Nuclear Safety Group (INSAG) defined five levels of defence in depth:

(a) Level 1: Prevention of abnormal operation and failures.

(b) Level 2: Control of abnormal operation and detection of failures.

(c) Level 3: Control of accidents within the design basis.

(d) Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents.

(e) Level 5: Mitigation of radiological consequences of significant releases of radioactive material.

The purposes of the levels are discussed in SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]:

(a) The purpose of the first level of defence is to prevent deviations from normal operation and the failure of items important to safety.

(b) The purpose of the second level of defence is to detect and control deviations from normal operation in order to prevent anticipated operational occurrences from escalating to accident conditions.

(c) The purpose of the third level of defence is to prevent damage to the reactor core and releases of radioactive material requiring off-site protective actions and to return the plant to a safe state by means of inherent and/or engineered safety features, safety systems and procedures.

(d) The purpose of the fourth level of defence is to prevent the progress of, and to mitigate the consequences of, accidents that result from failure of the third level of defence by preventing accident sequences that lead to large release of radioactive material or early release of radioactive material from occurring.

(e) The purpose of the fifth and final level of defence is to mitigate radiological consequences of a large release of radioactive material or an early release of radioactive material that could potentially result from an accident.

The full independence of the DiD levels mentioned above can be achieved only theoretically. In practice, the “reasonably achievable” independence of the levels is applied, meaning the common use of equipment and other provisions[3] for some of the levels. For example, in current operating nuclear power plants, usually the same normal control systems are used for level 1 and 2. However, it is widely accepted and anticipated to use fully independent systems for level 3, covering the design basis accidents, but there are also some common systems for their first 3 levels (such as the residual heat removal system in some designs). The independence of level 4 systems depends on the specific design of the reactor.

Table 3.1 *Design provisions for the different DiD levels in advanced large PWRs*

	Level	Systems	Procedures / further provisions
1	Prevention of abnormal operation and failures	Normal operational surveillance and control systems	• Conservative design, high level construction and operation • Normal operating procedures; operational conditions and limits (OLC)
2	Control of abnormal operation and detection of failures	Normal operational surveillance and control systems; limiting and protection systems (e.g. reactor protection systems)	• Periodic testing material inspection • Preventive / periodic maintenance • Normal operating procedures; operational conditions and limits (OLC)
3	Control of accidents within the design basis	Engineered safety systems, ESFs (e.g. emergency core cooling systems)	• Emergency operating procedures (EOP) • High reliability ESFs through special design solutions (single failure criterion, redundancy, diversity, fail-safe design, etc.)
4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary safety systems (e.g. severe accident management systems)	• Severe accident management procedures • Application of non-conventional systems and procedures
5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response	• National and local emergency response system • Protective measures for workers and population

3 According to the definition of [GIV04], the term "Provision" is used to indicate specific feature which is an integral part of the safety architecture. Provisions include technical provisions and/or organizational measures (i.e. the safety architecture, the security architecture). Technical safety provisions include: structures, active and/or passive systems, and components. The operational provisions include: operating rules; technical specifications; inservice inspection; normal, incident and accident procedures, etc..

It is worth to note that the role of independence of DiD level has been considered even more important since the accident of the Fukushima Daiichi NPP.

According to [WENRA01], the independence of systems, structures and components (SSCs) can be achieved, if the SSCs are able to perform their safety functions independently from the operation or failure of other SSCs needed on other DiD levels **and** independently from the occurrence of the effects resulting from the postulated initiating event (internal or external), for which they are required to function.

This independence can be achieved by the application of:

- diversity;
- physical separation, structural or by distance;
- functional isolation.

The levels of defence are sometimes grouped into three safety *layers*: hardware, software and management control. Table 3.1 shows a summary of the generally applied systems and procedures for the levels.

3.2 Plant states of present advanced nuclear power reactors

The IAEA Glossary [IAEA-Glossary] defines operational states and accidental conditions for nuclear reactors. Operational states include the normal operation (incl. refuelling, maintenance, power changes, etc.) and AOOs (anticipated operational occurrences). Accident conditions include design basis accidents (DBA) and their extension to DEC conditions without and with significant core degradation.

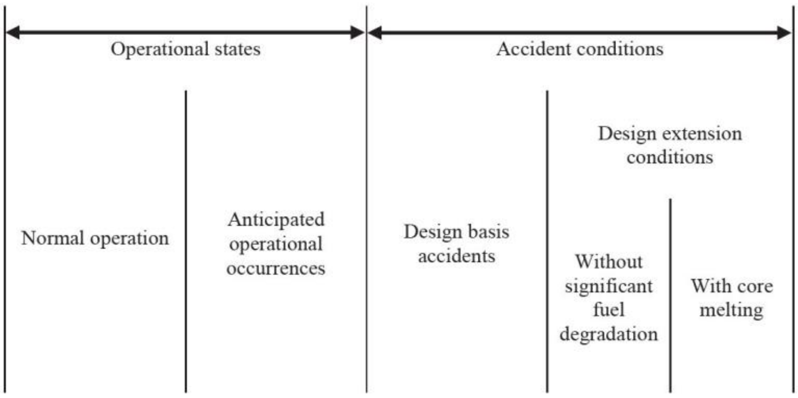

Fig. 3.1 *System of plant states according to the IAEA Glossary* [IAEA-Glossary]

INSAG-10 [INSAG-10] suggested that the frequency of events with severe core damage should be below 10^{-5}/year for future nuclear power plants, with the practical elimination of accident sequences that could lead to large early radioactive releases. In practice, the first requirement is equal to the frequency of occurrence of “design extension conditions with core melt” plant states, usually called DEC-B or DEC-2. The referred INSAG-10 also suggests that consequences of severe accidents that could imply late containment failure would necessitate only protective measures limited in area and in time.

Table 3.2 *Possible anticipated operational occurrences and design basis accident categories used in some States for new reactors* [IAEA-SSG-2]

Plant state	Alternative names used in some States	Indicative frequency range (per year)
Anticipated operational occurrences	Faults of moderate frequency: DBC-2, PC-2	$f > 10^{-2}$
Design basis accidents	Infrequent faults: DBC-3, PC-3 Limiting faults: DBC-4, PC-4	$10^{-2} > f > 10^{-4}$ $10^{-4} > f > 10^{-6}$

Note: DBC — design basis condition; PC — plant condition. The designations DBC-1 and PC-1 are used for normal operation. Some other accidents for which the frequency is $<10^{-6}$ need to be considered because they are representative of a type of risk from which the reactor has to be protected.

Table 3.3 *Plant states for advanced large PWRs*

	Short description [IAEA-Glossary]	Remarks
Normal operation	Operation within specified operational limits and conditions	Including power operation, power changes, startup and shutdown process of the reactor, refuelling, testing
Anticipated operational occurrences (AOO)	A deviation of an operational process from normal operation that is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.	Including the individual failure of a normal plant system or failure of a control system. Typical examples are turbine trips or loss of the offsite power.
Design Basis Accidents (DBA)	A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits.	Typical examples are loss of coolant accidents (LOCA), loss of feedwater or main steam line break accidents. Design reactivity accidents are classified as DBA as well.
Design Extension Conditions (DEC) with / without core melting	Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.	Design conditions without core melting include the complex failure of multiple systems (such as station blackout or loss of ultimate heat sink). Design conditions with core melting include severe accidents with significant core damage or melt. Examples for such events are LOCA accidents without available emergency core cooling systems.

Besides this high-level requirement, the regulation of some countries define further frequency limits for the different events leading to DBC states. For example, the frequency of events leading to AOO shall be less than 10^{-2}/year (reflecting the expectance of occurrence of such events once during the lifetime of the plant). For further examples see Table 3.3.

3.3 Application of the defence in depth concept for SMR designs

The basic document for the application of DiD is the INSAG10 (Defence in Depth in Nuclear Safety, [INSAG-10]) released by the IAEA 25 years ago. However, the recent changes in the field of nuclear reactor development made it necessary to update this basic document considering the novel advanced reactor types (such as SMRs and non-water cooled designs).

The IAEA published an addendum this year to the INSAG-10, titled Application of the Principle of Defence in Depth in Nuclear Safety to Small Modular Reactors [INSAG-28]. This addendum applies for the following SMR types:

(1) Land based water cooled SMRs;
(2) Marine based water cooled SMRs;
(3) High temperature gas cooled SMRs;
(4) Liquid metal cooled fast neutron spectrum SMRs;
(5) Molten salt SMRs;
(6) Microreactors.

In this respect, SCW-SMR may belong to group (1), although supercritical water-cooled reactors are not specified in [INSAG-28] directly.

[INSAG-28] reminds to the fact, that new advanced reactors may have a core damage / large early release frequency much lower than the suggested limits of 10^{-5}/year and 10^{-6}/year, respectively, adding that the uncertainties for such low values can cause difficulties.

The smaller radioactive core inventory results in greater possibilities in using passive safety systems for residual heat removal, for some designs even with the possibility of air cooled RHR systems. However, there are some safety disadvantages for SMRs, such as the question of shared (safety) systems of multiple units or the possibility of common cause failures as the result of lack of proper physical separation.

[INSAG-28] emphasizes that for some novel reactor type Level 4 considerations may need cautiousness, as the elimination of core melt scenarios can induce a new approach for definition of this level. This should be justified with an in-depth analysis. At the same time, this justification does not apply for level 5, which is not related to the plant technology, but to the protection of the society. This can be established based on the uncertainties for level 3 and 4.

The application of DiD for SMRs draws attention to the human aspects as well (as it is part of the defence in depth concept). The plans for unprecedented siting and production modes – such as marine-based plants or district heating / industrial heat plants – necessitate the re-evaluation of safety culture knowledge management.

The compactness and the modular arrangement of SMRs can cause some challenge for the ensuring of reasonable independence of DiD levels, but the importance of the issue needs to be recognized.

Table 3.4 *DiD levels applicability for Pressurized Water SMRs* [ANSARI]

DiD Level	PWR SMR features [ANSARI]
Level 1	• Negative reactivity coefficients • Elimination of liquid boron reactivity control system • Relatively low core power density • Integral design of primary circuit with in-vessel location of steam generators and (hydraulic) control rod drive mechanisms • Compact modular design of the reactor unit • Primary pressure boundary in pressurized, low enthalpy containment • Leaktight reactor coolant system with internally immersed pumps • A single, small diameter double connecting line between the primary coolant pressure boundary and auxiliary systems • Natural circulation-based heat removal from the core in normal
Level 2	• Active systems of instrumentation and control • Negative reactivity coefficients over the whole cycle • A relatively large coolant inventory in the primary circuit, resulting in large thermal inertia • High heat capacity of nuclear installation as a whole • Favourable conditions for implementation of the leak before break concept, through design of the primary circuit • Little coolant flow in the low temperature pressurized water containment enclosing the primary pressure boundary • Redundant and diverse passive or active shutdown systems
Level 3	• Negative reactivity coefficients over the whole cycle • Relatively low core power density and primary coolant temperature • Large thermal inertia with large coolant inventory in primary circuit • High heat capacity of nuclear installation as a whole • Primary pipelines being connected to the hot part of the reactor • Use of once-through steam generators • A dedicated steam dump pool located in the containment building • Self-pressurization, large pressurizer volume, elimination of sprinklers, etc. • Limitation of inadvertent control rod movement by an overrunning clutch and by the limiters • Low heat-up rate of fuel elements predicted in a hypothetical event of core uncovery, owing to design features • Passive emergency core cooling, often with increased redundancy and grace period • Passive system of reactor vessel bottom cooling • Natural convection of water in flooded reactor cavity following small LOCA • Dedicated pool for steam condensation under a steam generator tube rupture • Low enthalpy pressurized water containment embedding the primary pressure boundary or double containment
Level 4	• Very low leakage containment; elimination or reduction of containment vessel penetrations • Reasonably oversized reactor building with passive cooling system • Relatively small, inert, pressure suppression containment • Reduction of hydrogen concentration in the containment by catalytic recombiners and selectively located igniters • Sufficient floor space for cooling of molten debris; extra layers of concrete to avoid containment basement exposure directly to such debris
Level 5	• Mainly administrative measures • Relatively small fuel inventory, less nonnuclear energy stored in the reactor, and lower integral decay heat rate • Design features of Levels 1–4 could be sufficient to achieve defence in depth Level 5

According to [INSAG-28], the principle of graded approach shall be used for the assessment of SMR safety as well. This takes the magnitude of risks arising from the given facility into consideration – resulting in more strict requirements for larger NPPs. There is some experience gained with the application of graded approach for research reactors, having smaller core inventory. These experiences can be used for the establishment of graded approach for SMR designs as well.

Because of the large number of SMR designs and the great technology variability of them the application of the safety principles described above shall be evaluated on a case-by-case methodology. As an example, [ANSARI] describes the applicability of the levels of defence in depth for PWR SMRs (see Table 3.4), listing the possible design provisions for the different levels.

3.4 Application of DiD levels for SCW-SMR

In the following section the special features for the DiD levels of SCW-SMR concept are listed, based on the considerations described above. The safety goals formulated in D5.1 and the lessons learned from the D5.2 (describing the safety features of the design) and D5.3 (describing the applicable safety requirements in a pre-licensing study) are taken into account as well.

Level 1 - Prevention of abnormal operation and failures

For the Level 1, the prevention of failures necessitates a conservative design, high level construction and operation of the reactor. The conservative design means a well-established, thorough design for reactor physics, thermal hydraulics, material selection (e.g. reactivity coefficients; neutron flux distribution inside the core; coolability of the fuel assemblies; behaviour of fuel during the burnout process; etc.).

Table 3.5 *Level 1 of Defense in Depth for SCW-SMR*

Defence in Depth Level 1 Prevention of abnormal operation and failures	
Level 1 related features of SCW-SMR	**Reference**
Negative reactivity coefficients during cycle	[ECC-D4.3]
Use of burnable poison materials	[ECC-D4.3]
Maximum fuel enrichment: 10%	[ECC-D4.3]
Elimination of liquid boric acid reactivity compensation system	[ECC-D3.3]
Decreased peak factors	[ECC-D3.3]
Selection of proper cladding material (corrosion)	ECC-SMART Project WP2
Selection of proper cladding material (heat transfer)	ECC-SMART Project WP3
Coolability of FAs in normal operation	[ECC-D3.3]
Natural circulation RHR system is possible	[ECC-D3.3]
Hydraulic control rod drives	[HPLWR]
Containment?	[HPLWR]
Primary pumps	[HPLWR]
Coolant outlet temperature (target)	[ECC-D3.3]
etc.	

The systems applied for the level include the normal operational surveillance and control systems, such as measurements or design of control and safety rods or other reactivity control systems.

The further provisions for the level include the normal operational procedures, based on the establishment of operational conditions and limits (OLC).

At this development phase, no operating procedures are available, and instead of OLCs, only some target values for normal operations (such as core outlet temperature) are available. Concerning the normal operating systems, there is some preliminary design for the fuel rods and assemblies, reactor core and reactor vessel, however, no further details are known for normal operating (NO) systems. Where applicable, other considerations are made, based on the HPLWR technology or other literature review.

Further remarks:

- Unlike PWR SMRs, in SCW-SMR integral design is not applicable as it has only one cooling circuit.
- The volumetric power density of the core is much lower than the one for large PWRs (similarly to BWR reactors);
- At this point, the design of normal operation surveillance and control systems are not available.
- Possibility of modular arrangement (with possibly shared systems) are not known yet.
- Containment system design is not available yet

Level 2 - Control of abnormal operation and detection of failures

For the Level 2, the purpose is to detect and control deviations from normal operation in order to prevent anticipated operational occurrences from escalating to accident conditions. The main task of this level is to keep the unit within the established operational limits and conditions (OLC), with the help of surveillance, control and reactor protection systems. In most present reactors, Level 1 and Level 2 systems are shared.

Table 3.6 *DiD Level 2 for SCW-SMR*

Defence in Depth Level 2 Control of abnormal operation and detection of failures	
Level 2 related features of SCW-SMR	**Reference**
Negative reactivity coefficients	[ECC-D4.3]
Large coolant inventory in the RPV (?), large thermal inertia	[ECC-D3.3]
Large amount of metal structures inside the RPV, large thermal inertia	[ECC-D3.3]
Hydraulic reactor shutdown system	[HPLWR]
Diverse shutdown system with boric acid injection	[HPLWR]
Thermal margins (linear heat rate, heat flux, etc.)	[ECC-D4.3]
Behaviour of the core in case of AOOs	ECC-SMART Project WP4
Behaviour of fuel cladding during lifetime / in case of AOOs	ECC-SMART Project WP2
Thermal hydraulic behaviour of the reactor in case of AOOs	ECC-SMART Project WP3
etc.	

The systems applied for the level include the normal operational surveillance and control systems, such as measurements or design of control and safety rods or other reactivity control systems. Additionally, reactor protection systems (such as SCRAM functions) are included. Periodic testing and material investigations are part of this level, as they can help to *detect* deteriorating parameters before getting out of OLCs. Non-destructive material testing methods belong here as well.

The further provisions for the level include the normal operating procedures, administrative systems for surveillance, maintenance and ageing management programs.

As described above, at this point of the development, no design for normal operating systems is available, therefore only features resulting from the core design or derived from other literature review (such as the HPLWR design) can be listed in Table 3.6.

Further remarks:

- No data available about instrumentation and control (I&C) yet;
- No data available about the possible application of leak before break concept
- Although boric acid long term reactivity control is eliminated from the design, the application of boric acid as a diverse shutdown system is possible.

Level 3 - Control of design basis accidents

The aim of the Level 3 of the DiD concept is the proper management of design basis accidents in order to prevent fuel damage and to limit radioactive release into environment. For this purpose, a main task is to maintain the soundness of as many barriers as possible. In some countries, Level 3 includes also design extension conditions without core melting[4]. (i.e., Level 3 includes DBC3, DBC4 and DEC1 conditions as well.) This classification reflects the approach that the distinction between the levels is based on the status of the fuel (accidents with core melt belong to Level 4).

Systems for Level 3 are traditionally the most important engineered safety features (ESFs) of the plant, representing high reliability and strict design principles, such as single failure criteria etc. These systems include emergency heat removal systems and the emergency electricity supply as well. For SCW-SMR a preliminary sketch of safety systems was suggested by T. Schulenberg and improved further for calculations in D3.3. The proposed safety systems include the following items:

- The reactor shut down system by control rods or by a boron injection system as a second, diverse shut down system.
- Containment isolation by active and passive containment isolation valves (CIV) in each line penetrating the containment to close the third barrier in case of an accident.
- Steam pressure limitation by pressure relief valves (PRV).
- Automatic depressurization system (ADS) of the steam lines into a pool inside the containment through spargers to close the coolant loop inside the containment in case of containment isolation.
- An emergency core cooling system (ECCS) to refill coolant into the pressure vessel after intended or accidental coolant release into the containment.
- A pressure suppression pool (PSP) to limit the pressure inside the containment in case of steam release inside the containment.
- A residual heat removal system for long term cooling of the containment.
- Hydro accumulators
- Gravity driven injection systems

Further provisions for Level 3 include procedures for accident management (Emergency Operating Procedures – EOPs). These procedures are usually developed as state-based OPs,

[4] An other approach for classification is based on the design basis / design extension condition. This results in classification of DEC1 (or DEC A) into Level 4 together with DEC2.

keeping the key parameters of the plant between some limits, ensuring that radioactive releases can be kept below authorized limits as well.

One of the most important tasks during the development is the definition of the design basis of the reactor, i.e. what external and internal hazards should be taken into consideration for the design. Further remarks:

- The cooling system of the SCW-SMR has not been developed yet, although it will have great effect on the arrangement and operation of safety systems applied for Level 3 accidents (ESFs).
- As the coolant circuit is one of the engineering barriers, the design will have an effect on it as well.
- The reliability expectations for safety systems, determining the necessary redundancy levels is not known.
- The containment and its connecting systems have not been developed yet, although the design will have an effect on the arrangement and operation of safety systems applied for Level 3 accidents (ESFs). Containment is also an engineering barrier.

Table 3.7 *DiD Level 3 for SCW-SMR*

Defence in Depth Level 3 Control of design basis accidents	
Level 3 related features of SCW-SMR	**Reference**
Negative reactivity coefficients over the whole cycle	[ECC-D4.3]
Relatively low core power density and coolant temperature	[ECC-D3.3]
Large thermal inertia with large coolant inventory in primary circuit	[ECC-D3.3]
High heat capacity of nuclear installation as a whole	[ECC-D3.3]
Emergency shutdown system with hydraulic system	[ECC-D3.3]
Diverse emergency shutdown system by boron injection	[ECC-D3.3]
Steam pressure limitation by pressure relief valves (PRV)	[ECC-D3.3]
Automatic depressurization (ADS) of the steam lines	[ECC-D3.3]
Pressure suppression pool in the containment	[ECC-D3.3]
Passive isolation condenser system	[ECC-D3.3]
Containment isolation by active and passive containment isolation valves (CIV)	[ECC-D3.3]
Emergency core cooling system (ECCS) to refill coolant into the pressure vessel	[ECC-D3.3]
Hydro accumulators	[ECC-D3.3]
Gravity driven injection system	[ECC-D3.3]

- For management of multiple failure events (Level 3.b.) further systems and further provisions may be necessary, such as additional emergency electricity supply systems or passive long-term residual heat removal systems.
- According to D3.3, some preliminary safety analyses have already been performed by UNIPI and BME. Both institutes have performed analysis for long-term station blackout scenario (LTSBO), which is usually considered as an initiating event leading to DEC A conditions in novel reactors.
- Simulations of UNIPISA performed with RELAP/SCDAPSIM code gave an insight into the dynamic behaviour of SCW-SMR. The results showed that a minimal configuration of passive systems may be sufficient for natural circulation heat removal, ensuring the core cooling during a postulated LTSBO.

- Simulations performed by BME with the help of the APROS system code, gave similar results for LTSBO scenarios. However, the resolution of BME model is more refined compared to the UNIPISA's one, so it may have a greater accuracy.

Level 4 - Management of severe accidents

The aim of Level 4 is to manage severe accidents in order to mitigate the consequences of the accident and to prevent the large or early releases. As it was stated previously, complex accidents (DEC-A) without core melt can be classified to Level 3 or Level 4, depending on the strategy of DiD application. Based on the recommendation of WENRA [WENRA], in this report we classify DEC-A events to Level 3, so Level 4 stands for severe accidents only.

According to the safety objective formulated by WENRA [WENRA] for severe accidents, "only limited protective measures in area and time are needed for the public and that sufficient time is available to implement these measures". Safety systems and provisions at Level 4 aim to ensure that only limited consequences can occur even in case of severe accidents. The prevention of large release shall be ensured in the long term (i.e. as long as it is necessary; this can be months or even years).

According to WENRA, severe accidents include not only the significant melting of the fuel, but also other significant fuel degradation, which can lead to radioactive release into the environment exceeding the dose limits for public and workers.

Systems for Level 4 include the safety systems necessary to prevent the loss of the integrity of the containment (defined as the last barrier between radioactivity and the environment), including the following systems:

- Corium management systems – the localization and cooling of melted fuel is inevitable for the severe accident management. This can be achieved by retaining the melted core inside the reactor vessel with an in-vessel retention system (IVR), i.e. outer cooling of the vessel, or with ex-vessel corium management, i.e. using a core catcher system.
- Residual heat removal system for long term containment cooling and pressure reduction of the containment.
- Pressure reducing system for the management of non-condensable gases (e.g. containment venting system).
- Hydrogen management system for prevention of hydrogen explosion.

Other SAM systems – among others – provide severe accident measurements and SA electricity supply.

Further provisions for Level 4 aim to help to prevent or decrease large radioactive release in case of severe accidents, including SAM procedures.

Provisions also include the practical elimination of such accidents which would result in radioactive release requiring protective actions for the public exceeding limited effects in area and time, and accidents with early radioactive release (i.e. not leaving sufficient time for protective measures for the population) shall be also practically eliminated. In this respect, practical elimination means that the given scenario is either physically impossible or it can be proved with proper confidence that it's probability is extremely low.

Table 3.8 *DiD Level 4 for SCW-SMR*

Defence in Depth **Level 4** **Management of severe accidents**	
Level 4 related features of SCW-SMR	**Reference**
SS cladding material results in lower hydrogen generation	[YADAV01]

Further remarks:

- Severe accident mitigation strategy for SCW-SMR is not decided yet. Both in-vessel corium retention (with outer RPV flooding) and core catcher solutions are feasible, however, SMR designs tend to apply IVR methods.
- Hydrogen mitigation system is not designed yet, however, H2 production is expected to be less because of different cladding material. However, passive autocatalytic hydrogen recombiners or hydrogen igniters are definitely necessary for SA management.
- Containment features, including passive long term residual heat removal and management of non-condensable gases are not available yet. It can be assumed, that – similarly to other novel SMR designs – large amount of water will be available inside the containment for passive heat removal, while the containment passive cooling will be solved with the help of additional heat exchangers (cooled with air or with an external water storage tank).
- Severe accident behaviour of the core will be determined by the amount of structural material in the reactor vessel, which is considered to be significantly more than in other SMR designs.

In the framework of WP3, the severe accident analysis was performed for SCW-SMR by IPP [ECC-D3.3]. The investigated transient was a LB LOCA event without available ECCS system. The study included three phases of accident progression:

- Stage 1 - CFD calculation of the initial dynamics of the transient caused by the LOCA event, covering the pressure wave propagation caused by the LB LOCA, simulation of depressurization and transition from supercritical to subcritical state, resulting in pressure and temperature fields as input for the 2nd stage.
- Stage 2 – Corium relocation calculated by the MELCOR severe accident code for the SCW-SMR concept with horizontal core layout.
- Stage 3 - Calculation of the late phase of the severe accident, based on complete structural elements molten materials characteristics (temperatures and component composition) obtained in Stage 2. Parametric analysis of the minimum time to RPV failure depending on the external cooling conditions.

This method was applied because of certain modelling difficulties. For the horizontal channels in the reactor core, the classical MELCOR code models were not directly applicable. With the help of a CFD code it was possible to evaluate the loads on the internal vessel structures caused by the depressurization from supercritical pressure, while the MELCOR code was adopted to consider the phases of core degradation, corium relocation and vessel failure.

Based on the results, it can be stated that the existence of supercritical state coolant in the core does not mean a great difference compared to traditional light water reactors concerning the severe accident behaviour of the core. However, the application of horizontal fuel assemblies results in quite different corium relocation processes. In D3.3 [ECC-D3.3] the in-vessel retention has been investigated as well, showing the necessity of forced external cooling for the reactor vessel (or the consideration of a core catcher concept instead).

Level 5 - Management of off-site consequences

The Level 5 of the DiD concept represents somewhat different approach as it deals with the management of off-site consequences of a severe accident in order to protect the population from radiation exposure, assuming that systems and other provisions at Level 4 failed. That is, Level 5 does not cover any design-related technical issue, safety system or other provisions, but the local / national emergency preparedness and response systems.

Systems for Level 5 include the necessary radiation monitoring systems, tools for radiation dispersion and dose evaluation, and the necessary software and hardware for protection of the population (such as stable iodine).

Further provisions for Level 5 include the whole emergency preparedness and response system of the given country, covering the necessary legal systems, human and financial conditions for ensuring the emergency management, national and international notification systems and regular emergency drills.

However, some specific features of SMR reactors (and so for the SCW-SMR) can make some emergency preparedness and response activities unnecessary. With practical elimination of off-site consequences protective zones can be decreased or even eliminated.

Table 3.9 *DiD Level 5 for SCW-SMR*

Defence in Depth Level 5 Management of off-site consequences	
Level 5 related features of SCW-SMR	**Reference**
Relatively small fuel inventory, lower decay heat rate	[ECC-D3.3]

Further remarks:

- Design features of Levels 1–4 could be sufficient to achieve defence in depth Level 5

3.4.1 Successive physical barriers for SCW-SMR

An important task of DiD is to ensure the integrity of successive physical barriers, placed between radioactive material and the environment. For the SCW-SMR, the generally used four barriers can be applied with some further consideration:

(1) fuel matrix – traditional UO2 and MOX fuel is considered for SCW-SMR, however, the enrichment is expected to be significantly higher than for present reactors (up to 7-8%);
(2) fuel cladding – instead of the traditional Zr alloy cladding, stainless steel and Ni-Cr-Fe alloy cladding are planned, which will change the accident behaviour and failure processes of the fuel rod;
(3) primary system – much higher primary pressure than PWRs, no design available yet;
(4) leaktight containment building – may be similar to advanced BWR containments, no design available yet.

3.5 Safety functions of advanced reactors

According to the [IAEA1570], the fundamental safety functions for a nuclear power plant are: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases. These fundamental safety functions shall be ensured for all plant states.

Fundamental safety functions can be broken down into further, technology-specific safety functions. [IAEA-SG-46] gives an example of such detailed safety function list for large light-water reactors.

(1) To prevent unacceptable reactivity transients;

(2) To maintain the reactor in a safe shutdown condition after all shutdown actions;

(3) To shut down the reactor as necessary to prevent AOOs from leading to DBAs and to shut down the reactor to mitigate the consequences of DBAs;

(4) To maintain sufficient reactor coolant inventory for core cooling in and after accident conditions not involving the failure of the reactor coolant pressure boundary;

(5) To maintain sufficient reactor coolant inventory for core cooling in and after all postulated initiating events considered in the design basis;

(6) To remove heat from the core after a failure of the reactor coolant pressure boundary in order to limit fuel damage;

(7) To remove residual heat in appropriate operational states and in accident conditions with the reactor coolant pressure boundary intact;

(8) To transfer heat from other safety systems to the ultimate heat sink;

(9) To ensure necessary services (such as electrical, pneumatic, and hydraulic power supplies and lubrication) as a support function for a safety system;

(10) To maintain acceptable integrity of the cladding of the fuel in the reactor core;

(11) To maintain the integrity of the reactor coolant pressure boundary;

(12) To limit the release of radioactive material from the reactor containment in accident conditions and conditions following an accident;

(13) To limit the radiation exposure of the public and site personnel in and following DBAs and selected severe accidents[5] that release radioactive materials from sources outside the reactor containment;

(14) To limit the discharge or release of radioactive waste and airborne radioactive materials to below prescribed limits in all operational states;

(15) To maintain control of environmental conditions within the plant for the operation of safety systems and for habitability for personnel necessary to allow performance of operations important to safety;

(16) To maintain control of radioactive releases from irradiated fuel transported or stored outside the reactor coolant system, but within the site, in all operational states;

(17) To remove decay heat from irradiated fuel stored outside the reactor coolant system but within the site;

(18) To maintain sufficient subcriticality of fuel stored outside the reactor coolant system but within the site;

(19) To prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of an SF.

[5] Including also multiple failures without core melt

(20) To maintain the integrity of the reactor containment in accident conditions and conditions following an accident.

(21) To limit the effects of releases of radioactive materials on the public and environment.

According to our evaluation, all of the above safety functions apply for SCW-SMR as well. However, the content of given safety functions may differ from the typical content meant for large LWRs. For example, addressing heat transfer deterioration can be managed in a similar way as for critical heat flux issues in PWR reactors, as part of other, more general safety functions, such as SF6 and SF7 in the list above.

Safety functions can be grouped based on different considerations: according to their fundamental safety function, or the DiD Level or plant states they are necessary to operate. In a later phase of design (conceptual design) safety systems and further provisions can be matched with the proper safety functions. This serves as the basis of safety classification of operational and safety systems, which determines the reliability requirements against these systems.

Fig. 3.2 [IAEA-SG-46] shows the classification of safety functions based on the main (fundamental) safety function they belong to.

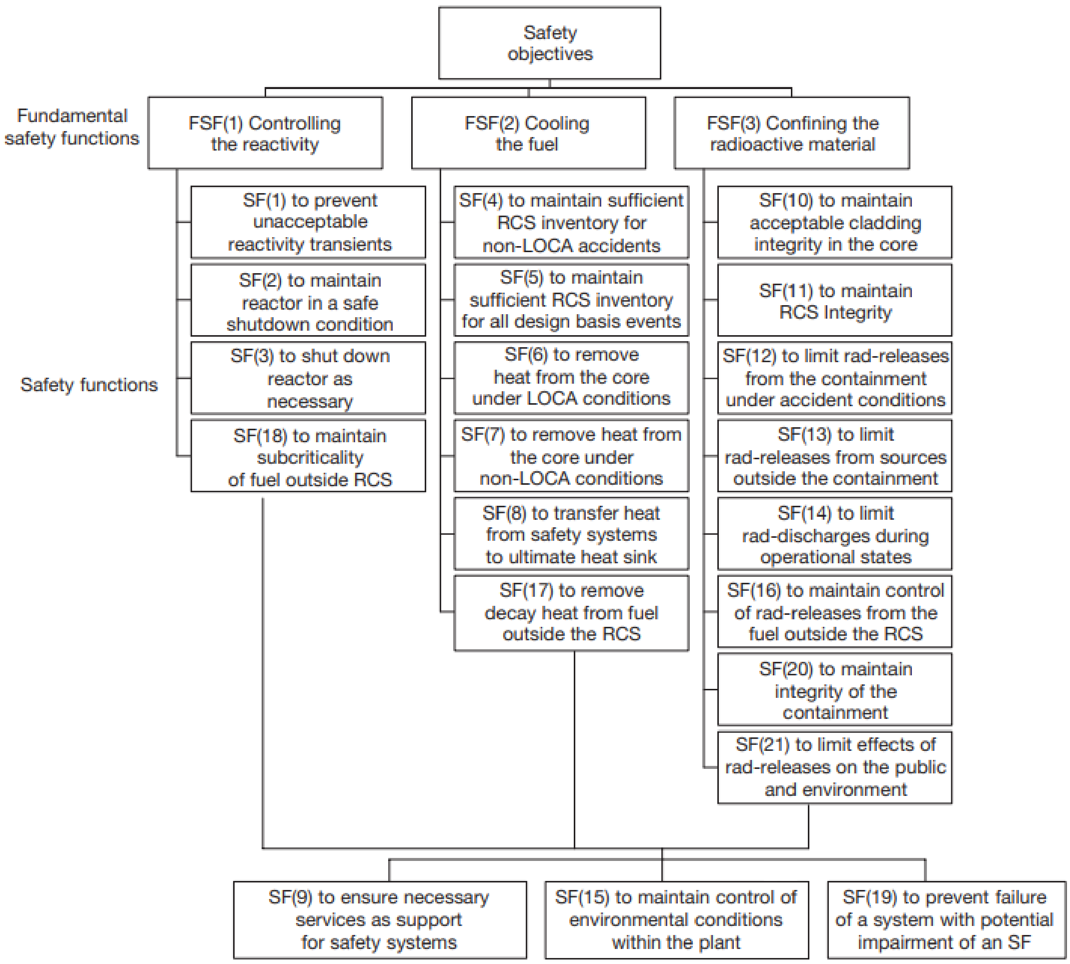

Fig. 3.2 *Overview and grouping of safety functions for the evaluation* [IAEA-SG-46]

3.6 Operational Conditions for SCW-SMR

For the definition of plant states of SCW-SMR, the categories for light-water reactors described above (see chapter “Plant states of present advanced nuclear power reactors”) and the suggested plant states for HPLWR [HPLWR] were taken as a starting point.

3.6.1 Operational conditions of HPLWR

For the HPLWR design, the following plant states have been defined, in accordance with the present suggestions for advanced large PWR reactor units [HPLWR]:

Table 3.10 *Suggested plant states for HPLWR* [HPLWR]

Plant state	Features
1. Normal operation (DBC1) controlled by operating systems	• Conservative design, reliability, availability • Proven technology, quality assurance
2. Operational occurrences (DBC2, $>10^{-2}$/year) controlled by control and limitation features	• Surveillance, diagnostics • Inherent safety, nuclear stability
3. 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
4. Design Extension 1 (DEC1): Multiple failure scenarios (e.g. station blackout, total loss of feedwater, loss of coolant accidents), severe external events (e.g. military or large commercial airplane crash)	• Diversified systems • Design against external event loads
5. Design Extension 2 (DEC2): Severe Accidents	• Mitigative features • Prevention of energetic consequences which could lead to large early containment failure (e.g. steam explosion, direct containment heating, global hydrogen detonation)

3.6.2 Plant states of SCW-SMR

Based on the general description and HPLWR states described above, the planned plant states for SCW-SMR have been defined. Table 3.10. shows these states and the corresponding DiD level.

The suggested states can be applied not only for the reactor itself but also for spent fuel stored in SFP or in an on-site interim storage.

3.7 Association of safety functions and plant states for SCW-SMR

During the next development stages, design of SCW-SMR shall be developed further keeping in mind that the three fundamental safety functions shall be met in all plant conditions.

The detailed safety functions (presented above based on [IAEA-SG-46]) can be associated one-by-one to the suggested plant states (see Table 3.11), so design requirements can be developed for the given function for the necessary plant state. This association for SCW-SMR is presented in Table 3.12, showing that most of the safety functions are to be met in all plant states.

Table 3.11 *Plant states for SCW-SMR*

Plant state	Abbreviation	Frequency of events leading to the given state (per year)	Corresponding DiD level	Remarks
Normal operation (NO)	DBC1		Level 1	Normal operation includes power change, startup / shutdown process, outages, refuelling, fuel manipulation / transportation as well.
Anticipated operational occurrences (AOO)	DBC2	$<10^{-2}$	Level 2	
Design basis accidents (DBA)	DBC3 DBC4	$10^{-2} > f > 10^{-4}$ $10^{-4} > f > 10^{-6}$	Level 3a	The separation of the DBA conditions into DBC3 and DBC4 states reflects the method applied in some EU countries (see Table 3.2.)
Design extension condition (without core melt)	DEC1		Level 3b	
Design extension condition (with core melt)	DEC2		Level 4	The frequency of possible severe accidents (i.e. accidents with core melt / significant core damage) is limited in some EU countries. Severe accident can have only limited off-site consequences. For novel SMR designs practical elimination of off-site consequences may be a design requirement.

Table 3.12 *Association of safety functions and plant states for SCW-SMR (the table may be the subject of further modifications during next development stages)*

Safety functions	DBC1[1]	DBC2	DBC3	DBC4	DEC1	DEC2
(1) To prevent unacceptable reactivity transients;	o	o	o	o	o	o
(2) To maintain the reactor in a safe shutdown condition after all shutdown actions;	o	o	o	o	o	o
(3) To shut down the reactor as necessary to prevent AOOs from leading to DBAs and to shut down the reactor to mitigate the consequences of DBAs;		o	o	o		
(4) To maintain sufficient reactor coolant inventory for core cooling in and after accident conditions not involving the failure of the reactor coolant pressure boundary;			o	o	o	
(5) To maintain sufficient reactor coolant inventory for core cooling in and after all postulated initiating events considered in the design basis;	o	o	o	o	o	
(6) To remove heat from the core after a failure of the reactor coolant pressure boundary in order to limit fuel damage;			o	o	o	
(7) To remove residual heat in appropriate operational states and in accident conditions with the reactor coolant pressure boundary intact	o	o	o	o	o	o[2]
(8) To transfer heat from other safety systems to the ultimate heat sink;	o	o	o	o	o	o
(9) To ensure necessary services (such as electrical, pneumatic, and hydraulic power supplies and lubrication) as a support function for a safety system;	o	o	o	o	o	o[3]
(10) To maintain acceptable integrity of the cladding of the fuel in the reactor core;	o	o	o	o	o	
(11) To maintain the integrity of the reactor coolant pressure boundary;	o	o	o[5]	o[4]	o[5]	
(12) To limit the release of radioactive material from the reactor containment in accident conditions and conditions following an accident;			o	o	o	o
(13) To limit the radiation exposure of the public and site personnel in and following DBAs and selected DEC accidents that release radioactive materials from sources outside the reactor containment;			o	o	o	o
(14) To limit the discharge or release of radioactive waste and airborne radioactive materials to below prescribed limits in all operational states;	o	o	o	o	o	o
(15) To maintain control of environmental conditions within the plant for the operation of safety systems and for habitability for personnel necessary to allow performance of operations important to safety;	o	o	o	o	o	
(16) To maintain control of radioactive releases from irradiated fuel transported or stored outside the reactor coolant system, but within the site, in all operational states;	o	o	o	o	o	o
(17) To remove decay heat from irradiated fuel stored outside the reactor coolant system but within the site;	o	o	o	o	o	o
(18) To maintain sufficient subcriticality of fuel stored outside the reactor coolant system but within the site;	o	o	o	o	o	o
(19) To prevent the failure or limit the consequences of failure of a structure, system or component whose failure would cause the impairment of an SF.	o	o	o	o	o	o
(20) To maintain the integrity of the reactor containment in accident conditions and conditions following an accident.			o	o	o	o
(21) To limit the effects of releases of radioactive materials on the public and environment.	o	o	o	o	o	o

[1] Including power change / shutdown / outage periods

[2] Considering in-vessel retention corium management strategy.

[3] Not necessary for passive safety systems
[4] For non-LOCA events
[5] ECCS may include steam-driven high pressure injection pumps, medium pressure hydro accumulators and low pressure gravity-driven injection.

3.8 Safety Systems of SCW-SMR

For novel reactor designs, usually passive means are designed as safety systems for DBC and DEC conditions. [ECC-D3.3] describes the suggested minimum set of safety systems for SCW-SMR in details. Therefore, here only a short summary is given about the necessary safety systems for the three main safety functions. The arrangement of the systems is shown in Fig. 3.3.

For the control of reactivity the following safety systems are necessary:
- safety rods for normal emergency shutdown;
- emergency boron injection for rapid shutdown in case of unavailability of safety rods.

For residual heat removal in all plant conditions:
- passive emergency condenser system for the heat removal from the reactor (this might be either an air-cooled or a water-cooled condenser);
- in-containment water storage for safety systems;
- emergency core cooling systems (ECCS) for loss-of-coolant accidents;
- automatic depressurization system (ADS) for making low-pressure injection possible.

For the localization of radioactive materials:
- containment isolation valves (CIV);
- depressurization system for the coolant system (PRV);
- pressure suppression pool (PSP) for containment pressure control.

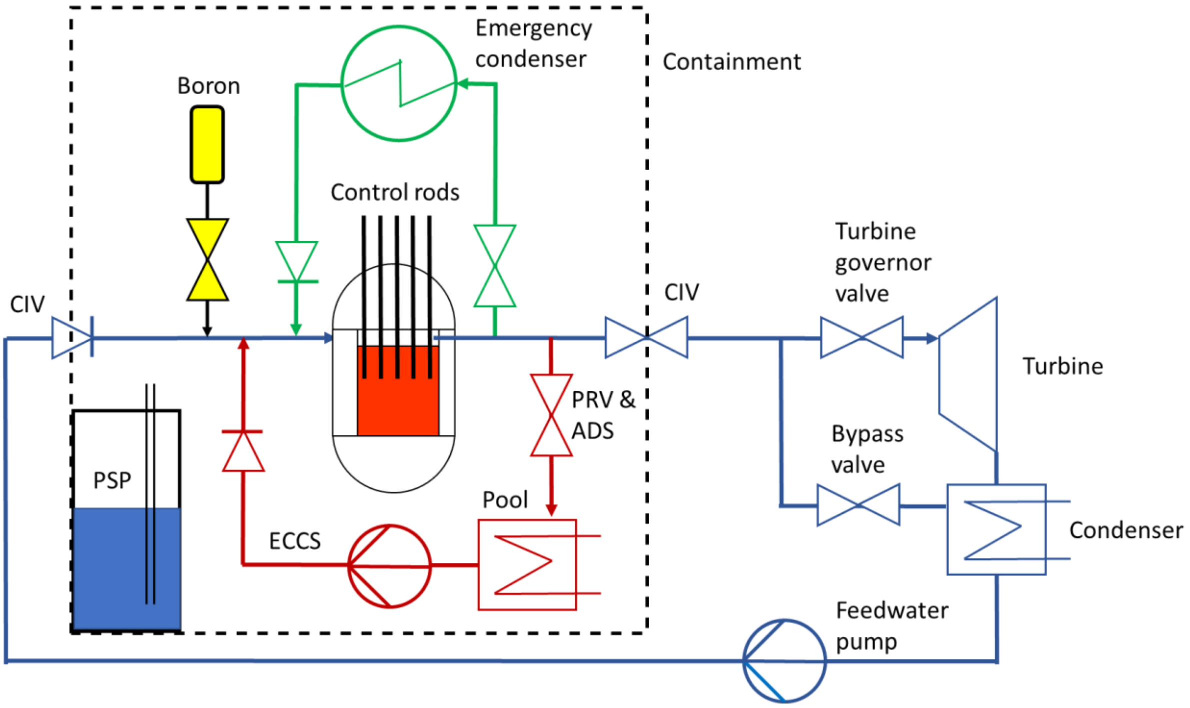

Fig. 3.3 *Minimum set of safety systems suggested in* [ECC-D3.3]

4 Requirements for Safety Demonstration

This chapter provides guidelines for the safety requirements, safety criteria and methods for safety demonstration, needed for future demonstration of SCW-SMRs.

In the section 4.1 on safety requirements, the preparation for pre-licensing is described first. Preparation for pre-licensing is typically done in parallel with conceptual design. An example of NUWARD (PWR-SMR under development in France by EDF) is given and GIF safety approach (note that safety approach is important part of safety assessment [IAEA-GSR-P4], including defence in depth, safety margin and multiple barriers, to which safety criteria are related). Then guidelines and instructions regarding applicability of Gen IV goals and WENRA safety objectives of SMR are given, and recommendations regarding IAEA standards for design.

In the section 4.2 on safety criteria their role is described in the relation to safety requirements. Then major criteria for the three barriers are described, i.e. fuel safety criteria, primary circuit criteria, and containment criteria, followed with example of U.S. NRC hierarchical licensing requirements and their evaluation criteria.

Finally, in the section 4.3, methods of safety demonstration are described. Summary of the GIF goals in developing SMR reactor with particular attention on SCW SMR, summary of legislation from IAEA down to the local legislation, the concept of practical elimination, and guidelines for the definition of preliminary safety report are given.

4.1 Safety Requirements

In D5.3 pre-licensing study [ECC-D5.3] the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 were related 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. This was done separately for the following international legislation or guidance:

- IAEA relevant standards (SF-1 [IAEA-SF-1], SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]
- OECD/NEA fuel safety criteria (23 criteria) [OECD01]
- OECD/NEA GIF safety and reliability goals (judgement of applicability to SCW-SMR) [GIV01]
- WENRA safety objectives for new NPPs (judgement of applicability to SCW-SMR) [WENRA01]

Fig. 4.1 shows typical hierarchy of regulatory rules. D5.1 covers most Level 1a and Level 1b requirements and criteria. Level 1a WENRA requirements are part of regulatory harmonisation in Europe and countries are obliged to implement WENRA requirements in national legislation. Level 1b are top level IAEA safety standards. It should be noted that IAEA safety standards are not legislation but reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. Countries are not obliged to implement IAEA requirements into national legislation, but they may do this. There are three categories of IAEA standards: safety fundamentals, safety standards and safety guides. Level 2 are nuclear process-oriented documents (e.g. IAEA safety guides), while Level 3 documents are presented in section 5.2. Level 4 comprise of conventional codes and standards, not considered in this report.

Levels of rules	Description
1a	**Level 1a: Legislation & safety regulations** • Country legislation (examples of Canada, China, Czech Republic, Finland and United Kingdom) • WENRA, Safety of new NPP designs • WENRA, Applicability of the Safety Objectives to SMRs
1b	**Level 1b: IAEA (International Atomic Energy Agency) Safety Standards** • IAEA SF-1, Fundamental Safety Principles • IAEA SSR-2/1 Rev. 1 Safety of Nuclear Power Plants: Design
2	**Level 2: Nuclear process oriented documents** • Quality assurance (e.g. ISO 19443:2018, IAEA GS-G-3) • Design and operation (e.g. guidance documents of countries, including IAEA general specific guides (GSG) and specific safety guides (SSG))
3	**Level 3: Nuclear component oriented documents** • Pressure boundary codes and standards • Codes and standards for electrical equipment • Operations and maintenance codes
4	**Level 4: Conventional codes and standards** • Usually applied to the structures, systems and components of conventional facilities • Conventional pressure vessel codes and standards

Fig. 4.1 *Typical hierarchy of regulatory rules*

The conceptual design of SCW-SMR itself is not the objective of the ECC-SMART project. For comparison, in case of ASTRID nuclear island at the end of the pre-conceptual design phase the design [LADUR01], [SAEZ01] include the primary and secondary circuits and their components (intermediate heat exchanger, pump, steam generator, sodium-gas heat exchanger), the components handling systems, the fuel handling systems, safety related elements (core catcher, decay heat removal), the vault, sodium auxiliary systems, and gas auxiliary systems. Therefore, instead of compliance of design with the requirements, the requested information needed to judge the compliance of future SCW-SMR conceptual design to safety requirements and criteria has been provided in D5.3 [ECC-D5.3].

The SF-1 [IAEA-SF-1] safety objective and safety principles form the basis for deriving the IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] 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 comply with safety requirements established in IAEA safety standards, the compliance to IAEA SF-1 [IAEA-SF-1] objective and principles will be confirmed. In large extent, also WENRA objectives [WENRA02] will be satisfied, as they were derived from IAEA safety principles. However, as proposed by GIF, the Integrated Safety Assessment Methodology (ISAM) [GIV04] is intended to support the achievement of a safety that is "built-in" rather than "added on", coherent with the objectives and principles applicable to achieve a safe Generation IV design. As shown for NUWARD SMR conceptual design, the Safety Options file (called the Dossier d'Options de Sûret´e (DOS)) has been submitted to the French regulator for their opinion in July 2023 [NUW01]. Simultaneous consideration, on one side, of the safety objectives and principles and guidelines and, on the other hand, of other additional criteria (like the economy, ease of operation or maintenance, availability or absence of a significant feedback experience), lead to the definition of "safety options" for the selection and the detailed organization of provisions that build up the safety architecture [GIV04]. In the pre-licensing phase, it should be justified that options allow getting close to the safety goals and meeting the safety objectives (i.e. not to demonstrate the compliance with requirements and criteria). As stated in [GIV04] "design safety option" is the way (i.e. the design strategy) to perform the mission required to meet the objective(s) of the safety function(s). It means that is not the implemented solution itself. In such manner, safety objectives could be judged for conceptual design. Instead of final design, the design strategy is judged in the early phases. This will be more explained in the following Section 4.1.1, dealing with preparation for pre-licensing.

Guidelines and instructions for the future demonstration of the safety of the SCW-SMR concept are drafted in Section 4.1.2.

4.1.1 Preparation for pre-licensing

First an example of NUWARD SMR pre-conceptual and conceptual design will be briefly presented, and preparation for pre-licensing, which run in parallel to the design process. Next, safety approach by GIF will be presented, which requires implementation of a re-examined agreed safety approach for innovative reactor design and assessment in comparison to light water SMRs, which design has been developed from a foundation in large PWR design.

4.1.1.1 Example of NUWARD SMR pre-conceptual and conceptual design, and preparation for pre-licensing

In the NUWARD [NUW01] example preparation for pre-licensing was going on during pre-conceptual design and conceptual design. Pre-conceptual design aims at investigating innovative options that could be integrated into the reactor. At this stage of the process, the team is not looking to ensure coherence between the options nor provide a finalised conceptual design. Conceptual design started with choosing the reference options and aims at providing a coherent, finalized conceptual design. The NUWARD conceptual design [NUW01] suggested general site layout, a presentation of nuclear island building internals, where each reactor is enclosed in metallic (steel) containment. The NUWARD SMR design main reactor components are housed inside the RPV, this includes the pressuriser, the six Compact Plate Steam Generators (CSGs) and two Safety-CSGs (S-CSGs), the 76 fuel assemblies that make up the fuel core and the Control Rod Drive Mechanism (CRDM) that sits above them. The foundation of the NUWARD SMR safety approach is embedded in internationally accepted nuclear safety guidance, with positioning in relation to international recommendations, in particular IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1], Regulators Association (WENRA) reports [WENRA02], the European Utilities Requirements (EURs) [EUR01] and applicable French regulation, in particular ASN guide no. 22 for design [ASN01]. Initial objectives for NUWARD SMR were WENRA SMR [WENRA02] qualitative objectives O2 (Accidents without core melt) and O3 (Accidents with core melt). Additionally, the NUWARD SMR project adopts a design objective to reduce the need for protective measures for accidents with core melt with design target for no need for off-site protection measures) [NUW01]. The basis of NUWARD SMR safety approach is Defence-in-Depth, considering independence among DiD levels as far as practicable. The NUWARD SMR reactor and associated safety systems are designed to manage design basis accidents (DBAs) passively for more than 3 days, and complex sequences, called DEC-A actively with simple diagnosis and implementation of diversified systems. For severe accident management (called DEC-B) In-Vessel Retention of the corium (IVR concept) is considered in the design.

The documents presenting foundation of NUWARD SMR safety approach support the D5.1 deliverable [ECC-D5.1], which identify all above safety guidance with the exception of France national legal framework (discussed were safety requirements in Canada, China, Czech Republic, Finland and United Kingdom). Instead of ASN guide no. 22 for design, Canadian REGDOC-2.5.2 [CNSC01] document for design was considered, which sets out requirements and guidance for new license applications for water-cooled nuclear power plants (NPPs or plants). It establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted international codes and practices. To a large degree, the REGDOC 2.5.2 [CNSC01] regulatory document represents the CNSC's adoption of the principles set forth in IAEA SSR-2/1 (rev. 1) [IAEA-SSR-2/1] as adapted to align with Canadian requirements. EUR is also recommended in D5.1 [ECC-D5.1]. The novelty of NUWARD SMR project is that they use EUR Chapter 5 [EUR02], and they group EUR 26 Key Positions (KP) on SMLWR into following five topics: A Safety, B Systems and Components, C Performance, D Operations and Maintenance, E Cost and Constructability. Safety includes the following seven KP: KP1 – probabilistic design targets, KP2 – emergency planning zone, KP3 – defence-in-depth approach, KP4 – complex sequences, KP5 – autonomy objectives, KP 6 – external hazards and KP 7 – safety of multi-module units.

Main topics discussed during the Joint Early Review (JER) of NUWARD [NUW02] were:
Topic 1. The general safety objectives.

Topic 2. The list of design basis conditions and design extension conditions.

Topic 3. The use of passive cooling systems.

Topic 4. The development plan for computer codes.

Topic 5. The integration of two reactor units in a single facility.

Topic 6. The Probabilistic Safety Assessment (PSA) approach.

Topics 1 to 4 were at least indirectly considered in D5.3 [ECC-D5.3]. For general safety objectives (topic 1) it was concluded that WENRA objectives for SMR is minimum and that NUWARD SMR should take advantage of the possibilities offered by its design to aim for a further reinforcement of safety objectives. For example, targeting for accidents with core melt, the WENRA O2 objective is currently applied for accidents without core melt.

Regarding DBC and DEC (topic 2), it was observed that the Czech, Finnish and French approaches for the categorization of these events were different. Nevertheless, the general process to identify DBC scenarios to be considered in the safety demonstration of the NUWARD SMR reactor looks globally consistent with the regulators’ expectations.

Regarding the use of cooling passive systems (topic 3), it was considered that the designer should highlight the specificities of passive systems that may require to adapt the methodology of implementation of safety principles in the safety demonstration, and highlighted the importance of the reliability assessment of passive systems.

Regarding the qualification of the computer codes (topic 4), the working group observed that the approach and development plan proposed for the NUWARD SMR are generally consistent with the regulatory frameworks of the three countries involved. However, computer codes shall be qualified for each application and validated, with particular emphasis on the experimental validation matrix.

4.1.1.2 GIF Safety Approach for Design

The design of current evolutionary plants (Gen III) is based on past experience without putting into question the major principles established for the safety architecture[6]. Their safety demonstration is achieved in a deterministic way, supplemented by probabilistic methods and appropriate research and development work [GIV01]. For example, the NUWARD SMR design [NUW01] has been developed from a foundation in large PWR design, construction and operation, drawing on knowledge from its partners (notably on integrated reactor designs) and using proven Light Water Reactor (LWR) technology, whilst also integrating some key innovations. The systems selected by the Gen IV initiative shows a large variety of technologies, issues and options to address these issues. This variety justifies the implementation of a re-examined agreed safety approach for their design and assessment [GIV01].

RSWG document on the basis for safety approach [GIV01] has been revised by [GIV02]. The primary objective of [GIV02] was to discuss GIF safety goals, safety principles and evaluation methodology of the next generation systems. It is also intended to motivate the need for innovative safety approach. The RSWG [GIV02] "*believes that an optimally effective approach to ensuring the safety of Generation IV nuclear facilities and systems must be based on a well-developed safety philosophy that applies to both design and operation. Such a safety philosophy must be much more than just a collection of prescriptive design requirements*. *In fact, it is preferred that the safety philosophy not be prescriptive in nature at all, but rather should articulate the desired objectives and principles applicable to achieve a safe Generation IV design*." The proposed safety approach is to strengthen the implementation of defence-in-depth, increase the robustness and transparency of the safety demonstration, and continually improve the safety culture.

The safety approach should keep coherence with the following criteria [GIV02]:

- "*agreement with current and the - foreseen - future regulations,*
- *ability to prove the full implementation of the defence in depth: prevention, detection and control of the abnormal situations, mastery of the accidents, management of severe plant conditions and mitigation of their consequences, and potential off-site measures,*
- *allowing for the installation's design / analysis to manage simultaneously deterministic practices and probabilistic objectives,*
- *ability to handle internal and external hazards so as to achieve, as much as possible, the coherency with the approach adopted for internal events, i.e. in guaranteeing a common global treatment,*
- *allowing to improve the safety demonstration for the domains where gaps still exist in the current state of art, and*
- *allowing the demonstration of the achievement of a level of safety equivalent or even better with regard to the current Generation III systems*."

The adoption of these criteria should, on one hand, ensure that all Generation IV designs adhere to a consistent set of principles and, on the other hand, support the identification of the necessary crosscutting and specific R&D efforts required to validate the selection of innovative options for

[6] The full set of provisions – inherent characteristics, technical options and organisational measures – selected for the design, the construction, the operation including the shut down and the dismantling, which are taken to prevent the accidents or limit the effects [GIV01].
Note: In accordance to IAEA TECDOC 1570 [IAEA1570] the safety architecture includes: inherent plant safety features and characteristics; engineered safety features; on-site accident management procedures established by the operating organization; and off-site intervention measures established by appropriate authorities in order to mitigate radiation exposure if an accident has occurred.

these designs. Main safety principles are DiD (the ideal outcome will be a design that optimizes both capital costs and safety by applying defence in depth where it will have the desired effect), risk-informed design (PSA when combined with an consequences anticipated with postulated accidents, it can also provide risk insights), and simulation, prototyping, and demonstration (modelling and simulation to be used in the design and evaluation of complex technologies - reactor physics, thermal hydraulics, fuel performance, materials behaviour, and a number of other issues that are central to reactor design and development).

For current plants the plant conditions that are considered in the design are conventionally subdivided into Design Basis Conditions (DBC) and Design Extension Conditions (DEC). As a complement to this deterministic approach, probabilistic insights are considered for the DBC through categorization of initiating events. The results from probabilistic analyses are applied for DEC safety assessment.

For innovative systems, the design would be iterative. According to [GIV04], as shown in Fig. 4.2, based on the functions (see left side of Fig. 4.2) the mechanisms (initiating events) are identified to define the conditions the system has to deal with (postulated initiating events - PIE). The implementation of specific provisions to address these PIEs, leads to possible provisions' failures which also need to be considered.

In parallel, the definition of the controlled and safe plant states allows defining the missions which are requested and so giving the needed inputs for the provisions' design (right side of the Fig. 4.2). This is done for all the levels of the defence in depth. As a complement to the treatment of these internal events, an improved coherence with the treatment addressing internal and external hazards has to be looked for.

The design process as shown in Fig. 4.2 needs to integrate regulator principles, recommendations and guidelines [GIV04]. In this context the designer develops his safety approach, that is: defines the strategy, chooses safety goals and objectives as well as the safety options which form the base of the architecture. Once the safety approach is defined and the situations which have to be considered for the design basis identified, the construction of the safety architecture can begin with the selection and the sizing of provisions to be implemented.

As can be seen from footnote 6, safety architecture also includes inherent plant safety features and characteristics; and engineered safety features, which were in the scope of D5.2 [ECC-D5.2] and D5.3 [ECC-D5.3].

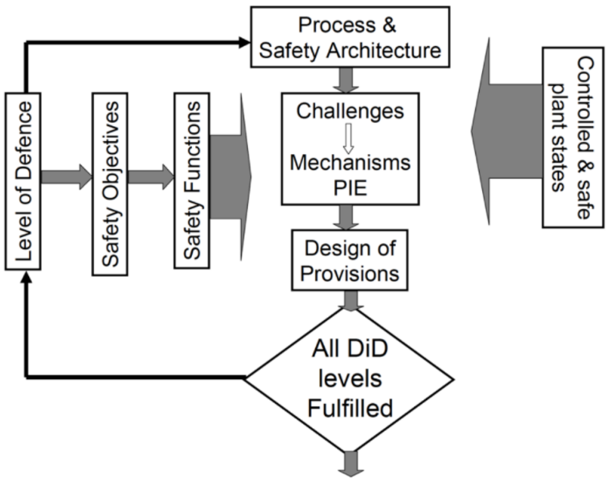

Fig. 4.2 *Iterative process for the construction of the safety architecture (Fig. IV of [GIV02])*

Fig. 4.3 is called in [GIV04] as flowchart for the design/assessment/discussions/endorsement process, scheme for the design and the implementation of the safety architecture. The iterative process for the construction of the safety architecture (Fig. 4.3) does correspond - and can be recognized to the lower part of the flowchart. The design options have to be justified against the safety goals, safety objectives, safety principle, safety requirements and safety guidelines (upper part of flowchart).

To better understand Fig. 4.3, the reader can refer to the glossary of the flowchart given in Appendix 1 of [FIOR01] for the following terms: challenges, controlled state, SSCs design criteria, design and operational safety specifications, mechanism, mission, provisions, safety architecture, safety goals, safety guidelines, safety objectives, safety options, safety principles, and safety requirements. Design Options and Provision File (DOPF) is a set of volumes whose main objective is to organize the presentation and the discussion between the designer and the safety authority on the basis of a predefined table of contents [FIOR01].

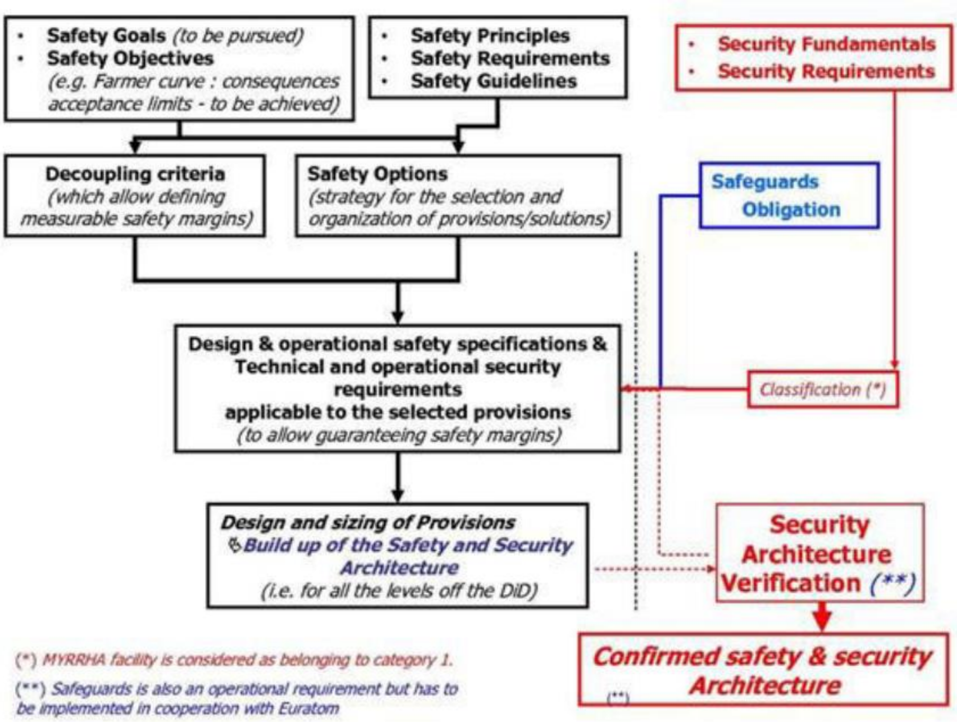

Fig. 4.3 *Relationship between the different steps and terms for the DOPF. (Fig. 1 of [FIOR01])*

4.1.2 Requirements identified for future demonstration of SCW-SMR safety

Emphasis is given to the safety requirements, for which the WP5.3 task claimed that (2) fulfilment is probable, but further research is needed, and (3) fulfilment is improbable, design changes are needed.

4.1.2.1 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 [ECC-D5.1]. It should be noted that GIF safety and reliability goals aim specifically to generation IV reactors [GIV01], while WENRA safety objectives for new NPPs [WENRA01] are upper level principles that should be applicable to all types of reactors (including SMRs) (see Fig. 4.1). Because the size or the technology or the intended use alone do not define the characteristics of the facility, but the deployment scheme has an effect, too, the applicability of the WENRA Safety Objectives to SMRs has been discussed [WENRA02]. 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, also in the frame of ECC-SMART project the applicability of GIF safety and reliability (SR) goals to future SCW-SMR conceptual design has been discussed.

SR1 goal - Operational Safety and Reliability
Generation IV nuclear energy systems operations will excel in safety and reliability.

<u>Discussion on applicability to future SCW-SMR conceptual design (see Section 5.1 of D5.3)</u>: Based on information from HPLWR design and analyses document [HPLWR] it is judged that SR1 goal [GIV01] is applicable also to future SCW-SMR conceptual design.

<u>Guidelines and instructions</u>: As shown in Sections 4.1.1.1 and 4.1.1.2, safety could be improved following the internationally accepted nuclear safety guidance like IAEA, WENRA and European Utility Requirements. Design improvement [GIV02] could be obtained by full implementation of the defence in depth (DiD). This mean that all the DiD levels have to be considered. Hazards has to be considered according to the most recent bases and knowledge (PSA, event analysis, combinations with internal events). Concerns regarding "physical protection" should be also considered. The impacts linked to the radioprotection and the environment (effluents & waste) should be minimized. Actions for the decommissioning should be considered already during design. Provisions (inherent, passive or active, procedures) dedicated to the robustness of the architecture should implemented. Finally, robustness in safety demonstration can be achieved by considering all operational conditions, validated and verified computer codes, incorporation of inherent and passive safety systems, addressing uncertainties, and with simple and reliable safety systems.

SR2 goal - Core Damage
Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.
<u>Discussion on applicability to future SCW-SMR conceptual design (see Section 5.2 of D5.3)</u>: Based on the information in D5.1 [ECC-D5.1] 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 [GIV01] is applicable to future SCW-SMR conceptual design.

<u>Guidelines and instructions</u>: In D5.1 [ECC-D5.1] the SR2 goal [GIV01] on core damage was reformulated in the following: "*The frequency of DEC-A scenarios shall be very low and its damage extension limited.*" This goal is similar to WENRA SMR objective O2. NUWARD SMR project [NUW01] consider WENRA objectives O2 and O3 to SMR [WENRA02] as initial objectives. Future SCW-SMR conceptual design should also be similarly ambitious.

SR3 goal - Offsite Emergency Response
Generation IV nuclear energy systems will eliminate the need for offsite emergency response.
Discussion on applicability to future SCW-SMR conceptual design: For SR3 goal [GIV01] it is stated in HPLWR [HPLWR] 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 [HPLWR].

Guidelines and instructions: In D5.1 [ECC-D5.1] the SR3 goal [GIV01] on core damage was reformulated in the following: "*DEC-B scenarios shall be eliminated*." NUWARD SMR project [NUW01] consider WENRA objectives O2 and O3 to SMR [WENRA02] as initial objectives. Namely, NUWARD SMR project adopts a further design objective to reduce the need for protective measures for accidents with core melt Design Extension Condition (DEC) B. This is design target for no need for off-site protection measures. Also, future SCW-SMR conceptual design should be similarly ambitious.

4.1.2.2 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 [WENRA02] is judged with respect to applicability of objectives to future SCW-SMR conceptual design. IAEA SF-1 [IAEA-SF-1] principles no. 3 and no. 5 to 8, which were used to ground the proposed seven WENRA SMR safety objectives O1 to O7 for new reactors [WENRA02].

O1. Normal operation, abnormal events and prevention of accidents [WENRA02]

- *"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."*

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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*[7] *which helps to reduce the probability of human errors*."

Judgement for SCW-SMR conceptual design: According to D5.2 [ECC-D5.2] 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 [WENRA02]: "*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...);*

[7] Author opinion: in this context more likely term is 'intervetion' than orginal 'invention'.

- *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.

Guidelines and instructions: This goal is similar to GIF SR1 goal entitled "Operational Safety and Reliability". Both emphasize GIF SR1 [GIV01] and WENRA SMR objective O1 [WENRA02] require enhanced safety of their designs. Therefore, guidelines and instruction of GIF SR1 are applicable also to WENRA SMR objective O1.

O2. Accidents without core melt [WENRA02]

- *"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."*

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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.

Guidelines and instructions: Both WENRA SMR objective O2 [WENRA02] and GIF SR2 goal [GIV01] focus on minimizing the risk and consequences of severe accidents. GIF SR2 goal is more oriented towards design innovation to achieve a low likelihood of core damage, and WENRA O2 is focused on regulatory safety requirements to practically eliminate core melt accidents with significant radioactive releases. NUWARD SMR project consider WENRA objectives O2 and O3 as initial objectives. Also future SCW-SMR conceptual design should be similarly ambitious.

O3. Accidents with core melt

- *"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"

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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.

Guidelines and instructions: WENRA SMR O3 [WENRA02] is more grounded in current safety practices and focuses on ensuring that even in the worst-case scenario of a severe accident, public safety measures remain limited and temporary, while GIF SR3 goal [GIV01] is more ambitious, long-term target aimed at making reactors so safe that offsite emergency responses become unnecessary. NUWARD SMR project consider WENRA SMR objectives O2 and O3 as initial objectives. Namely, NUWARD SMR project adopts a further design objective to reduce the need for protective measures for accidents with core melt Design Extension Condition (DEC) B. This is design target for no need for off-site protection measures. Also future SCW-SMR conceptual design should be similarly ambitious.

O4. Independence between all levels of defence-in-depth

- *"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."*

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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[8] [2][9]), 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 [ECC-D5.2] for the HPLWR, the design of the safety systems basically adheres to the "Defence-in-Depth" (DiD) safety principle. Also, it is stated [ECC-D5.2]: "*The situation of supercritical water-cooled SMR is*

[8] Advances in Small Modular Reactor Technology Developments. A Supplement to: IAEA Advanced Reactors Information System (ARIS). 2020 Edition. https://aris.iaea.org/Publications/SMR_Book_2020.pdf

[9] Reference [2] is [WENRA02].

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) (Reference [12] in [ECC-D5.2]*) 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.

Guidelines and instructions: It is recommended to follow iterative process for the construction of the safety architecture, as shown in Fig. 4.2. Independence among DiD levels are implemented as far as practicable in the NUWARD SMR safety strategy. It is recommended to use ISAM [GIV05], which provides the tools which will help the designer to construct a DiD architectures, and the analyst to assess the pertinence of the solutions. The ISAM tools more specifically related to safety architecture and DiD are Qualitative Safety Features Review (QSR) and Objective Provision Tree (OPT) [GIV05]. QSR is structured following the DiD levels, and provides a systematic mean of ensuring and documenting that the evolving GIF concept of design, incorporates the desirable safety attributes and characteristics. OPT can be useful in focusing and structure the analyst's identification and understanding of initiators and abnormal conditions, accident phenomenology, success criteria, and related issues.

O5. Safety and security interfaces

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

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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 [ECC-D5.1] just safety has been discussed. Enhanced security by underground location, new challenges like remote operation and transportation of modules with loaded core seems not feasible for future SCW-SMR conceptual design.

Guidelines and instructions: Paper [EVAN01] presents new technologies and new physical protection approaches that can help optimize protection costs for new facilities. According to [EVAN01] security-by-design is highly recommended to current and future nuclear reactor vendors to avoid costly retrofits, reduce long-term operational costs, and enable assessment of the effectiveness of advanced security technologies. Four technologies are described in the paper [EVAN01]: radar; video analytics (VA); light detection and ranging, known as lidar; and artificial intelligence (AI)–based detection algorithms.

O6. Radiation protection and waste management

- *"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."*

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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 [ECC-D5.2] 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.

Guidelines and instructions: WENRA SMR objective O6 [WENRA02] is similar to GIF sustainability goal 2 [GIV01], which aim is that Generation IV nuclear energy systems will minimise and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment. While both GIF sustainability goal 2 and WENRA SMR objective O6 emphasize reducing nuclear waste and minimizing environmental impacts, GIF sustainability goal 2 takes a broader approach that includes long-term strategies for waste reduction, fuel recycling, and proliferation resistance, positioning it as a more comprehensive sustainability goal.

According to [GIV06] enhanced sustainability is achieved primarily through the adoption of a closed fuel cycle including the reprocessing and recycling of plutonium, uranium and minor actinides in fast reactors and also through high thermal efficiency. This approach provides a significant reduction in waste generation and uranium resource requirements.

O7. Leadership and management for safety

- *"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."*

Description of discussion on features of SMRs that differ from the present-day reactors [WENRA02]: "*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 [ECC-D5.2] does not provide any information on leadership and management for safety. However, IAEA SF-1 [IAEA-SF-1] 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 SMR O7 objective [WENRA02] is applicable to future SCW-SMR conceptual design.

Guidelines and instructions: For leadership and safety the IAEA GSR Part 2 standard requirements could be followed [IAEA-GSR-P2]. Effective application of the requirements of IAEA GSR Part 2 [IAEA-GSR-P2] will satisfy the fundamental safety Principle 3, which states that "*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*."

4.1.2.3 Pre-licensing study results of SCW-SMR for IAEA standards

In D5.3 [ECC-D5.3] compliance to IAEA SF-1 [IAEA-SF-1] objective and principles has not been judged due to insufficient detailed information on future SCW-SMR conceptual design in D5.2 [ECC-D5.2]. Namely, D3.3 [ECC-D3.3] deliverable, which reports in greater detail the features of the reactor concept proposed by Schulenberg and Otic [SCHUL01], was released after the publication of D5.2 [ECC-D5.2].

Innovative concept of a small modular reactor proposed by Schulenberg and Otic [SCHUL01] was based on the concept of the High Performance Light Water Reactor [HPLWR]. It 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 was designed with horizontal fuel assemblies of 40 fuel rods each, including an internal water channel to improve moderation. Supercritical water was heated up in seven steps, when running through these assemblies, and the reflector around the core was used as mixing channels. A single channel analysis provided a first estimate of pressure losses, coolant and peak cladding temperatures. It was concluded that more design optimization and analyses will be needed, however, to assess the feasibility of such a concept.

D3.3 [ECC-D3.3] presents general considerations about the Canadian and Chinese reactor concepts, and reports in greater detail about the features of the concept proposed by Schulenberg and Otic [SCHUL01]. Then the results related to the study of the pre-conceptual core layout and passive safety concept for a SCW-SMR based on [SCHUL01] are summarized. The early work for model development and code improvement and application for the analysis of passive safety aspects, with particular attention to natural circulation phenomena is presented. The study of the

LB LOCA dynamics, in view of the possible vulnerability of the SCW-SMR reactor pressure vessel design is also presented.

This design requirement document shall serve as a basis for a future conceptual design project of SCW-SMR [ECC-D5.3]. 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 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.

4.1.2.3.1 Compliance of design – requested information

Table 4.1 summarizes the list requirements of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]. For pre-licensing study, presented in D5.3 [ECC-D5.3], Requirements 42-58 of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] were considered as focal points for safety assessment of future SCW-SMR conceptual design. In general, all recommendations of IAEA SSG-2 (Rev. 1) [IAEA-SSG-2] are applicable for judging the compliance with Requirement 42 of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]. SSG-52 [IAEA-SSG-52] provides recommendations on the design of the reactor core to meet the Requirements 43-46 established of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]. SSG-56 [IAEA-SSG-56] on the design of the reactor coolant system and associated systems for nuclear power plants provides recommendations on how to meet the Requirements 47-53 of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1]. SSG-53 [IAEA-SSG-53] on the design of the reactor containment and associated systems for nuclear power plants provides recommendations on how to meet the requirements 54-58 of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1].

To judge the future SCW-SMR all 82 requirements of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] are recommended to be used.

Table 4.1 *Safety criteria and requirements of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] (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)
		Requirement 12: Features to facilitate radioactive waste management and decommissioning (4.20)
Section 5: General Plant Design requirements	Design basis	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 4.2 *Safety criteria and requirements of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] (Sheet 2 of 2)*

Section of IAEA SSR 2/1 (Rev. 1)	Area	Requirement
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)
		Requirement 53: Heat transfer to an ultimate heat sink (6.19A–6.19B)
	Containment structure and containment system	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)
		Requirement 57: Access to the containment (6.25–6.26)
		Requirement 58: Control of containment conditions (6.27–6.30)
	Instrumentation and control systems	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	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)

4.1.2.3.2 Challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR – knowledge judgement

In D5.3 pre-licensing study [ECC-D5.3], the safety criteria and requirements for the SCW-SMR concept developed in WP5.1 were related 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. Emphasis is given to the safety requirements related to challenges (C) shown in Table 4.3, issues (I) shown in Table 4.4, and phenomena (P) shown in Table 4.5, for which the task WP5.3 claimed that (2) fulfilment is probable, but further research is needed, and (3) fulfilment is improbable, design changes are needed.

Table 4.3 *Challenges (C) to safety-related features derived from SCWR, HPLWR and BWR technology [ECC-D5.2]*

ID	Description
C1_SCWR	*"application of novel manufacturing processes"*
C2_SCWR	*"aggressive chemical effects of SCW in the reactor core"*
C3_SCWR	*"SCW has not been used in highly radiative environment"*
C4_SCWR	*"correctness of assumptions and extensions do require confirmation"*
C5_SCWR	*"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"*
C6_SCWR	*"material issues identified (specific issues for the investigated reactor designs)"*
C1_HPLWR	*"core power distribution is heavily influenced by the coolant density distribution through the neutron moderation parameters"*
C2_HPLWR	*"another important issue is the problem of coolant and moderator flow stability"*
C3_HPLWR	*"possible Xenon oscillation instabilities"*
C4_HPLWR	*"possible larger stresses because of the high temperature difference"*
C5_HPLWR	*"high peak cladding temperature, low fuel burn-up and high hot channel factors require further analyses"*
C2_BWR	*"large changes in neutron flux, coolant density along the FAs"*
C4_BWR	*"Special containment arrangement (drywell / wetwell)"*

Table 4.4 *SMR special issues, and WP2, WP3 and WP4 issues (I) [ECC-D5.2]*

ID	Description
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.)"*
I4_WP2	*"Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature"*
I2_WP3	*"Knowledge gap exists for: formulation of design and safety concept"*
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)"*
I5_WP4	*"set of refuelling strategy"*

Table 4.5 *WP2 and WP3 most significant phenomena (P) [ECC-D5.2]*

ID	Description
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)"*
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"*
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"*

Results of D5.3 [ECC-D5.3] for relation of IAEA SSR-2/1 (Rev. 1) [IAEA-SSR-2/1] standard requirements and criteria with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR are shown in Table 4.6. Judgement is done regarding the knowledge, indicating the research needs.

Table 4.6 *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 47	I2_WP3	FULFILMENT IS IMPROBABLE
Requirement 54	C4_BWR	FULFILLED
Requirement 55	I2_SMR	FULFILLED
Requirement 58	I4_WP2	FULFILMENT IS PROBABLE

<u>Requirement 42 partly related to P1_WP4 (see Table 4.5)</u>: 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 [ECC-D5.2]. 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.

Guidelines and instructions: In future safety analyses there is a need to have available all data needed for performing safety analyses.

Requirement 43 related to C1_SCWR (see Table 4.3): FULFILMENT IS PROBABLE
Comment: Novel manufacturing processes carry a risk due to they have a first-of-its-kind design and may introduce new phenomena occurring in the new design itself.

Guidelines and instructions: For the design careful material selection is important, which is nuclear qualified for the SCW environment (see section 5.2.2). This may require development of new standards. All structures, systems and components important to safety should be environmentally qualified. IAEA's Nuclear Harmonization and Standardization Initiative [NHSI-TG] established "Common practices on Codes and Standards" topical group. This group aims to develop a platform on information sharing, focused on identifying common practices on, among others, engineering standards, equipment qualification standards, codes and standards used in various SMR projects and advanced manufacturing.

Requirement 43 related to C2_SCWR (see Table 4.3): FULFILMENT IS PROBABLE
Comment: The aggressive chemical effects of SCW will require experimental testing and demonstration.

Guidelines and instructions: Key water chemistry issues in a supercritical-water-cooled pressure-tube reactor are described in [NT01]. For experimental testing one may refer to D2.3 [ECC-D2.3] and D2.4 [ECC-D2.4].

Requirement 43 related to C3_SCWR (see Table 4.3): FULFILMENT IS PROBABLE
Comment: SCW as a coolant in highly radiative environment requires significantly high level of experimental testing and demonstration before adopting.

Guidelines and instructions: For experimental testing one may refer to D2.3 [ECC-D2.3] and D2.4 [ECC-D2.4]. For example, the irradiation rig including specific specimens was developed, designed and manufactured in CVR.

Requirement 43 related to C5_SCWR (see Table 4.3): 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.

Guidelines and instructions: The irradiation damage was investigated in the D2.4 [ECC-D2.4]. The radiation damage in terms of displacement per atom (DPA) of the selected materials 310S, 316L, and 800H has been evaluated.

Requirement 43 related to C6_SCWR (see Table 4.3): 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.

Guidelines and instructions: The above estimate [SCHUL02] was based on a very limited amount of data, therefore incore irradiation experiments are required at SCWR conditions to validate this estimate. Additional experimental data on thermal conductivity, fuel qualification, and performance of (Th, Pu)O2 is required for its implementation in SCWR.

Requirement 44 related to C4_HPLWR (see Table 4.2): FULFILMENT IS PROBABLE
Comment: The studies in the frame of HPLWR indicated areas of design optimization for stress and deformation analyses of the reactor pressure vessel, the major reactor internals and of the assembly boxes [HPLWR]. Also, the following it is stated [HPLWR]: "*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 [HPLWR] 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.

Guidelines and instructions: This challenge is related to design. Paper [NED01] presented reactor pressure vessel (RPV) and internals designed such that the occurring deformations stay inside a prescribed limit for all sealings to be tight.

Requirement 45 related C1_HPLWR (see Table 4.2): FULFILMENT IS PROBABLE
Comment: As written in D5.2 [ECC-D5.2] this is feature of HPLWR. For HPLWR reactor analysis of the core power distribution iterative analysis is required [IAEA-SSG-2]: "*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*."

Guidelines and instructions: As it is explained above, for HPLWR reactor analysis of the core power distribution iterative analysis is required [IAEA-SSG-2]. Similar approach may be used for SCW-SMR.

Requirement 45 related C2_HPLWR (see Table 4.2): FULFILMENT IS PROBABLE
Comment: As written in D5.2 [ECC-D5.2] 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 [GIV01] states that important SCW technology gaps are in the areas of SCWR safety, including power-flow stability during operation. In accordance with [GIV01] an SCWR safety research activity is recommended for power-flow stability assessments.

Guidelines and instructions: Paper [DAUR01] provides general information about Boiling Water Reactor (BWR) stability. The main concerned topics are: phenomenological aspects, experimental database, modelling features and capabilities, numerical models, three-dimensional modelling, BWR system performance during stability, stability monitoring and licensing aspects.

Requirement 45 related C3_HPLWR (see Table 4.2): 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 [ECC-D5.2]). To understand the behaviour of possible Xenon oscillation instabilities a full core model is required [HPLWR]: "*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*."

Guidelines and instructions: To understand the behaviour of possible Xenon oscillation instabilities a full core model, a full-core model applying fast computational methods will be required.

Requirement 45 related C2_BWR (see Table 4.3): FULFILMENT IS PROBABLE
Comment: D5.2 [ECC-D5.2] 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 [DAUR01] 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*."

Guidelines and instructions: The analysis of the phenomena involved in instability requires a multidisciplinary approach [DAUR01] 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 modelling and suitable numerics and specific user guidelines are implemented.

Requirement 47 related I2_WP3 (see Table 4.4): FULFILMENT IS IMPROBABLE
Comment: D5.2 [ECC-D5.2] provides 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 [SCHUL01], 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*".

Guidelines and instructions: The conceptual design is not in the scope of ECC-SMART project. Example of NUWARD in section 4.1.1.1 suggest that pre-conceptual and conceptual design, and preparation for pre-licensing run in parallel.

Requirement 58 related I4_WP2 (see Table 4.4): FULFILMENT IS PROBABLE
Comment: In the report D5.2 [ECC-D5.2] 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 [IAEA-SSG-53].)

Guidelines and instructions: Effect of radiolysis in SCW was studied in [ECC-D2.4].

Requirement 42 related to PIRT WP2 and WP3 phenomena (see Tables 4.6 and 4.7):
Comment: In D5.2 [ECC-D5.3] identified PIRT WP2 and WP3 phenomena are shown in Tables 4.6 and 4.7, respectively.

Table 4.6 *PIRT for WP2*

1	Through wall penetrations produced by general or localized corrosion
2	Oxide build-up that impedes heat transfer
3	Oxide release from the cladding surface
3a	Oxide release by dissolution / evaporation
4	Pellet cladding interaction
5	Environmental Assisted cracking (EAC)
6	Changes in the mechanical properties of the materials produced by ageing and/or irradiation
7	Changes in the geometry of tubes produced by irradiation, creep
8	Radiolysis processes
9	Physicochemical properties of water within the SC region
10	Resistance of cladding materials under LOCA conditions SCWR
11	Impurity enrichment
12	Oxide release from the cladding surface by spalling
13	Irradiation embrittlement due to He
14	IASCC
15	Hydriding
16	Cladding collapse
17	Overheating of the Cladding
18	Overheating of Fuel Pellets
19	Cladding rupture
20	Fuel Rod Mechanical Fracturing
21	Strain Fatigue
22	Fretting Wear

Table 4.7 *PIRT for WP3*

1	Steep non-linear change of SCW fluid material properties
2	Heat transfer in water under supercritical pressure conditions
3	Pressure drop (Δp) in water under supercritical pressure conditions
4	Turbulent heat and mass transfer in water under supercritical pressure conditions
5	Heat and mass transfer along corroded and rough surfaces
6	Deterioration of heat transfer (DHT)
7	Transition from supercritical to subcritical pressure
8	Steam and liquid water two phase flow
9	Natural circulation of water under super - or sub -critical pressure conditions
10	Strong coupling between the thermal hydraulics and the reactor physics
11	Depressurisation of the primary loop and the travelling depressurisation wave
12	The effect of the presence of large and hot structural components
13	Flow instability under supercritical pressure conditions
14	Allowable maximum cladding temperature
15	Flow stratification in horizontal channels
16	Flooding
17	TH and Neutronic instabilities
18	CHF near the critical point
19	Flow induced vibration
20	Mechanical deformation
21	Pellet/cladding interaction

In the following the results of knowledge judgement for most important phenomena are summarized regarding R&D needs. The phenomena from PIRT for WP2 and WP3 are labelled as 'P_WP2_' and P_WP3_', respectively, plus ID number from Tables 4.6 and 4.7 (e.g. P_WP3_21 for pellet/cladding interaction phenomenon no. 21 in Table 4.7).

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) [IAEA-SSG-2] 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 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 were identified (including all five most significant phenomena identified in D5.2 [ECC-D5.2] PIRT analysis, see Table 4.4) 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 [ECC-D5.2] PIRT analysis, see Table 4.4).

4.2 Safety Criteria (JSI)

According to IAEA definition [IAEA-Glossary] the acceptance criteria are specified bounds on the value of a functional indicator or condition indicator used to assess the ability of a structure, system or component to perform its design function[10]. According to [OECD01], the acceptance criteria are directly or indirectly related to the three barriers. Decoupling techniques that cover the range from plant processes to environmental impact are applied to consider the barriers. The fuel safety decoupling criteria for accident conditions are defined to limit fuel damage and ensure radiological releases remain acceptable, provided criteria for the primary circuit and containment are also met. These criteria are based on the mechanical behavior of the systems and include concepts like maximum pressure tolerance, safety factors adjusted for reactor conditions, and design pressure limits.

Document [IAEA-GSR-P4] requires that safety criteria must be identified for safety assessment. Another requirement is that safety criteria for judging safety shall be defined for safety analysis,

[10] **Condition indicator** is characteristic of a structure, system or component that can be observed, measured or trended to infer or directly indicate the current and future ability of the structure, system or component to function within acceptance criteria.
Functional indicator is condition indicator that is a direct indication of the current ability of a structure, system or component to function within acceptance criteria.

sufficient to meet fundamental safety objective and safety principles, as well as requirements of designer, the regulatory body and the operating organisation. Acceptance criteria used to judge the acceptability of the results of safety analysis are described in [IAEA-SRS123]. These may be set numerical limits on the values of predicted parameters; set conditions for plant states during and after an accident; set performance requirements on systems; and set requirements on the need for, and the ability to credit, actions by the operator. Acceptance criteria are most commonly applied to licensing calculations.

4.2.1 Fuel safety criteria (first barrier)

An extensive compilation of fuel safety criteria is provided in the reference [OECD01]. According to the document [OECD01], most of the current fuel safety criteria were established during the 1960s and early 1970s. For the sake of simplicity, the postulated events are divided into anticipated operational occurences (AOOs) and postulated accidents (PAs), which include loss of coolant accident (LOCA) and reactivity initiated accident (RIA). By the mid-1980s, changes in pellet microstructure were observed at higher burn-up levels, accompanied by an increased rate of cladding corrosion and hydriding, which contributed to the degradation of mechanical properties. This highlighted that different phenomena were occurring at high burn-up and/or under evolving operating conditions, making it evident that data extrapolated from low burn-up levels and traditional operating conditions was no longer adequate for reliable predictions. In a 1996 the Committee on the Safety of Nuclear Installations (CSNI) recommended that “fuel damage limits at high burn-up” be recognised as a safety research area to which priority should be assigned. As a consequence, the CSNI and Committee on Nuclear Regulatory Activities (CNRA), decided to undertake an effort involving a much broader (than only high burn-up related issues) look at fuel behaviour and requirements needed to assure appropriate safety margins of modern fuels and core designs. To achieve high discharge exposures and gain thermal margins, more advanced fuel designs were introduced. The pins becomes smaller and the number of rods per assembly has increased in both BWR and PWR applications. The cladding materials for light water reactor (LWR) fuel have also undergone significant evolutions. Review of the fuel safety criteria has been done in 2012 [OECD01], with discussion on possible implications from new design changes on all currently approved fuel (safety) criteria. These criteria are already discussed in D5.3 [ECC-D5.3].

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

Table 4.8 summarizes the results of OECD/NEA fuel safety criteria [OECD01] related with the challenges, issues and phenomena of safety-related behaviour of SCW-SMR (for more details refer to Section 4 of D5.3 [ECC-D5.3]).

In the following comments for each judgement regarding the knowledge, indicating the research needs, are summarized for OECD/NEA fuel safety criteria [OECD01] related with challenges, issues and most significant phenomena of safety-related behaviour of SCW-SMR.

<u>Criterion 1 related to P_WP3_18</u>: 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.

Guidelines and instructions: According to [OECD01] it is unlikely that critical heat flux methodology, the related safety limits, or the methods used to establish these limits, would be subject to significant change. Some testing seems to be needed, including full scale testing to establish the proper thermal-hydraulic modelling of new assembly designs.

Table 4.8 *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 (critical heat flux)	P_WP3_18	FULFILMENT IS PROBABLE
Criterion 2 (reactivity coefficients)	I2_WP4	FULFILMENT IS PROBABLE
Criterion 3 (criticality and shutdown margin)	N.A.	NOT JUDGED
Criterion 4 (fuel enrichment)	I3_SMR	FULFILMENT IS PROBABLE
Criterion 5 (CRUD deposition)	N.A.	NOT JUDGED
Criterion 6 (stress/strain/fatigue)	P_WP2_21	FULFILMENT IS PROBABLE
Criterion 7 (oxidation and hydriding)	P_WP2_15	FULFILMENT IS PROBABLE
Criterion 8 (rod internal gas pressure)	N.A.	NOT JUDGED
Criterion 9 (thermal mechanical loads and PCMI)	WP2_4	NOT JUDGED
Criterion 10 (pellet cladding interaction (PCI)/stress corrosion cracking (SCC))	P3_WP2 (i.e. P_WP2_4)	FULFILMENT IS PROBABLE
	P_WP3_21	FULFILLED
Criterion 11 (fuel melting)	P_WP2_18	FULFILLED
Criterion 12 (linear heat generation rate (LHGR) limits)	I5_WP4	FULFILMENT IS IMPROBABLE
Criterion 13 (RIA cladding failure)	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 14 (fuel fragmentation and fuel dispersal)	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 15 (non-LOCA cladding embrittlement/temperature)	C5_HPLWR	FULFILMENT IS IMPROBABLE
	P_WP3_14	FULFILMENT IS PROBABLE
Criterion 16 (LOCA cladding embrittlement)	P2_WP2 (i.e. P_WP2_10)	FULFILMENT IS PROBABLE
Criterion 17 (Blowdown/seismic/ transportation loads)	N.A.	NOT JUDGED
Criterion 18 (Assembly hold-down force)	N.A.	NOT JUDGED
Criterion 19 (fretting wear)	P_WP2_19	FULFILMENT IS PROBABLE
Criterion 20 (coolant activity)	N.A.	NOT JUDGED
Criterion 21 (fuel gap activity)	N.A.	NOT JUDGED
Criterion 22 (Source term)	I2_SMR	NOT JUDGED
Criterion 23 (burn-up)	C5_HPLWR	FULFILMENT IS IMPROBABLE
	P_WP4-1_7	NOT JUDGED
	P_WP4-1_9	NOT JUDGED

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 [OECD01] 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.

Guidelines and instructions: All temperature reactivity coefficients considered in SCW-SMR core design analyses are negative, but there are differences in the different stages of coolant flow. It is recommended to further optimise the design.

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 [OECD02]. 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 [OECD02]. If enrichment will exceed 8%, design changes may be potentially needed or new design limits should be set.

Guidelines and instructions: Document [IAEA-20] discusses perspective and challenges for LWR fuel enrichment beyond the five per cent limit. Many fuel cycle experts have suggested that the industry aim to license nuclear fuel cycle facilities for operation up to 20% ^{235}U (limit chosen to be in compliance with the non-proliferation treaty).

Criterion 6 related to P_WP2_21: FULFILMENT IS PROBABLE
Comment: 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 medium important phenomenon (16th rank), however the knowledge level is partial (19th rank). As P_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 Fig. 5.1).

Guidelines and instructions: According to [OECD01] mechanical and physical properties used in these fuel performance codes depend on parameters like material composition, fabrication, fluence and hydrogen content. New design changes, particularly those involving high burn-up, may impact the properties. Therefore, ongoing verification and validation of fuel design models are crucial to establish a solid foundation for design and operation.

Criterion 7 related to P_WP2_15: 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. P_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 Fig. 5.1).

Complete or sufficient information is not available for fuel (safety) criterion cladding oxidation and hydriding [OECD01].

Guidelines and instructions: Corrosion of traditional zirconium-based alloys is likely a key factor limiting nuclear fuel lifetime [OECD01]. This highlights the need to reassess the adequacy of current limits on maximum local oxidation and hydriding levels in cladding, particularly considering the performance of high burn-up fuel.

Criterion 10 related to P3_WP2 (i.e. P_WP2_4): FULFILMENT IS PROBABLE
Comment: WP2 phenomenon no. 4 (P_WP2_4, see Tab. 4.11) 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 Fig. 5.1).

Guidelines and instructions: Currently, there is a solid foundation for SCC-PCI limits extending to and beyond 50 GWd/t [OECD01]. Continued ramp testing is recommended to strengthen the basis for higher burn-ups and align with adopted fuel designs, as exemplified by the OECD/NEA SCIP-II Project. Simultaneously, fuel performance codes should be advanced and benchmarked against these ramp tests. With robust modelling, the need for extensive testing can eventually be minimized.

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 [IAEA-SSG-52]. At present, studies are done for pre-conceptual core design and it is expected that design changes will be needed. D5.2 [ECC-D5.2] 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 4.4, I5_WP4. No information has been provided if I5_WP4 issue will be solved in the frame of this project.

Guidelines and instructions: SMRs are often designed for long cycles (up to several years) between refuelling, depending on compact designs and advanced fuels. New design solutions are recommended be considered for future SCW-SMR concept.

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 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 Fig. 5.1).

Guidelines and instructions: According to [OECD01] further experimental investigations are needed for various conditions, including medium burn-up fuel, MOX failure limits, and RIA transients initiated from non-zero power, which have been insufficiently studied or overlooked in the past. The new findings must be incorporated into the RIA criteria.

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 Fig. 5.1).

Complete or sufficient information is not available for fuel (safety) criterion fuel fragmentation, cladding embrittlement [OECD01].

Guidelines and instructions: According to [OECD01] for both LOCA and fuel at high burn-ups, there is a need for further understanding of the fuel dispersal process and the effects of high burn-up.

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 [HPLWR] 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]*[11] *for fresh fuel*." At the time of preparing D5.2 [ECC-D5.2], 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.

Guidelines and instructions: Further analyses are recommended to be considered for future SCW-SMR concept. C5_HPLWR challenge is related to future core design.

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 (1st 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.

Guidelines and instructions: According to [OECD01] an effect of cladding materials would also be expected for this criterion, but the leading mechanism high temperature oxidation kinetic does not depend strongly on the nature of cladding. However, the behaviour of highly burnt fuel under is relatively unknown from temperatures between 600 °C and 1600 °C. The relevance of the non-LOCA cladding embrittlement/temperature criterion should therefore be confirmed experimentally.

Criterion 16 related to P2_WP2 (i.e. 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 2nd highest value. It is high important phenomenon (1st rank), the knowledge level is very limited knowledge (22nd 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 Fig. 5.1).

[11] Reference 'Monti [21]' of [HPLWR] report is: L. Monti, Multi-scale, coupled reactor physics / thermal-hydraulic system and applications to the HPLWR 3 pass core, Dissertation University of Karlsruhe, FZKA 7521, 2009

Guidelines and instructions: It is recommended to follow findings of [NUR-CR01] that embrittlement is sensitive to fabrication processes – especially surface finish. Also, recommendations are given for types of tests that would identify LOCA conditions under which embrittlement would occur.

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 Fig. 5.1).

In general, complete or sufficient information is available for fuel (safety) criterion fretting wear [OECD01]. 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.

Guidelines and instructions: In [OECD01] it is suggested on this aspect to follow [EPRI-01] document on PWR grid-to-rod fretting. This criterion is more design oriented.

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 [OECD01]. 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*."

Guidelines and instructions: Further analyses are recommended to be considered for future SCW-SMR concept. C5_HPLWR challenge is related to future core design due to burn-up. According to document [OECD01] there is a need for further research.

4.2.2 Primary circuit criteria (second barrier)

The safety objective of second barrier is to maintain its capacity to cool down the reactor core and confine the fission products in case of failure of first barrier [OECD03]. The second barrier must maintain its integrity and a sufficient water coolant flow. The main loads which could challenge integrity are thermal load and mechanical loads. For example, for French second barrier safety criteria are established for Category I (transients related to normal operation), Category II (incidents of moderated frequency), Category III (very low frequency accidents) and Category IV (hypothetical accidents) [OECD03]. Variations or additions around those categories have been defined depending of the countries.

An example is shown in Table 4.9.

Table 4.9 *French safety criteria for second barrier concerning pressure loads* [OECD03]

Category	Safety criteria	Decoupling criteria
I-II	System integrity (fatigue damage risk)	Pressure limit 100% design pressure (fatigue damage)
III	System integrity (fast fracture risk)	Pressure limit: 110 % design pressure with all safety valves available 120 % design pressure in case of failure of one safety valve
IV	System integrity (fast fracture risk)	Pressure limit 130 % design pressure

Widely used rules and criteria for design and acceptance criteria of the pressure vessel, primary system components and piping are given in the ASME codes [ASME-BVP], or RCC-M [ASN-RCCM] or RCC-MRx [ASN-RCCMRx] (for details see Section 5.2.2).

For Category I a design pressure has been defined by applying a safety factor to the pressure leading to collapse [OECD04]. The same criterion is applied to Category II, as probability of Category II conditions is quite high. The transients of category III have a lower probability of occurrence therefore some relaxing in the safety factor is allowed by fixing limits slightly higher than the design loads for pressure and mechanical loads. For category IV, the safety criteria are not fully applied to the LOCA case because such event by definition is rupture. For the other general cases, the criteria are defined with a higher percentage of the design loads.

4.2.3 Containment criteria (third barrier)

The phenomena which determine the containment safety for light water reactors are thermo-mechanical loads which could provoke its rupture and the thermo-mechanical loads and irradiation effects which could increase the leakage rates [OECD03].

For the containment rupture the maximum temperature and pressure loads are reached during loss of coolant accident (LOCA), main steam line break (MSLB) inside the containment, and during pressure and temperature peaks resulting from burning the hydrogen. For the containment rupture the phenomena decoupling is used by limiting the load (design pressure) for ruptures related to overpressure loads and by limiting the quantity of hydrogen release for rupture by detonation. The maximum pressure reached during LOCA or MSLB shall be less than design pressure. To avoid effect of hydrogen, a maximum amount of hydrogen shall not exceed 1 % of the hydrogen production by total cladding oxidation (this is one of the ECCS criteria).

For leakages limitations are defined directly on the leak rate values. Maximum values of containment leakage have been defined. For French plants of 1300 MW electric power for all conditions the containment leakage rate of 1.5 % of total mass in the containment per day shall not be exceeded [OECD03].

An example of French safety criteria for the third barrier is given in Table 4.10.

Table 4.10 *French safety criteria for third barrier* [OECD03]

Category	**Safety criteria**	**Decoupling criteria**
all conditions	to preserve design	verification that containment pressure and thermal loads resulting from large break LOCA accident (enveloping case) are respected, to guarantee adequate behaviour of containment system components (also irradiation for containment isolation features)
	containment leakage rate % of total mass in the containment CPY[1]: 0.3 %/day PQY[2]: 1.5 %/day	

[1] Contrat Programme des réacteurs de type 900 MWe

[2] The term "PQY type" when referring to the French 1300 MWe nuclear reactors, specifically denotes a standardized class of four-loop pressurized water reactors (PWRs) designed by Framatome

Table 4.11 *Licensing environment that influences PIRT evaluation criteria*

Level	Source	Criteria				Conser-vatism	Criteria
1*	10CFR1.11	Protect public health & safety				Less limiting	Safety
2*	10CFR100	Limit fuel failure		Limit RCS breach	Limit containment breach		
3*	Appendix A	PCT, oxidation, hydrogen generation, long term cooling, coolable geometry	DNBR (PWR), MCPR (BWR), energy deposition, fuel temperature, cladding strain	RCS & steam systems temperature & pressure	limit containment P & T, leakage & hydrogen. etc.		
4*	SRP 6.2						
4*	SRP 15.1.4 to 15.6.1 Non-LOCA						
4*	10CFR 50.46 & SRP 15.6.5 LOCA						
5*	NUREG/CR-5818	Vessel inventory**					
6*	Interim PIRT	Important parameters					
7*	Interim PIRT	Dominant parameters					
8*	Interim PIRT	Ranked phenomena				More limiting	Behaviour

* Levels 1 – 5 are contained in NRC regulations or regulatory guidance

** Level 5 is key plant response criterion used for AP600 SBLOCA, MSLB and SGTR PIRTs

4.2.4 Example of hierarchical licensing requirements and resulting evaluation criteria

Table 4.11 summarizes the hierarchical licensing requirements and resulting evaluation criteria model. Levels 1-4 reflect the regulatory requirements. Level 5 is key plant response criterion used for AP600 SBLOCA, MSLB and SGTR PIRTs. Levels 6-8 indicate the generic model of the resulting evaluation criteria associated directly with PIRTs. The level 6 criterion is used to rank the relative importance of the phenomenon. The level 7 and 8 information help the level 6 phenomenon in context.

4.3 Methods of Safety Demonstrations

4.3.1 GIF goals and ECC-SMART SMR justification

This section summarizes GIF goals in developing SMR reactor with particular attention on SCW SMR. In GIF communities several goals and targets were underlined in order to design with advanced coolants and technologies. In particular some of the main technical goals focused:

- The development of cladding materials to withstand the high pressure and high temperature environment.
- Design materials and protective coatings capable of withstanding the harsh conditions of supercritical water and radiation, which is crucial for ensuring long reactor life and safe operation.
- The establishment of a chemistry-control strategy to minimize water-radiolysis effect and activation-product transport.
- The optimization of the fuel assembly geometry and configuration to enhance the power output and safety characteristics.
- Ensure passive safety features and robust emergency systems that guarantee reactor safety even during accidents or operational disruptions.
- Be compatible with the local legislation to be built in the selected site.

The SCW-SMR designs, including ECC-SMART (hereafter referred to as “ECC-SMART”), feature specific design elements. Identifying safety elements for ECC-SMART involves two steps:

1. Identification of Existing Safety Elements: This process involves a thorough literature review focused on identifying and evaluating existing safety systems used in both conventional reactors and emerging SMR designs. The review assesses passive and active safety features, considering their proven effectiveness in mitigating risks under normal and emergency conditions. Special emphasis is placed on determining the applicability of these elements to SCW SMRs, which operate under unique conditions such as high pressures and temperatures. The goal is to map out existing safety measures that can be adapted for SCW SMRs, ensuring that validated mechanisms are integrated effectively into these advanced reactor designs.
2. Development of Safety Elements: Where existing safety elements prove insufficient or incompatible with the unique conditions of SCW SMRs, new safety systems must be developed. This involves addressing risks specific to supercritical operation, such as material degradation, coolant behaviour, and system failure modes at extreme temperatures and pressures. New safety systems are designed and tested through simulations and experimental studies, focusing on improving passive safety features and

rapid response capabilities. These developments are rigorously validated under various operational and accident scenarios to ensure they meet regulatory standards and provide enhanced protection for SCW SMRs.

This process is guided by four principles, hereby described:

1. First Guiding Principle: Common Aspects with New Reactors (3rd Gen. NPPs): Safety elements at a general level derive from plant design, performance, and hazard characterization. If two nuclear systems share these aspects, similar top-level safety approaches can be applied. Fluid systems, nuclear steam supply system (NSSS) and balance of plant (BOP) layout and interface Key safety functions and Representative threats (initiating events, accident categories).
2. Second Guiding Principle: In the existing technology several Common Aspects between SMRs can be identified and used for mutual support in developing these technologies. Despite the diversity of SMR designs, they share basic features such as Modularity, with Power less than 300 MWe and Passive safety systems. These features potentially enhance safety by increasing system reliability, reducing hazard magnitude, and lowering dependency on supporting systems or actions. However, SMRs must demonstrate compliance with stringent safety objectives.
3. Third Guiding Principle: Common Aspects with 4th Gen. NPPs: 4th Gen. NPPs aim to address key issues of existing NPPs, leading to innovative designs with specific safety elements. These elements are generic and apply to all designs within the 4th Gen. NPP family.
4. Fourth Guiding Principle: Specific Aspects of ECC-SMART: Key design differences between SCWRs and 3rd Generation reactors include:
    - Higher energy conversion efficiencies
    - Higher operating temperatures and pressures
    - Differences in core design (e.g., two-pass/three-pass configurations)

These differences lead to unique design aspects in both normal and accident conditions:

- **Normal Conditions**: Heat transfer correlations, harsher corrosion environments, changes in water physicochemical properties, neutron physics parameters, and mechanical loads.
- **Accident Conditions**: Refilling/reflooding patterns, higher primary system depressurization and containment pressurization rates, stronger high-pressure melt ejection, direct containment heating, and RPV lower plenum attacks in case of corium relocation at high pressure

In such sense the effort done by the international community in developing an SCW-SMR is collected and partially described in ECC-SMART project deliverables. In particular, this experience is collected from design studies in EU, Canada and China to derive a joint design requirements document following the design targets based on the 4 principles described above. The expected electric power output of the SMR should be around (200 to 300) MW. The specific plant erection costs (€/kW installed electric power) should be less 20 % compared with SMR concepts based on a PWR.

The power plant shall remove the residual heat without the need of electric power at least within a time period of 3 days.

The specific fuel cost (€/MWh electric power) shall be smaller than those of SMR concepts based on a PWR, which may be accomplished by a higher efficiency compensating higher fuel production costs.

ECC-SMART SMR is the bridge between SMR LWRs and GIF IV technologies. All compatible safety elements of new reactors apply to ECC-SMART. Compatibility is defined by design and performance similarities, such as:

Safety elements specific to SMR designs apply to ECC-SMART on a case-by-case basis.

Safety demonstrations must include conservative safety margins due to higher uncertainties and limited operating experience.

All top-level safety requirements identified by the Generation IV Forum (GIF) apply to ECC-SMART, particularly the elimination of DEC-B scenarios.

No new challenging safety requirements specific to ECC-SMART in the accident domain are foreseen. However, quantitatively more demanding phenomena require robust safety demonstrations and conservative safety margins.

4.3.2 Summary of legislation

Up to now, European countries do not have specific legislation prepared to address the unique features of Small Modular Reactor (SMR) technologies. Typically, regulations have been designed for large power units such as EPR and VVER-1000. Consequently, specific legislative documentation and nuclear standards are missing to cover supercritical applications, including:

- High pressure
- High temperature
- Combination with neutron irradiation

The graded approach and the concept of practical elimination should be starting points for the implementation of SMRs in the European nuclear power plant (NPP) portfolio. For example, in the Czech Republic, Decree 329/2017 [CR-D329] covers the graded approach, which must also be conservative.

In Czech legislation, practical elimination is mainly associated with situations or events that can be excluded by physical state or by low probability (based on safety objectives). This duality in Czech legislation, partly due to language barriers, can create confusion for developers and utilities. The concepts expressed by Decree 329/2017 [CR-D329] are legally binding and should be considered in the design description.

Particular attention will be given to probabilistic and deterministic analyses, along with a strong experimental campaign, especially for reactors that can be placed near municipalities. The definition of passive systems is also an open issue in European legislation.

The ECC-SMART project aims to design a Supercritical Water-cooled Small Modular Reactor (SCW-SMR) feasibility concept, which will be developed and preliminarily assessed, highlighting its own issues in navigating the legislative process. The international SCW community faces several challenges:

- **Technical issues**: These involve two-phase flow affecting neutronics, thermal hydraulics, and materials.
- **Legislative issues**: These are characterized by problems in harmonization and in regulating reactors that use passive systems and are smaller units compared to standard NPPs.

However, legislations issued by regulatory bodies are periodically updated consisting of groups of live decrees and safety guidelines. <u>For instance, the Czech Republic is expected to issue a new legislative version in 2025</u>.

In any case, the graded approach and practical elimination concepts play a role in current legislation to ensure the safe operation of research facilities (e.g., research reactors) and future advanced installations. However, these definitions vary slightly from country to country.
Particular attention is also given to the confinement/containment systems, which play a key role in accident mitigation. Several designs claim that evacuation outside the exclusion area is unnecessary. Therefore, the failure of the containment should be practically eliminated in new designs, especially in Generation IV reactor concepts, where resilience should be improved.

Special attention is given to differentiating the criteria for defining radioactive releases. In this regard, the European community lacks sufficient harmonization. Some countries define limits and conditions in sieverts (Sv), while others use becquerels (Bq). This discrepancy can introduce additional challenges for designers when developing confinement systems.

4.3.3 Practical elimination definition

The concept of practical elimination in Czech legislation is quite complex, with some similarities to the IAEA definition. According to Decree 329 [CR-D329], the definition is as follows:

1. D329 §2: "Practically eliminated matter" means a condition, state, or event, the occurrence of which is considered physically impossible, or which is, with a high degree of confidence, very unlikely.
2. D329 §4: It is ensured that the following are practically eliminated:
    a. A radiation accident when there is not sufficient time to implement urgent action to protect the population (referred to as an "early radiation accident").
    b. A radiation accident requiring urgent action to protect the population that cannot be limited in terms of location or time (referred to as a "large radiation accident").
3. D329 §7: The requirement referred to in paragraph (6) shall also be fulfilled if, using a conservative approach, it is demonstrated in the nuclear installation design documentation that the occurrence of a severe accident is a practically eliminated matter.
4. D329 §44 (3): Nuclear installation design shall determine requirements so that a severe accident in the hermetically sealed zone is a practically eliminated event during an operational state with the nuclear reactor shut down and into the confinement.

In the IAEA Glossary [IAEA-Glossary], practical elimination is defined as follows:

The phrase "practically eliminated" is used in requirements for the design of nuclear power plants to convey the notion that the possibility of the potential occurrence of certain hypothetical event sequences in scenarios could be considered excluded ("practically eliminated") provided:

- It would be physically impossible for the relevant event sequences to occur, or
- These sequences could be considered with a high level of confidence to be extremely unlikely to arise.

The term "practically eliminated" can be misleading as it actually concerns the possible exclusion of event sequences from hypothetical scenarios rather than practicalities of safety. It can also be misinterpreted, misrepresented, or mistranslated as referring to the "elimination" of "accidents" by practical measures. Clear drafting in natural language would be preferable.

The two definitions are similar, but the interpretation in Czech legislation is mainly associated with situations or events that can be excluded by physical state or by low probability (based on safety objectives). This duality in Czech legislation, partly due to language barriers, can create confusion for developers and utilities. However, the concepts expressed in Decree 329 [CR-D329] in points §2a, §4, §7, and §44 are legally binding and should be considered in design descriptions.

Additional information on understanding Decree 329 [CR-D329] regarding practical elimination can be found in BN-JB-2.3 rev 0.0 [BN-JB-2-3] paragraph (3.6) and BN-JB-1.5 [BN-JB-1-5] chapter 8. In BN-JB-2.3 [BN-JB-2-3], practical elimination is addressed in paragraph (3.6):

At a minimum, postulated initiating events and scenarios should be selected for assessment if they are not considered achievable in accordance with the recommendations of paragraphs (8.10) and (8.14) of the Safety Guide BN-JB-1.5 [BN-JB-1-5]. The exclusion of these scenarios should be demonstrated with specific frequencies, i.e., those whose contribution to the Large Early Release Frequency (LERF) or Large Late Release Frequency (LLRF) is greater than 10^{-7}/year.

Based on the results of sensitivity studies (see BN-JB-1.5, paragraph 8.14), selected scenarios should also be analysed for contributions to LERF and LLRF greater than 10 % or frequencies of occurrence showing the largest contribution values to the overall uncertainty of the LERF and LLRF outcome. For postulated initiating events and scenarios leading to radiation accidents with confinement bypass, those with a frequency of occurrence less than 10^{-7}/year and greater than 10^{-8}/year should be considered, in accordance with point (8.17) of guidance BN-JB-1.5 [BN-JB-1-5], considering the whole uncertainty analysis.

This guideline directs applicants to apply practical elimination according to scenario frequencies, with cross-references to Chapter 8 of the Safety Guide BN-JB-1.5. Chapter 8 addresses implementing practical elimination using Probabilistic Safety Assessment (PSA) approaches. It sets frequency goals for scenarios and initial events to be considered practically eliminated, with values less than 10^{-7}/year. If uncertainty is greater than 10 %, frequencies (LERF and LLRF) should move from 10^{-7}/year to less than 10^{-8}/year, in agreement with BN-JB-2.3 paragraph 3.6. This procedure ensures compliance with Czech Legislation Decree 329 [CR-D329] §25.

Additionally, BN-JB-1.5 [BN-JB-1-5] Chapter 8 limits the number of scenarios and initial events required for analysis, as they are practically eliminated by low occurrence frequency or negligible radiation consequences (see Safety Objectives). This assessment should be supported by the applicant's PSA approaches, which will assess the event list. Scenarios demonstrated to be practically eliminated due to postulated criteria should be listed in the Final Safety Analysis Report. For example, the dust hazard from a volcanic eruption can be practically eliminated due to very low probability, based on the geographical evaluation of the Czech Republic (far from any active volcano).

In conclusion, practical elimination is used to assess the robustness of the reactors of GEN III+ or above in significantly limiting the source term maintaining the confinement/containment integrity. In particular in advanced SMR technology it is strongly pushed the practical elimination of the catastrophic confinement failure, which leads to uncontrolled release of source term. In such sense any new design including SCW SMR should address this point in particular if different units are placed in the same confinement.

4.3.4 Guidelines for the definition of preliminary safety report

Several tools are defined and continuously updated to help the designer in design the SMRs in order to be ready for the licensing. One of them is the NUREG-0800 [NUR-800], also known as the Standard Review Plan (SRP), which is a critical document prepared by the U.S. Nuclear Regulatory Commission (NRC). It establishes criteria for the review of applications to construct and operate nuclear power plants. This document ensures that all safety aspects are thoroughly evaluated, covering plant design, operation, and safety analysis. For SMRs, NUREG-0800 [NUR-800] provides a structured approach to address unique design features and safety considerations. The main goal is to provide a fixed structures for the designers to guide them in the preparation of a Preliminary Safety Analyses Report. This document also influenced the other legislation in preparation of similar guidelines such as the BN-JB-1.3 [BN-JB-1-3].

Another important tool to evaluate the readiness of the technology is the Technological Readiness Levels (TRLs), which are used to assess the maturity of technologies involved in nuclear facility design. This framework helps determine whether a technology is sufficiently developed to be integrated into a system with minimal risk. For SMRs, particularly the SCW-SMR design under the ECC-SMART project, TRLs guide the development process, ensuring that each component meets the necessary standards before full-scale implementation.

Also, the European Utility Requirements (EUR), which are defined by the utilities in Europe, are important definition of the general concept in the design. The EUR document outlines the expectations of European utilities for new Light Water Reactor (LWR) designs, including SMRs. It covers safety, performance, constructability, and economic aspects of nuclear power plants. The EUR provides a harmonized set of requirements that vendors must meet, ensuring consistency and reliability across different designs. For the ECC-SMART project, adhering to EUR ensures that the SCW-SMR design aligns with European standards and utility expectations. However, this document is normally applicable for large units and, at the present stage, it is not fully focused on the SMR technology.

Additionally, along with the TLRs system, there is the Phases of Design. The design of a nuclear facility typically follows several key phases (see also Section 2.3.1, describing design stages):

- Conceptual Design: Initial ideas and feasibility studies are conducted to outline the basic parameters and goals of the project.
- Basic Design: More detailed plans are developed, including safety analyses and preliminary engineering work.
- Detailed Design: Comprehensive engineering designs are created, specifying all components and systems. Also qualification of systems and equipment is done.
- Construction: The facility is built according to the detailed designs, with ongoing inspections and quality assurance.
- Commissioning: The facility undergoes testing to ensure all systems function correctly before it becomes operational.

For the ECC-SMART project, these phases along with TRL can define the proper development of the SCW-SMR design helping the designers to define its maturity level. Each phase incorporates feedback from technological readiness assessments, compliance with NUREG-0800 [NUR-800], and adherence to EUR requirements.

The ECC-SMART project aims to design a Supercritical Water-cooled Small Modular Reactor (SCW-SMR) feasibility concept. This project addresses both technical and legislative challenges, ensuring that the design meets safety and performance standards while navigating the complex regulatory landscape. The project highlights the importance of a harmonized approach, integrating various tools and frameworks to achieve a robust and reliable SMR design.

By combining these elements, the ECC-SMART project exemplifies the comprehensive approach required to design a modern nuclear facility. This integrated methodology ensures that SMRs, like the SCW-SMR, are safe, efficient, and compliant with both national and international standards.

5 Requirements on Experimental Support for Safety Demonstrations

The experimental support required for the development of new reactors can be subdivided into several categories. The first one is related to the new material development, which is more important for advanced and high temperature technologies. Such experimental program is then applied into the codes and standards, which are very important in the designing process of any nuclear facility. And any authority would not approve any component of the nuclear facility, if it is made from the material which is not in the standards or has in the standards defined application to different purpose than is used. These topics are described in the subchapters 5.1.

The specific area of the experimental programs covers the development of specific technological systems. Because the concept of SCW-SMR does not yet fixed, it has no sense to try to describe needs of the experimental program for not yet existing systems. That is the reason that there is no such subchapter in this report, but this need is pointed out here. What is included is the basis of requirements for different kind of component and systems in the subchapter 5.2.

The additional topic included in the subchapter 5.3 is focused on the analytical (computational) programs and their needs for the improvement in the relation to the SCW-SMR development. Generally, the further improvement of the analytical programs requires experimental support as well as transfer from the material research into the definition of material properties applied in the computer programs.

5.1 Experimental support of material development for SCW-SMR technology

The safety of materials in advanced reactors, such as supercritical water-cooled small modular reactor (SCW-SMR), is critical for ensuring the long-term operation of these systems under extreme conditions. To support safety demonstrations during the conceptual project development, it is essential to provide a solid experimental basis. This basis must be aligned with the identified phenomena that are most likely to challenge material performance in SCWR environments.

The PIRT (Phenomena Identification and Ranking Table; see Deliverable 5.3 and Fig. 5.1) [ECC-D5.3] highlights several key phenomena that have the potential to impact the safety and integrity of SCWR materials. These phenomena must be rigorously evaluated through experimental studies to ensure that the chosen materials can withstand the operating conditions typical of SCWRs, which involve high temperatures, high pressures, and radiative environments.

5.1.1 Key Critical Phenomena and Challenges

During the preparation of ECC-SMART proposal several critical phenomena affecting materials under SCWR conditions were selected. These phenomena were included in the PIRT table. A short explanation of each one is included below:

1. **General and localized corrosion (ID 1 and ID 2) oxide release from the cladding surface (ID 3) by spalling (ID 12):** The combination of high temperatures and pressures, along with dissolved oxygen in supercritical water, presents a significant risk of both general and localized corrosion in key structural components. Corrosion can lead to the gradual degradation of materials and spallation of the oxides, potentially compromising their structural integrity and the operation of the reactor. In addition to this, although there is information about corrosion behavior of materials in simulated SCWR conditions most of them they were obtained after short corrosion tests. One of the challenges addressed in ECC-SMART is to study the

corrosion behavior of cladding materials for long exposures by a deep characterization of the oxides formed on the material´s surfaces.

ID	Phenomenon	Ranking - w/o weights		Ranking - weigths		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

Fig. 5.1 *Phenomena Identification and Ranking Table (PIRT)*

2. **Resistance of cladding material under LOCA conditions SCWR (ID 10):** The phenomena associated with LOCA (Loss of Coolant Accident) involve the loss of coolant in a nuclear reactor, causing a rapid temperature increase in core materials. This leads to oxidation of the fuel cladding and potential release of fission products. Additionally, thermal shock can compromise the structural integrity of reactor materials, leading to cracking and deformation.

3. **Environmental Assisted Cracking (EAC) (ID 5):** EAC is a critical failure mechanism in nuclear environments, where materials are subjected to simultaneous mechanical stress and corrosive conditions. This can lead to the initiation and propagation of cracks, which are particularly concerning in SCWR systems due to their high operating pressures and temperatures. EAC was studied in candidate materials like 800H and 310S and advanced materials like AFA.

4. **Water radiolysis (ID 8) and Physicochemical properties of water within the SC region (ID 9):** In SCWRs, radiolysis of water can generate reactive oxygen species that enhance corrosion rates. Controlling radiolysis is crucial, as it can significantly accelerate material degradation, especially in areas with intense radiation fields. Radiolysis processes in SCWR are

very little known. On the other hand, changes in pressure and temperature within the SC field can affect the corrosion mechanisms, a detailed electrochemical study is necessary to define the operating conditions of the reactor.

5. **Changes in mechanical properties due to aging and irradiation (ID 6):** Materials exposed to high levels of neutron irradiation over extended periods may experience hardening, embrittlement, or changes in ductility, all of which can compromise the safety and reliability of reactor components.

These phenomena guided the experimental approach used in Work Package 2 (WP2) of the project, which focused on materials testing to assess the behavior of selected alloys (A 800H, SS 310S and AFA) under conditions relevant to SCWR operation.

5.1.2 Experimental Activities in WP2

WP2 provided critical experimental support by evaluating the corrosion resistance and mechanical behavior of several candidate materials for SCWR systems. The experimental work was divided into four main subtasks, each addressing different aspects of material performance.

1. **Subtask 1: Procurement, Characterization, and Fabrication of Materials**

Three materials were chosen for comprehensive testing, each offering specific advantages in terms of corrosion resistance and mechanical properties at high temperatures:

- Alloy 800 H: This alloy is widely used in high-temperature applications (e.g. steam generator of PWRs) due to its resistance to corrosion. It has excellent creep properties at elevated temperatures, making it suitable for use in components exposed to the intense heat of SCWRs.
- 310S Stainless Steel: A high-alloy austenitic stainless steel, 310S is known for its high corrosion resistance, particularly in oxidizing environments. It retains strength and durability at elevated temperatures and is often used in environments where other stainless steels would fail.
- Alumina-Forming Austenitic Steel based on 310S (AFA): This advanced material is designed to form a protective alumina (Al_2O_3) layer, which provides superior oxidation resistance compared to chromium oxide-forming steels. The alumina layer significantly reduces the corrosion rate in aggressive environments such as supercritical water.

A800H and steel 310S were purchased in tube shape to study the effect of geometry in their corrosion resistance. One of the objectives of WP2 was to study the materials in conditions as close as possible to real components. The geometries of the tube are shown in Deliverable 2.1; on the other hand, AFA was manufactured as plate by the Chinese partners (USTB).

These materials were subjected to a thorough characterization process, including microstructural analysis using techniques like scanning electron microscopy (SEM), Energy Dispersive X-ray Spectroscopy (EDX) and Electron BackScatter Diffraction (EBSD) as well as mechanical testing characterization.

2. **Subtask 2: Corrosion Behavior Evaluation**

A series of immersion corrosion tests were conducted to evaluate how the selected materials perform under SCWR operational conditions. Tests were carried out at two temperatures, 500 °C and 380 °C, and two pressure levels, 23 MPa and 25 MPa, with an oxygen concentration of 150 parts per billion (ppb). These conditions mimic the high-temperature, high-pressure water environment within an SCWR core.

The materials were exposed to these conditions for varying durations, allowing for the assessment of corrosion rates over time up to 7000 hours in some cases. The formation of oxide layers on the surface was also evaluated to determine the effectiveness of protective oxides in slowing down the corrosion process. Selected samples of studied materials were also tested under simulated LOCA conditions.

Additionally, Slow Strain Rate Testing (SSRT) procedure was employed to assess the susceptibility of the materials to Environmental Assisted Cracking (EAC). SSRT is a sensitive method that simulates the combined effects of stress and corrosion, which is critical for identifying early signs of cracking under reactor operating conditions. The SSRT tests with A800H and steel 310S were performed using specimens cut from longitudinal and circumferential directions of the tubes. AFA specimens were cut from the plate.

3. **Subtask 3: Corrosion Testing of Irradiated Materials**

In this subtask, the corrosion behavior of materials previously irradiated with neutrons up to a dose of 0.3 displacements per atom (dpa) was investigated. Autoclave immersion tests were conducted on irradiated samples to simulate the conditions inside an SCW-SMR, where neutron irradiation can significantly alter the microstructure and chemical properties of the materials.

The results of these tests will provide valuable insights into how irradiation affects the corrosion resistance of the materials. Irradiation-induced changes, such as increased hardening or altered oxide layer formation, were closely monitored.

4. **Subtask 4: Electrochemical Testing and Radiolysis Suppression**

To gain a deeper understanding of the corrosion mechanisms, electrochemical tests were carried out under various temperatures and pressures in supercritical water conditions. These tests helped to clarify how changes in temperature, pressure, and water chemistry influence the electrochemical behavior of the materials, including their susceptibility to localized corrosion and EAC.

Preliminary studies were also conducted to explore the potential suppression of water radiolysis in supercritical water. Suppressing radiolysis could reduce the production of corrosive radicals, thereby extending the lifespan of materials in high-radiation environments.

By connecting the critical phenomena from the PIRT table with the experimental work performed in WP2, we demonstrate how our research directly addresses some of the safety concerns related to material performance in SCWR systems. The experiments carried out on Alloy 800H, 310S Stainless Steel, and the Alumina-Forming Austenitic alloy (AFA) provide a robust dataset for understanding corrosion resistance, EAC susceptibility, and the effects of irradiation, all of which are key to ensuring the safe and reliable operation of SCWRs in the long term.

5.2 Qualification of Individual Systems, Components, and Equipment

This subchapter is focused on the overview of existing Codes and Standards for various areas of applications during the development of the nuclear technology. Specific attention is taken to their applicability for the SCW-SMR with pointing our needs of the scope extensions. The areas of Codes and Standards included are following:

- Design and Construction Rules for Mechanical Components,
- In-service Inspection Rules for Mechanical Components,
- Design and Construction Rules for Containments and Civil Structures,
- Design and Construction Rules for Electrical and I&C Systems and Equipment.

Each of this area has an own sub-chapter.

Table 5.1 *ASME BPVC sections applicable for nuclear facility components*

ASME BPVC Section	Year latest edition
Sec. II Materials - Part A - Ferrous Material Specifications (Vol. 1+2)	2023
Sec. II Materials - Part B - Nonferrous Material Specifications	2023
Sec. II Materials - Part C - Specifications for Welding Rods, Electrodes and Filler Materials	2023
Sec. II Materials - Part D - Properties Customary	2023
Sec. II Materials - Part D - Properties Metric	2023
Sec. III Rules for Construction of Nuclear Facility Components - Subsection NCA - General Requirements for Division 1 and Division 2	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB - Class 1 Components	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NCD - Class 2 and Class 3 Components	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NE - Class Metal Containment (MC) Components	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NF - Supports	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NG - Core Support Structures	2023
Sec. III Rules for Construction of Nuclear Facility Components - Division 2 - Code for Concrete Containments	2023
Sec. III Rules for Construction of Nuclear Facility Components – Division 5 – High Temperature Reactors	2023
Sec. III Rules for Construction of Nuclear Facility Components - Appendices	2023
Sec. V Non-destructive Examination	2023
Sec. XI Rules for Inservice Inspection of Nuclear Reactor Facility Components - Division 1 - Rules for Inspection and Testing of Components of Light-Water-Cooled Plants	2023
Sec. XI Rules for Inservice Inspection of Nuclear Reactor Facility Components - Division 2 - Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities	2023
Sec. IX Welding, Brazing and Fusing Qualifications	2023
Sec. XIII Rules for Overpressure Protection	2023
Code Cases: Nuclear Components (CC-NUC)	2023

5.2.1 Overview of Applicable Codes & Standards

Codes & standards exclusively for the SCW-SMR or SCWR in general, both for its design & construction and operation (i.e. in-service inspection (ISI), maintenance, repair, defect assessment) do not exist. Common nuclear codes standards (NC&S) for mechanical components and structures, such as the ASME Boiler & Pressure Vessel Code (BPVC) Section III, are, in principle applicable to any nuclear reactor following their title ("Rules for Construction of Nuclear Facility Components"). But right at the beginning temperature limits are given in Sec. III, practically limiting the applicability of Sec. III to light-water reactor (LWR) designs. Thus, the task is to assess, to what extent common NC&S are applicable to the SCW-SMR or SCWR in general, and to provide hints on what needs to be added, in case a NC&S is not directly applicable to the SCW-SMR or SCWR.

Table 5.1 lists all ASME BPVC sections with all their parts, divisions and sub-sections, where existent, that are applicable to mechanical components and structures of nuclear reactors. Sections that are not applicable to components of nuclear facilities (e.g. Sec. I – Rules for Construction of Power Boilers) or not in scope of the ECC-SMART project (e.g. Sec. III, Div. 3 - Containment Systems for Transportation and Storage of Spent Nuclear Fuel and High-Level

Radioactive Material) have been omitted from Table 5.1.

Table 5.2 *AFCEN codes*

AFCEN Code	Year latest edition
RCC-M – Design & Construction Rules for Mechanical Components of PWR Nuclear Islands	2022
RSE-M – In-Service Inspection, Installation & Maintenance Rules for Mechanical Components of PWR Nuclear Islands	2022
RCC-MRx - Design & Construction Rules for Mechanical Components of Nuclear Installations: High Temperature, Research & Fusion Reactors	2022
RCC-C - Design and Construction rules for Fuel Assemblies of PWR Nuclear Power Plants	2023
RCC-E – Design & Construction Rules for Electrical and I&C Systems and Equipment	2022
RCC-CW - Rules for Design & Construction of PWR Nuclear Civil Works	2023
RCC-F - Design & Construction Rules for Fire Protection of PWR Nuclear Plants	2024

Table 5.2 lists the AFCEN codes. The AFCEN codes are primarily meant for French PWRs with the exception of RCC-MRx. The latter was initially introduced in support of the sodium fast reactor (SFR) development in France, to provide rules on the design and manufacturing of mechanical components of SFRs. The scope of RCC-MRx has been enlarged to other types of reactors and there are continuous efforts and initiatives underway (e.g. CEN Workshop 64) to specifically cover mechanical components of lead-cooled fast reactors (LFRs) and molten salt reactors (MSRs). Also here the task is to assess, to what extent AFCEN codes are applicable to the SCW-SMR or SCWR.

In the following sections the applicability of the (N)C&S in Tables 5.1 and 5.2 are discussed to an extent that the author of this section has access and is knowledgeable about the NC&S. The discussion will be brief and not detailed in the sense of making a comprehensive assessment of each section of the individual NC&S. The author opted to perform the assessment per main category of structure, system and component (SSC) in the following order: (1) design & construction rules for mechanical components, (2) ISI rules for mechanical components, (3) containment & civil structures, and (4) design & construction rules for electrical and I&C systems and equipment.

5.2.2 Design & Construction Rules for Mechanical Components

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Subsection NCA - General Requirements for Division 1 and Division 2 contains, as its name already indicates, general requirements for construction of mechanical components and their support structures (all Sec. III Div.1 Subsections) and of concrete containments (Sec. III Div.2). These involve:

- General provisions (NCA-1100) and general requirements for items and their installation (NCA-1200);
- General requirements for the classification of components & supports (NCA-2000);
- Responsibilities & duties of subcontractors (suppliers, metallic material suppliers, welders) in terms of certification and material quality system or programme (NCA-3000);
- Quality assurance (QA) requirements (NCA-4000), including for suppliers and contractors;
- Authorised inspections for QA (NCA-5000);
- Referenced standards (NCA-7000) of ASME, ASTM, AWS, ACI, ANSI for specific component design (i.e. dimensional standards, e.g. for flanges, forged fittings, …) and testing of materials (metallic materials and concrete);
- Requirements for certificates, name plates, certification marks and data reports (NCA-8000);
- Glossary, i.e. general and re-occurring terms in all ASME BPVC sections (NCA-9000).

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB - Class 1 Components contains rules on design and manufacturing of class 1 mechanical components. Specifically these involve:

- Introductory remarks including temperature limits for applicability (NB-1000);
- Rules on the choice of structural materials of class 1 components, heat treatments, mandated tests to demonstrate their mechanical properties (sufficient level of ductility), non-destructive examination (NDE) of forged class 1 components (NB-2000);
- Rules on the design of class 1 components, including criteria & special considerations, vessel design, pump design, valve design, pipe design (NB-3000);
- Rules for fabrication & installation of class 1 components (NB-4000);
- Rules for NDE of the fabricated class 1 component and in view of pre-service inspection (NB-5000);
- Rules for pressure testing of class 1 components (NB-6000);
- Rules on overpressure protection of class 1 components (NB-7000)

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NCD - Class 2 and Class 3 Components contains rules on design and manufacturing of class 2 and 3 mechanical components. The document structure is identical the one of Subsection NB and specifically these involve:

- Introductory remarks including temperature limits for applicability (NCD-1000);
- Rules on the choice of structural materials of class 2&3 components, heat treatments, mandated tests to demonstrate their mechanical properties (sufficient level of ductility), non-destructive examination (NDE) of forged class 2&3 components (NCD-2000);
- Rules on the design of class 2&3 components, including criteria & special considerations, vessel design, pump design, valve design, pipe design (NCD-3000);
- Rules for fabrication & installation of class 2&3 components (NCD-4000);
- Rules for NDE of the fabricated class 2&3 component and in view of pre-service inspection (NCD-5000);
- Rules for pressure testing of class 2&3 components (NCD-6000);
- Rules on overpressure protection of class 2&3 components (NCD-7000)

Given that a SCW-SMR or SCWR has one primary loop similar to a BWR and is overall similar than a BWR in terms of design, the same safety classification of SSCs as for a typical LWR can be applied:

- SC1: Any SSC whose failure would lead to consequences of high severity,

- SC2: Any SSC whose failure would lead to consequences of medium severity,
- SC3: Any SSC whose failure would lead to consequences of low severity,

Subsections NB and NCD provide requirements for strength and pressure integrity of class 1 and class 2&3 components respectively, whose failure would violate the pressure-retaining boundary. The rules in subsections NB and NCD only cover initial construction requirements. They do not cover deterioration that occurs in service as a result of corrosion, radiation effects or instability of material. *NB-1120 Temperature limits* and *NCA-1120 Temperature limits* provide limits on the applicability of Subsections NB and NCA for class 1 and class 2&3 components respectively. Subsections NB and NCA shall not be used for class 1 and class 2&3 components made of ferrous materials subject to temperatures exceeding 375 °C, because above that temperature creep and stress rupture characteristics become significant factors for those kind of materials, which are not accounted for in these subsections. *NB-1120 Temperature limits* and *NCA-1120 Temperature limits* also set limits on the use of fatigue design curves and fatigue analysis methods in Sec. III – Appendices in the sense that these are not applicable for class 1 and class 2&3 components made of ferrous materials and austenitic stainless steels subject to temperatures exceeding 370 °C and 425 °C respectively.

So to summarise ASME BPVC Sec. III, Subsections NB and NCD are only applicable to those class 1 and class 2&3 components made of ferrous materials and austenitic stainless steels of the SCW-SMR or SCWR that are subject to temperatures not exceeding 370 °C and 425 °C respectively. As pointed out above these temperatures creep and stress rupture characteristics become significant factors and designing reactor components for use at elevated temperatures beyond the two temperature limits above is well covered by ***ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 5 – High Temperature Reactors***. Originally introduced for high-temperature gas-cooled reactors (HTGRs) substantial parts of Sec. III, Div. 5 address graphite material and graphite structures, which are not relevant for the SCW-SMR and SCWR. However, Sec. III, Div.5 also addresses metallic components of high temperature reactors, providing design rules against creep and stress rupture, which are indeed relevant for the SCW-SMR and SCWR, those components that operate beyond the two temperatures above. Sections and parts of Sec. III, Div.5 relevant for the SCW-SMR and SCWR in general are as follows:

- General requirements – metallic materials, including classification of components and supports (Subsection HA, Subpart A);
- Rules on material, design, fabrication & installation, NDE, pressure testing, overpressure protection, mandatory and non-mandatory appendices of Class A Metallic Pressure Boundary Components, both for low-temperature service and mainly elevated temperature service (Subsection HB);
- Rules on material, design, fabrication & installation, NDE, pressure testing, overpressure protection, mandatory and non-mandatory appendices of Class B Metallic Pressure Boundary Components, both for low-temperature service and mainly elevated temperature service (Subsection HC);
- Rules on material, design, fabrication & installation, NDE and mandatory appendices of Class A Metallic Core Support Structures, both for low-temperature service and mainly elevated temperature service (Subsection HG).

Unlike Sec. III, Div.1 with classification of components to class 1, 2 and 3, Sec. III, Div. 5 distinguishes between Class A and Class B components. According to Sec. III, Div. 5, Article HAA-1100 General Class A and Class B components can be regarded as equivalent to Class 1 and Class 2 components respectively in terms of consequences of their failure.

In Sec. III, Div.5, subsections HB, HC and HG distinction is made between service at low-temperatures and elevated temperatures. However, the articles on low-temperature service are very short, essentially just referring to their equivalents in Sec. III, Div.1, meaning Subsections NB, NCD, NF and NG. Thus, Sec. III, Div.5, Subsections HB, HC and HG are almost entirely on elevated temperature service.

Two of the remaining subsections of Sec. III, Div.1, meaning ***Subsection NE - Class Metal Containment (MC) Components*** and ***Subsection NF – Supports*** are only applicable to the SCW-SMR and SCWR, for metallic liners and support structures subject to temperatures where creep and stress rupture is negligible. The text structure of the two subsections is as follows.

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NE - Class Metal Containment (MC) Components contains rules on design and manufacturing of metallic containment (MC) vessels. Specifically these involve:

- Introductory remarks including temperature limits for applicability (NE-1000);
- Rules on the choice of structural materials of MC vessels, heat treatments, mandated tests to demonstrate their mechanical properties, NDE of MC vessels (NE-2000);
- Rules on the design of MC vessels, in particular stress analysis (NE-3000);
- Rules for the fabrication & installation of MC vessels (NE-4000);
- Rules for NDE of fabricated MC vessels (NE-5000);
- Rules for pressure testing of MC vessels (NE-6000);
- Rules on overpressure protection of MC vessels (NE-7000).

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NF – Supports contains rules on design and manufacturing of support structures of class 1, 2 & 3 components and MC vessels. Specifically these involve:

- Introductory remarks (NF-1000);
- Rules on the choice of structural materials, heat treatments, mandated tests to demonstrate their mechanical properties, NDE of support structures (NF-2000);
- Rules on the design of support structures, in particular stress analysis (NF-3000);
- Rules for the fabrication & installation of support structures (NF-4000);
- Rules for NDE of fabricated support structures (NF-5000);
- Mandatory appendices on materials and welding of support structures (NF-I), design of single angle members (NF-II) and energy absorbing support material (NF-III);
- Non-mandatory appendices on structural bolt preloading (NF-A), design allowable stresses for plate and shell and linear-type support (NF-B & -C), tolerances (NF-D) and dampers, energy absorbers & snubbers (NF-E).

The last subsection of Sec. III, Div.1, ***ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NG - Core Support Structures,*** contains rules on design and manufacturing of core support structures. Specifically these involve:

- Introductory remarks (NG-1000);
- Rules on the choice of structural materials, heat treatments, mandated tests to demonstrate their mechanical properties, NDE of core support structures (NG-2000);
- Rules on the design of support structures, in particular welded joints (NG-3000);
- Rules for the fabrication & installation of core support structures (NG-4000);
- Rules for NDE of fabricated core support structures (NG-5000).

In Div.5, Subsection HG is said that the rules of Div.1, Subsection NG apply except where indicated otherwise. This means that Sec. III, Div.1, Subsection NG is to a large extent applicable to the SCW-SMR and SCWR in general.

As an overall conclusion the parts on metallic components of the ***ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 5 – High Temperature Reactors*** are all applicable for the SCW-SMR and SCWR. Given that Sec. III, Div.5 makes to a large extent reference to Sec. III, Div.1, Subsections NB, NCA, NE, NF and NG, they are all applicable to the SCW-SMR and SCWR, except where stated otherwise in Sec. III, Div.5.

Sec. III, Div.1 and Div.5 make reference to the parts of Sec. II Materials, which contains specifications (i.e. chemical composition, mechanical properties) for all materials that are used for components in nuclear reactor, pressure or heating boilers. Assuming that common reactor materials (e.g. SA 533 for RPV, 316L for reactor internals) are to be used in the SCW-SMR and SCWR, Sec. II is applicable to the SCW-SMR and SCWR.

Fuel cladding materials are not in scope of ASME BPVC Sec. II, but subject to specific standards (e.g. ASTM B811 for zirconium-alloy fuel cladding for LWRs). Thus for potential fuel cladding materials of the SCW-SMR or SCWR like 800H and 310S that were in scope of WP2 of the ECC-SMART project, specific standards comparable to ASTM B811 need to be developed. However, 800H is included in ASME BPVC Sec. II (under the designation UNS N08810), so is a commonly used material, where as 310S is not included in ASME BPVC Sec. II.

The AFCEN codes, meaning ***RCC-M – Design & Construction Rules for Mechanical Components of PWR Nuclear Islands*** and ***RCC-MRx - Design & Construction Rules for Mechanical Components of Nuclear Installations: High Temperature, Research & Fusion Reactors***, are similar to ASME BPVC Sec. III, Div.1 and Div.5 respectively. Thus, all the above statements and conclusions on the applicability of ASME BPVC Sec. III, Div.1 and Div.5 to the SCW-SMR and SCWR apply to RCC-M and RCC-MRx.

5.2.3 In-Service Inspection Rules for Mechanical Components

ASME BPVC Sec. XI Rules for Inservice Inspection of Nuclear Reactor Facility Components - Division 1 - Rules for Inspection and Testing of Components of Light-Water-Cooled Plants contains rules for in-service inspection and testing of LWRs. It also contains procedures on assessment of defects, fatigue, fracture, etc. and risk-informed in-service inspection (RI-ISI). Specifically these involve:

- General requirements (scope & responsibilities, examination & inspection, standards for examination evaluation, repair / replacement, system pressure tests) (Subsection IWA)
- Requirements for Class 1 components (scope & responsibilities, examination & inspection, acceptance standards, system pressure tests) (Subsection IWB)
- Requirements for Class 2 components (scope & responsibilities, examination & inspection, acceptance standards, system pressure tests) (Subsection IWC)
- Requirements for Class 3 components (scope & responsibilities, examination & inspection, acceptance standards, system pressure tests) (Subsection IWD)
- Requirements for MC and metallic liners of containments (scope & responsibilities, examination & inspection, acceptance standards, system pressure tests) (Subsection IWE)
- Requirements for class 1, 2, 3 and MC and supports (scope & responsibilities, examination & inspection, acceptance standards) (Subsection IWF)

- Requirements for concrete components (scope & responsibilities, examination & inspection, acceptance standards, system pressure tests) (Subsection IWL)
- Ultrasonic Examinations (requirements, coverage, supplements, records & reporting, ultrasonic examinations of vessel & piping welds) (Mandatory Appendices I, II & III)
- Eddy Current Examinations (scope, general system & personnel requirements, qualification requirements, essential variable tolerances, records) (Mandatory Appendix IV, including supplements)
- Qualification of personnel for visual examination (scope, qualification levels, written practice, qualification requirements) (Mandatory Appendix VI, including supplements)
- Qualification of NDE personnel for ultrasonic examinations (scope, qualification levels, written practice, qualification requirements, qualification records) (Mandatory Appendix VII, including supplements)
- Performance demonstration for ultrasonic examination systems (scope, general examination system requirements, qualification requirements, essential variable tolerances, records) (Mandatory Appendix VIII, including supplements)
- Analytical evaluation of flaws (nonmandatory Appendix A)
- Analytical evaluation of flaws in piping (nonmandatory Appendix C)
- Conditioning of welds that require ultrasonic examination (nonmandatory Appendix D)
- Analytical evaluation of unanticipated operating events (nonmandatory Appendix E)
- Fracture toughness criteria for protection against failure (nonmandatory Appendix G)
- Analytical evaluation procedures for flaws in piping based on the use of a failure assessment diagram (nonmandatory Appendix H)
- Guide to plant maintenance activities and Section XI repair / replacement activities (nonmandatory Appendix J)
- Assessment of reactor vessels with low upper shelf Charpy impact energy levels (nonmandatory Appendix K)
- Operating plant fatigue assessment (nonmandatory Appendix L)
- Applying mathematical modelling to ultrasonic examination of pressure-retaining components (nonmandatory Appendix M)
- Written Practice Development for Qualification and Certification of NDE Personnel (nonmandatory Appendix N)
- Analytical Evaluation of Flaws in PWR Reactor Vessel Head Penetration Nozzles (nonmandatory Appendix O)
- Weld Overlay Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping Weldments (nonmandatory Appendix Q)
- Risk-Informed Inspection Requirements for Piping (nonmandatory Appendix R)
- Evaluating Coverage for Section XI Non-destructive Examination (nonmandatory Appendix S)
- Reporting of Contracted Repair/Replacement Activities (nonmandatory Appendix T)
- Analytical Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Piping and Class 2 or 3 Vessels and Tanks (nonmandatory Appendix U)
- Mechanical Clamping Devices for Class 2 and 3 Piping Pressure Boundary (nonmandatory Appendix W)

Subsections IWA – IWL and mandatory Appendices I – VIII are all on the in-service inspection (ISI), mainly NDE, and inspection qualification of mechanical components of LWRs. Considering that the SCW-SMR and SCWR in general are water-cooled reactors, even a specific form of water, Subsections IWA – IWL and mandatory Appendices I – VIII are fully applicable. When in cold shutdown the SCW-SMR and SCWR should not be different to a conventional BWR or PWR. This means that ISI / NDE technologies for BWRs and PWRs (which are normally inspected in cold shutdown) can be widely used for the SCW-SMR and SCWR (with some adaptions here and there to account for different in-vessel structures) and that Subsections IWA – IWL and mandatory Appendices I- VIII are applicable to the SCW-SMR and SCWR.

Non-mandatory Appendices A, C, E, G, H, K, L, O, U are procedures for flaw evaluation, fatigue assessment or assessment against failure in general of class 1, 2, 3 components of BWRs and PWRs. As these procedures involve to some extent temperature material properties, are only meant for certain temperature ranges (excluding creep regimes) or are explicitly limited to certain LWRs materials or components, their applicability to the SCW-SMR or SCWR needs to be assessed on a case-by-case basis. In any case they provide guidance and hints how to evaluate flaws in pressure-boundary components of the SCW-SMR or SCWR.

The non-mandatory appendices D, J, M, N, Q, R, S, T are related to maintenance, inspections, risk-informed ISI or reporting of repair / replacement of components and are applicable to the SCW-SMR and SCWR.

ASME BPVC Sec. XI Rules for Inservice Inspection of Nuclear Reactor Facility Components - Division 2 - Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Reactor Facilities was just introduced a couple of years ago to address ISI of advanced reactors, mainly Gen IV reactor systems (non-water-cooled reactors). In contrast to Div.1, which is a set of codified rules on how to inspect LWRs and qualify NDT systems and to evaluate flaws, fatigue, failure of LWR components, Div.2 is rather a methodology / framework to demonstrate and manage the reliability and integrity of reactor components of the entire lifetime of a reactor. Div.2 assumes the availability of component reliability numbers (i.e. probability of failure), which is a challenge for reactor design for which no operational experience exists yet. A potential output of the methodology could be to redesign a component to improve its predicted reliability.

Div.2 is established as a methodology to demonstrate and manage the reliability and integrity of reactor components because for Gen IV reactor systems inspection of certain components is not easily possible. Components are not easily accessible (e.g. in-vessel components of lead-cooled fast reactors (LFRs), molten salt reactors (MSRs)) and there is lack of suitable NDT technology that can cope with the environmental conditions (i.e. high temperatures, non-transparent coolant).

Div.2 contains dedicated chapters on specific Gen IV reactor systems, meaning sodium fast reactor (SFR), LFR, high-temperature gas-cooled reactor (HTGR) and MSR. Given that Div.2 is still rather new certain chapters are not populated yet, including the ones on SFR, LFR and MSR. An own chapter dedicated to the SCWR is not included in Div.2.

Given that the SCW-SMR and SCWR in general is a water-cooled reactor, even if a specific form of water, the author recommends to use Sec. XI, Div.1 when it comes to inspection and inspection qualification of the SCW-SMR and SCWR in general. Non-mandatory appendices of Sec. XI, Div.1 on flaw evaluation, fatigue assessment and assessment of components against failure should be applied with caution to the SCW-SMR and SCWR, since the higher operating temperature may limit the validity of these evaluation and assessment procedures. Most likely they need to be adapted to the SCW-SMR and SCWR.

Table 5.3 *ENIQ guidance documents*

	European Methodology for Qualification of Non-Destructive Testing – Issue 4
RP2	Strategy and Recommended Contents for Technical Justifications – Issue 3
RP4	Recommended Contents for the Qualification Dossier – Issue 2
RP5	Guidelines for the Design of Test Pieces and Conduct of Test Piece Trials – Issue 3
RP6	The Use of Modelling in Inspection Qualification – Issue 3
RP7	Recommended General Requirements for a Body Operating Qualification of Non-Destructive Tests – Issue 2
RP8	Qualification Levels and Approaches (currently under revision) – Issue 3 (in 2025)
RP9	Verification and Validation of Structural Reliability Models and Associated Software to be used in Risk-Informed In-Service Inspection Programmes – Issue 2
RP10	Personnel Qualification – Issue 2
RP11	Guidance on Expert Panels in Risk-Informed In-Service Inspection – Issue 2
RP12	Strategy and Recommended Contents for Inspection Procedures
RP13	Qualification of Non-Destructive Testing Systems that Make Use of Machine Learning
	ENIQ Framework Document for Risk-Informed In-Service Inspection – Issue 2

RSE-M – In-Service Inspection, Installation & Maintenance Rules for Mechanical Components of PWR Nuclear Islands by AFCEN contains rules for ISI, installation, maintenance and defect and fatigue assessment of French PWRs. Its scope is the same as ASME BPVC Sec. XI, Div.1, so RSE-M can be used as an alternative for inspection and inspection qualification of the SCW-SMR and SCWR. Also for the RSE-M its flaw evaluation and fatigue assessment procedures should be applied with caution to the SCW-SMR and SCWR. When applying RSE-M to the SCW-SMR and SCWR it should be remembered that RSE-M was exclusively introduced for French PWRs. Appendix III of RSE-M listing the qualified inspections for pressure boundary components of French PWRs (class 1 components), is much tailored to the design and geometries of these components. ASME BPVC Sec. XI, Div.1 is applicable to BWRs and PWRs.

All the guidance documents by the European Network for Inspection and Qualification (ENIQ) in the areas of inspection qualification and RI-ISI are fully applicable to the SCW-SMR and SCWR when it comes to their inspection. ENIQ guidance documents are universal and not limited to any specific reactor design. The table 5.3 lists all current Issues of ENIQ guidance documents (RP = Recommended Practice).

5.2.4 Design and Construction Rules for Containments and Civil Structures

ASME BPVC Sec. III Rules for Construction of Nuclear Facility Components - Division 2 - Code for Concrete Containments contains rules on design, construction and testing of concrete containments. Specifically these involve:

- Introductory remarks (CC-1000);
- Rules and requirements on materials, both metallic (e.g. liners, tendons, anchors) and concrete (i.e. ingredients, properties), mandated tests to demonstrate mechanical and chemical properties, NDE of finished metallic components (mainly liner) (CC-2000);

- Rules on the design of containments, including design loads and stress analysis (CC-3000);
- Rules for fabrication of containment structure, including producing concrete, manufacturing & installation of reinforcements & pre-stressing system, tendon installation, tensioning, manufacturing of liner (CC-4000);
- Rules for concrete examination, NDE of fabricated metallic parts and leak tight tests of liner (CC-5000);
- Rules for structural integrity testing of concrete containments (CC-6000);
- Mandatory appendix with tables on pre-stressing and liner material (D2-I), glossary of terms (D2-II), approval of new materials (D2-III), qualifications of concrete inspection personnel (D2-V), leak testing by vacuum box technique (D2-VI), qualifications for arc welding reinforcement bars (D2-VIII);
- Non-mandatory appendices on load combination (D2-A), preheat procedures (D2-B), certification of Level 1 and 2 concrete inspection personnel (D2-C), liner dimensional tolerances (D2-D), certified material test reports for liner materials (D2-E), reinforcement fabrication and placing tolerances (D2-F).

Given that ASME BPVC Sec. III, Div. 2 is applicable to containments of LWRs, so both PWRs and BWRs, and that the SCW-SMR or SCWR most likely will have a BWR-like containment (at least European HPLWR is supposed to have BWR-like containment), it can be assumed that ASME BPVC Sec. III, Div. 2 is overall applicable to the SCW-SMR. The same most likely also applies to the AFCEN counterpart of ASME BPVC Sec. III, Div. 2, RCC-CW. This is an assumption by the author of this section, as he is not familiar with RCC-CW and does not have access to it.

5.2.5 Design & Construction Rules for Electrical and I&C Systems and Equipment

The assessment on the applicability of existing NC&S for electrical and I&C systems and equipment for the SCW-SMR and SCWR in general focuses on AFCEN's ***RCC-E – Design and Construction Rules for Electrical and I&C Systems and Equipment*** introduced for French PWRs, since the author of this sub-section is familiar with them and has access to them. RCC-E contains the following volumes:

Volume I - General and Quality Assurance contains general provisions and requirements on QA and QM.

Volume II - Specification of needs lists and describes briefly all the factors that need to be considered to demonstrate and ensure the safety of the system or equipment in scope. The factors are related to the item (technical characteristics), environment (environmental conditions in which item is operating, e.g. temperature, humidity, radiation), loads & hazards, maintenance & periodic testing to ensure functionality or grid interference.

Volume III - Automation and Control Systems contains rules on the design of I&C systems and maintaining their safety functions over their whole lifetimes. Volume III also covers verification and modification of software of digital I&C systems and simplified procedures for the qualification of digital devices of limited functionality (DDLFs).

Volume IV - Electrical System contains rules on the design of electrical power supply systems and equipment.

Volume V - Equipment Engineering contains rules on the qualification of electrical and I&C equipment, including cables, for harsh environments (i.e. radiation, high temperature, …), transients, hazards (e.g. vibrational loads, seismic events, fire) and severe accidents.

Volume VI - Layout of electrical and instrumentation and control systems contains rules on the layout of electrical and I&C equipment, meaning connections and laying of cables, physical and electrical separation and installation.

Volume VII - Inspection and test methods contains rules for inspections and tests for electrical and I&C equipment, including acceptance criteria.

Appendices:

List of standards (Z.1000)
Definitions and abbreviations (Z.2000)
Documentation (Z.3000)
Probationary phase rule – Nuclear management system (Z.4000)
Equipment specification guidance (Z.5000)
Detailed summary (Z.6000)

RCC-E refers to a large extent to standards of the International Electrotechnical Commission (IEC standards), in particular in Volume III, like e.g. IEC 61513, IEC 60880, ... Although introduced for electrical and I&C systems and equipment for French PWRs, RCC-E does not (unlike C&S for mechanical components) include any applicability limits in terms of reactor operating temperatures. Thus, RCC-E is fully applicable to the SCW-SMR and SCWR in general.

5.3 Future Development of Analytical Programs

This chapter addresses the challenges in simulating Supercritical Water facility (SCW) in existing computer codes, particularly for the ECC-SMART Small Modular Reactor (SMR) [ECC-SM1]. The Research Units have encountered several limitations during the preliminary concept development. The following categories are analyzed:

- **Finite Element Method (FEM) Codes**: such as ANSYS [ANS1 site], ABAQUS [ABA site]
- **Neutronics Codes**: such as SERPENT [SER site], DYN3D [DYN ref]
- **Probabilistic Safety Assessment (PSA) Codes**: such as CAFTA, PHOENIX [EPRI site]
- **Severe Accident Codes**: such as ASTEC [AST site]
- **Thermohydraulic Codes**:
    - System Codes: such as RELAP 5 [ISS site], ATHLET [GRS site]
    - Subchannel Codes: such as COBRA-TF [NCSU site], SubChanFlow [KIT site]
    - Computational Fluid Dynamics (CFD) Codes: such as ANSYS FLUENT [ANS2 site], STAR-CCM+ [SIME site]
- **Thermomechanical Fuel Codes**: such as TRANSURANUS [TRAN site]

The codes reported here are examples of the vast code portfolio existing worldwide.

Finite Element Method (FEM) Codes

FEM codes require materials and models that account for the specific conditions of the supercritical environment under gamma and neutron irradiation. Extensive experiments are necessary to establish data (e.g., lookup tables) and models to improve the predictability of FEM codes. Special attention is given to the resilience and mechanical characteristics of the Reactor Pressure Vessel (RPV) and internals, which operate at temperatures around (380-700) °C and 25 MPa [ECC-D2.1].

Probabilistic Safety Assessment (PSA) Codes

PSA codes provide live PSA models and Risk Monitors to predict state changes in a Nuclear Power Plant (NPP) and evaluate if they exceed operational safety limits. These codes are compatible with any NPP, including advanced technologies like ECC-SMART SMR. The main issues are the lack of data from the current stage of design and the engineering expertise needed to evaluate component and system reliability. This knowledge is crucial for modeling fault and event trees and preparing a risk monitor model.

Severe Accident Codes

Severe accident codes typically lack models and water material properties to simulate SCWR behavior under severe accident conditions. SCW may require specific phenomenology for the failure of advanced cladding materials under development. For example, the Generic Oxidation Model (GOX) in MELCOR 2.2 [MEL_RM] requires specific data on how cladding material oxidizes during severe accident progression.

Thermohydraulic Codes

Thermohydraulic codes are essential for reactor design and performance assessment under accidental conditions [IAEA1693], [IAEA1869]. SCW and advanced coolants lack heat transfer and turbulence correlations for proper fluid regime simulation. For ECC-SMART SMR, correlations predicting horizontal flow regimes (DHT and IHT) are undeveloped for horizontal flow and corrosion effects, significantly influencing heat transfer and conductivity [AMB01]. Insufficient number of experimental campaigns studying general flow regimes and specific effects in subchannels contribute to this issue. Transition from turbulent to laminar flow can generate inconsistent data in predicting bulk and heat structure temperatures increasing the code instability.

Neutronics Codes

Neutronics codes can simulate SCW as it does not differ significantly from water [ECC-D4.1]. Accurate data, particularly during water/steam interface, is crucial for setting moderation conditions, reconstructing flux shapes, and determining parameters like criticality and burnup effect. This issue is also common in Boiling Water Reactor (BWR) technology. Spiral flow should not pose a problem from a neutronics perspective. Procedures are consistent with Monte Carlo and deterministic methods, such as first collision solvers and nodal codes. Nodal codes require cross-sections calculated by lattice codes (deterministic or Monte Carlo).

Thermomechanical Fuel Codes

Thermomechanical fuel codes [TRAN Man], like FEM codes, depend on material analyses to be coded directly into the source code. Extensive experimental campaigns are required, where cladding and fuel materials are stressed under various conditions, including oxidation environments, power ramps simulating rod ejection, and performance in normal and accident conditions. Such extensive experiments are typically unavailable for advanced technologies like ECC-SMART SMR, requiring significant effort in capacity, funding, and time.

6 Specific Features of SCW-SMR Licensing

Generally, the legislation in EU countries is covering mostly LWR licensing (PWR or BWR or both). There is no common legislation in EU, because this area is in the responsibility of each of member state. None of EU countries has already prepared legislation for the advanced reactors, like SWCR design or any other of Gen IV. That is a reason that it does not make a sense to include contributions from all of the partners describing specific features of the SCW-SMR licensing in their countries. Based on these findings, it was decided that only four examples of the recent legislative basis or their preparatory initiatives are included in this document – Czech Republic, Slovenia, Spain and Hungary brief descriptions of their national limitations for the SCWR licensing. On other hand, there are countries in the world, which already operates nuclear reactors of which technologies are included in the Gen-IV technologies like:

- Russian Federation – BN-600 resp. BN-800 reactors are Sodium Fast Reactors in operation since 1980 resp. 2014, and BN-1200 under preparation,
- China – two HTR-PM small size reactors are producing power since the end of 2021 and in commercial operations since 2023,

but none of them SCWR technology. That implies that the legislation basis in these two countries are prepared for already applied technologies, but not for the specific SCW technology.

Significant development activities for GenIV reactors are also taking place in the USA, but so far they are only at the beginning of construction/application preparations. But USA is also very active concerning the preparation for the licensing of various technologies of GenIV reactors, specifically the proposal of the modified 10 C.F.R. Part 53 (prepared by the U.S. NRC) is now in the first phase of commenting, and a specific sub-chapter on this activity is also included.

6.1 National limitations for GenIV/SCWR licensing

This chapter contains a description of the situations in the Czech Republic, Slovenia, Spain and Hungary concerning the legislation basis for the licensing of the GenIV technologies.

6.1.1 Czech Republic

As introduced in Chapter 4 and presented in [ECC-D5.1], the Czech Republic still faces several limitations regarding the installation of SMR technology due to its legislative system [CRNACT], [CR-D329]. These limitations could become particularly challenging for GenIV technology [GIV07], which significantly differs from standard PWR reactors.

Czech legislation is oriented towards VVER technology, which is currently installed in the country [CEZ-ETE], [CEZ-EDU]. The guidelines, limits, conditions, safety objectives, and other important features for the installation of different technologies are typically customized based on the experience of existing power plants. Consequently, they are not fully applicable to “conventional” nuclear technologies such as Boiling Water Reactors (BWR). Another important limitation concerns the definition of SMR [CRNACT], which is not fully recognized in Czech legislation. In fact, the VVER 440 can be considered a sort of predecessor to SMR technology due to its modularity and the sharing of some systems between two reactors. In this sense, the “graded approach” [CR-D162] helps the licensing process to be customizable depending on the technology presented for evaluation by the licensing commission of SUJB.

Additionally, it is expected that legislation will change in 2025, introducing concepts such as SMR and becoming more open to different technologies than VVERs. The Czech Republic is a candidate to be a pioneer in SMR licensing in Europe, seriously considering the installation of an SMR fleet [CR-New1], [CR-SMR].

6.1.2 Slovenia

Slovenia legislation covers mainly light water reactors. Small modular reactors are not mentioned in the legislation. In the following some brief overview is given. At the top is act ZVISJV-1 [ZVISJV-1], which shall transpose into Slovenian legislation Council Directives. Chapter 4 of ZVISJV-1 [ZVISJV-1] is on radiation and nuclear safety. It classifies the facilities, ensures radiation and nuclear safety (including the articles on prohibitions and ensuring the safety of a facility), land use, construction projects or mining works (including articles on design bases for a radiation or nuclear facility, safety report, physical security plan), trial operation of radiation and nuclear facilities, operation of radiation and nuclear facilities, management of radioactive waste and spent fuel, emergency preparedness. Important rules are JV5 [JV5], Rules on radiation and nuclear safety factors and JV9 [JV9], Rules on the safety assurance of radiation and nuclear facilities. In the view of newbuild, both rules have been recently upgraded. This upgrade has mainly been done by implementing specific IAEA standards requirements and WENRA requirements.

In 2023 the JV9 rules [JV9] were upgraded. New sections according to requirements of IAEA standard SSR-2/2 (Rev. 1) [IAEA-SSR-2/2], e.g. management and control over facility systems, special requirements for reactor core and fuel assemblies in nuclear power plants (NPP) and research reactors (RR) were added. Existing section of emergency preparedness was aligned with the Decree on the content and elaboration of protection and rescue plans. Extension of requirements for daily, quarterly and annual reporting of NPP and annual reporting of RR has been done, and new annex on annual reporting of facility for management of radwaste. Finally, requirements were amended for aging management and obsolescence according to new WENRA requirements.

In 2024 the JV5 rules [JV5] have been upgraded. Design requirements were amended with more detailed requirements from IAEA standards SSR-2/1 (Rev. 1) [IAEA-SSR-2/1], SSR-2/2 (Rev. 1) [IAEA-SSR-2/2], SSR-3 [IAEA-SSR-3], GSR Part 4 (Rev. 1) [IAEA-GSR-P4], GSR Part 5 [IAEA-GSR-P5] and GSR Part 6 [IAEA-GSR-P6]. Cyber security requirements were added, too. Management system was amended according to requirements of GSR Part 2 [IAEA-GSR-P2] and WENRA safety reference levels (SRL) issue C. Design requirements for internal and external hazards caused by human activities were according to WENRA SRL issues SV and TU [WENRA03]. Design bases were amended for nuclear power plants, irradiation facilities and accelerators. New annexes were dedicated to preoperational testing (commissioning) according to SSR-2/2 (Rev. 1) [IAEA-SSR-2/2] and SSR-3 [IAEA-SSR-3], procedure for approval of program for preoperational testing in the course of issuing operating permit. Design requirements for human factor and human-machine interface were also added. JV5 rules [JV5] now provide design bases, issuing of consents and permits, safety documentation, and management system and leadership focused on safety. In annexes design basis for nuclear power plants, design basis for research reactors, design basis for low- and intermediate-level radioactive waste storage, design basis for spent fuel or high-level radioactive waste storage, design basis for radioactive waste disposal facilities, design basis for mining or hydrometallurgical tailings disposal, design basis for irradiation facilities, cybersecurity and pre-operational testing program are specified.

Finally, it was recognized by Slovenian Nuclear Safety Administration [SNSA-SMR] that the IAEA safety standards are based on good practices drawn from the experience of Member States. Much of the experience comes from large water-cooled reactors that are dedicated to electricity generation. New designs may use different types of coolant, nuclear fuel, neutron spectra and inherent safety. The IAEA considered whether the current requirements and recommendations are applicable to new technologies (see SRS-123 [IAEA-SRS123]). It was confirmed that the safety standards are applicable to new designs with some exceptions.

Spain

In Spain, the licensing of Generation IV reactors and Small Modular Reactors (SMRs) remains in the preliminary stages, with no projects currently under construction or licensed. The regulatory body responsible for overseeing and approving such installations is the *Consejo de Seguridad Nuclear* (CSN).

Although no formal projects have been launched, institutions such as CIEMAT and several universities are actively conducting research or participating in European funded projects on Generation IV reactors and SMRs.

Additionally, there are ongoing discussions and studies about the feasibility of SMRs in the Spanish context. Notably, proposals have been made to deploy SMRs in regions like the Canary Islands, where their modularity and enhanced safety features could support electricity generation and water desalination.

The Spanish nuclear industry, through initiatives like the Small Modular Reactors Working Group (SMR) of CEIDEN, has also developed a roadmap to explore the potential of these technologies. This roadmap aims to foster collaboration between institutions and companies, paving the way for future developments in advanced nuclear technologies.

Although still in an early phase, these efforts highlight Spain’s interest in aligning with global trends in nuclear innovation and exploring the potential benefits of SMRs and advanced reactor designs to enhance the country’s energy security and sustainability.

6.1.3 Hungary

At this moment, there is no specific legislation existing in Hungary for SMR reactors. Although the high-level legislation (Atomic Energy Act, CXVI. Act of 1996) [HU01] is quite general to be applied for all nuclear facilities, the legal requirements described in Nuclear Safety Codes (NSC) published by the Hungarian Atomic Energy Authority (HAEA) (1/2022. HAEA Decree on nuclear safety requirements for nuclear facilities and on related regulatory activities) [HU02] are specific for given nuclear applications. Design requirements for new nuclear power plants are listed in Volume 3a of NSC, however, these requirements have been formulated for large LWR units. Further volumes of NSC have to be applied for requirements concerning other fields (such as management systems, siting requirements, etc.).

The present licensing system for new nuclear power plants has been developed for the Paks 2 project, which aims the construction of two large PWR units at the Paks site. The licensing system is linear, the different main licenses (environmental licence, site licence, construction licence, commissioning / operation licence) can be granted after each other, which is a suitable approach for large NPP units, but may take unnecessarily long time for smaller units.

The modification of Hungarian legislation in order to facilitate SMR licensing has not officially started yet, however the review seems to be inevitable as there are already some companies interested in the technology, although no official was made yet.

6.2 Example of non-EU approach

There is a new approach to licensing of the advanced reactors under development by U.S. NRC. Regardless it is under development, the first proposal was released [NRC01] and could be very inspirative for the approach of advanced reactor licensing in Europe.

Following short description of the proposed modifications are cited from [NRC02].

Since the U.S. adoption of commercial nuclear power, large-scale commercial reactors — typically boiling or pressurized water designs — have been licensed under the provisions of 10 C.F.R. Part 50 (Part 50) and later 10 C.F.R. Part 52 (Part 52). Under these licensing approaches, license applicants submit construction and operating license permits in a multiyear process largely tailored to address the risks and controls necessary to operate bespoke, large-scale nuclear power plants at a specific site.

The push for a new licensing framework tailored to advanced reactors came in earnest in 2019 after Congress passed the Nuclear Energy Innovation and Modernization Act (NEIMA), which required NRC to establish a technology-inclusive framework for commercial advanced nuclear reactor applicants. NEIMA, as modified by the Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024 (ADVANCE Act), defines "advance nuclear reactor" as "a nuclear fission reactor or fusion machine … with significant improvements compared to commercial nuclear reactors ...." The Nuclear Regulatory Commission (NRC or Commission) considered using this term and definition to establish the scope of Part 53 but determined that "significant improvements" was not defined with enough specificity. Instead, NRC uses the broader term "commercial nuclear plant" to be technology inclusive.

Now a new generation of technology is emerging. This includes small modular reactors (SMRs) and microreactors, which are poised to be cheaper, more readily deployable, faster to construct, and simpler to operate. NRC is proposing Part 53 to address Congress' demand for a licensing process that will accommodate these new design developments through an efficient licensing framework tailored to the new way in which nuclear power will be deployed without compromising the level of safety ensured under Parts 50 and 52 today.

Following the comment period, which closes on December 30, 2024, NRC staff expects to provide the draft final rule to the Commission in 2025 and issue the final rule no later than the end of 2027.

Here are five key takeaways from the proposed rule.

1. ***Part 53 Establishes Licensing Framework That Is Technology-Inclusive, Risk-Informed, and Performance-Based***

The Part 53 framework is broken into subparts that address the various stages of a commercial nuclear plant's lifecycle. The licensing framework employs a probabilistic risk assessment (PRA)–led approach that builds on the Department of Energy (DOE) Licensing Modernization Project methodology. A PRA is a mathematical technique for determining what sequence of failures would be required to cause a release of radioactive material and calculating the probability that all the component failures required for that sequence to occur would actually happen.

Under Part 53, a PRA must be performed for each nuclear commercial plant to identify potential failures, susceptibility to internal and external hazards, and other contributing factors to event sequences that might challenge safety functions. This departs from the Part 50 and 52 processes, which prescribe the safety framework new reactors must be evaluated against.

The proposed approach is meant to provide flexibility for PRAs to be developed and assessed based on the application they are used to support. NRC is seeking advice on what additional guidance may be needed for PRA acceptability. Additionally, NRC is seeking recommendations on the proposed organization of Part 53 requirements and how certain provisions can be consolidated or reorganized to make the rule clearer and more concise.

2. Part 53 Includes Eight Different Types of License Applications

Under the proposed rule, there would be eight license applications.

- *Early Site Permit: In the initial stage, an applicant may seek an early site permit for approval before seeking a construction permit or combined license.*
- *Limited Work Authorization (LWA): An applicant may also request an LWA in conjunction with an early site permit to perform specified activities.*
- *Standard Design Approval (SDA): An SDA provides an option for receiving approval of a final standard design for a nuclear power reactor which can be referenced in future CP, OL, COL, or ML applications.*
- *Standard Design Certification (SDC): An SDC provides approval of a standard design for a nuclear power facility through a rulemaking.*
- *Construction Permit (CP): A CP allows the license holder to construct a commercial nuclear plant. Under the proposed Part 53, a CP would be issued prior to an OL and would be converted into an OL upon completion of the facility and Commission action.*
- *Operating License (OL): An OL allows the license holder to operate a commercial nuclear plant. A COL combines the CP and OL and provides all necessary conditions.*
- *Manufacturing License (ML): An ML authorizes the manufacture of nuclear reactors. Regarding MLs, NRC has set out multiple provisions for which it requests advice and recommendations. Two of these requests are for (1) comments on the sufficiency of the proposed regulations to govern various scenarios for manufacturing and deployment of manufactured reactors and (2) comments on whether Part 53 should allow a CL or OL applicant or holder to reference an ML.*

3. Part 53 Supports Efficiencies for Multiple Plants of a Common Design

NRC proposes to allow the combination of applications for multiple sites using an identical design ("common design"). This would apply to CP, OL, and COL applications with a common design, allowing one or more applicants to seek common review of a license to construct and operate nuclear power reactors located at multiple sites.

NRC is seeking comments on whether more flexibility can be added under this provision. In particular, NRC requested comments on whether this provision should consider applications that are not completely identical and, if so, what process would be used for determining whether common review would be appropriate.

4. Part 53 Modifies Standard Part 50/52 Technical Requirements in Light of New Technologies

a. Changes to Parts 26 and 73

Part 53 is supplemented by provisions in Parts 26 and 73 to enhance the safety of facilities. Under Part 26 provisions, Part 53 licensees will be required to implement Fitness for Duty Programs (e.g., drug and alcohol testing and fatigue management) no later than the start of construction to ensure that personnel at facilities are fit for duty, trustworthy, and reliable.

Part 73 will support Part 53 with alternative physical protection requirements. For certain applicants who do not meet the existing requirements under Parts 50 and 52, 10 C.F.R. § 73.100 will provide an alternative physical protection program aimed at providing assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.

b. Comprehensive Risk Metrics

Part 53 proposes the use of comprehensive risk metrics and associated risk performance as one of several performance standards. This includes the use of individual early fatality risk, the individual latent cancer fatality risk, and the quantitative health objectives from the NRC Safety Goals Policy Statement. NRC is seeking recommendations on what other performance standards could be used to address the comprehensive risk posed by proposed commercial nuclear plants.

c. Defense in Depth

Proposed 10 C.F.R. § 53.250 would establish requirements to assess and provide defense in depth to address uncertainties of commercial nuclear plants during licensing-basis events. NRC is requesting comments on the inclusion of such requirements and whether specific provisions should be added to more explicitly address the possible role of inherent characteristics of some structures, systems, and components in preventing or mitigating unplanned events.

5. Part 53 Includes Waste Management Requirements

Part 53 establishes and details requirements for waste management throughout the lifecycle of a commercial nuclear plant. For operational requirements, § 53.850 would require every holder of an OL or COL to maintain a Radiation Protection Program and a program for the control of radioactive effluents. The program for radioactive effluents must be contained in an Offsite Dose Calculations Manual. OL and COL holders must also have a Process Control Program for solid radioactive waste processing, process parameters, and surveillance requirements.

Part 53 is poised to provide the industry with significant design and operational flexibilities to alleviate what is often viewed as a key impediment to a new “nuclear renaissance” — the inability to obtain the regulatory approvals necessary to design and build a nuclear reactor in a cost-efficient manner. While NRC expects the rulemaking won’t be final until 2027 and licensing a reactor under Part 53 comes after that, this proposed rulemaking is a significant, long-awaited step in bringing new advanced reactor technologies to market.

7 Conclusions

Work Package 5 (WP5 - Synthesis and Guidelines for Safety Standards) focuses the development of generic and specific safety criteria and requirements for the SCW-SMR concept based on the available guidelines and regulatory documentation developed for Generation IV reactors. The WP5 also synthesizes main safety-related findings and conclusions of the research work of the other technical work packages WP2, WP3, WP4) and their main achievements described in corresponding deliverables. The final task of WP5 is to develop a pre-licensing study demonstrating the feasibility of the SCW-SMR concept and to develop guidelines for the demonstration of safety for the further design phases of the concept. The participants of WP5 are: JSI, CVR, JRC, CIEMAT, ENEN, BME, IPP, VTT, KIT. The WP leader is JSI.

For the aforementioned goals, four task groups have been set up into sub-work packages:

WP5.1 Generic and specific safety criteria and requirements,
WP5.2 Safety-related findings and conclusions of the WPs 2-4,
WP5.3 Pre-licensing study,
WP5.4 Guidelines for the demonstration of safety in the further development stages.

This report is the output from the activities performed in WP5.4 focused on the preparation of guidelines for the safety demonstrations in the development stages of the SCW-SMR design.

A special characteristic of SCW-SMR is the mixed design – it can be considered as a light water-cooled reactor (practically similar to Gen III BWR technology), while being part of the set of typical Gen IV designs offering high core outlet temperatures. On the other hand, the SMR size results in special features – usually more favourable safety parameters, but less advantageous neutronics. The principal difference from the LWR technology is the coolant parameters, which represent the supercritical water conditions, i.e. pressure above 25 MPa and core exit temperature in the range (450 to 500) °C. From this point of view, the design of the SCW-SMR has many similar features to LWR, but the principal difference is in the design of the core, reactor pressure vessel and primary circuit, because of much stronger requirements for materials. The difference also applies to the safety assessment because some type of analyses requires specific knowledge, experimental data and analytical tools, validated to these specific conditions.

If we look at the possible accident scenarios, then accidents initiated by transients or loss of flow (LOFA) will be completely different, since in these cases the primary circuit will start at nominal pressure parameters. On the other hand, the response of the primary circuit in a LOCA type of accident will be very similar in principle, with the exception of a short initial interval when the pressure drops from supercritical values above 25 MPa to values typical for LWR reactors. However, the difference will be for the response of the containment, when the energy of the coolant that escapes into the containment, especially in the case of a large LOCA, will be higher. This specific feature must be taken into account by the designer when considering the size of the free volume of the containment. An appropriate size will ensure that there is no over-pressurization and thus a threat to the integrity of the containment during the initial phase of the coolant blowdown of a LOCA accident. The safety analyses of the containment response themselves are identical and do not require any specific experimental or analytical program. Concluding this idea, the specific programs for SCW-SMR must cover the phenomena related to the behavior of the core, primary circuit and materials used for these structures.

Guideline for the safety demonstration in the various phases of a process of new reactor technology development consists of following main steps.

1. Identification of phase of the design development (pre-conceptual design, conceptual design, basic design, detailed design), that implies what scope of the safety demonstration has to be done
    a. It produces also updated design description (data required for safety analysis)
2. Definition/update of safety requirements of the design under development
    a. To be in accordance with legislative requirements
3. Definition/update of safety criteria related to the appropriate safety requirements of the design under development
    a. To be in accordance with legislative requirements
4. State of the art – that implies need for further experimental support and analytical tool development and validation
5. Performing an appropriate supporting experimental program to produce new knowledge or demonstrate the viability of tested systems or equipment, or to qualify individual equipment or systems
6. Performing an appropriate analytical program to demonstrate fulfilling safety requirements via. confirmation of fulfillment of defined safety criteria

It has to be emphasized that the completion of the individual phases of the design process requires several iterative loops through all five steps of this guideline because the identification of any need for design modification which can come from the experimental or analytical safety verification has an impact on the design description and any change in the design requires review of the requirements and criteria definition.

Detailed descriptions of the basis and requirements for the performing of individual steps are with the aim of the SCW-SMR development process described in the individual chapters of this document. Here is the identification of relations among steps and chapters of this document:

Step 1 – identification of design phases and scope of the safety demonstration is described in Chapter 2.

Step 1.a – description of the current version of the SCW-SMR design/concept is described in Chapter 3, not in the format of the data for safety analyses, but here in the version to demonstrate a phase of designing process.

Step 2 – definition or update (in the later phases of the design development) of safety requirements is described in Chapter 4, and it is focused on specific requirements for SCWR technology.

Step 3 – definition or update of safety criteria is very closely tied to the previous step because each of the safety requirements must have defined its own safety criteria. Guiding for the criteria definition is also included in Chapter 4 linked to the requirement definition.

Step 4 – identification of the status of knowledge concerning the further experimental needs, needs of the updating of the codes and standards, and also further development of the analytical tools is included in Chapter 5, this chapter contains also recommendations for future development of analytical computer programs.

Steps 5 and 6 are related to the application of the demonstration via experimental or analytical program, their definitions are always defined depending on the topics, so the general guidelines cannot describe all of them.

Step 5 – the experimental program must start with the identification of the state-of-the-art, followed by a definition of the goals, experimental program matrix, designing of the experimental facility, performing the program and results processing including identification of results uncertainties.

Step 6 – the analytical program has to start with a review of the state-of-the-art analytical tools (computer programs in the terminology of [IAEA-SSG2]) to identify if a suitable analytical tool already exists or a new one has to be developed. If the program already exists the next step is the program implementation, testing and independent validation by the user. These steps are crucial because their results prove the user's qualification to use the program and to produce trustworthy results. A negligible part of any analytical work is also an evaluation of the results' uncertainties. The second case related to the new program development is a little more complicated because the program developer must provide code verification and validation which covers all phenomenological areas of the code application. The code application follows the same steps as in the case of the application of an already existing program – the only exception is in the program implementation and validation if the user is also the developer. Then these activities do not need to be repeated.

Demonstrating safety is a complex process that applies to all phases of reactor development. As designs mature, the scope of requirements for safety demonstration and verification increases. This document outlines these requirements in a basic and generic manner since the lack of a conceptual design prevents the creation of design-specific safety demonstration guidelines. As previously mentioned, this process is iterative; therefore, the program developed for one phase of design development will become more extensive and demanding in the next phase, reflecting the expanding scope of safety assessment.

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