ECC-SMART PROJECT

Joint European Canadian Chinese development of Small Modular Reactor Technology

Grant Agreement Number: 945234
H2020 – NFRP-2019-2020

Start Date of the Project: 1/09/2020
Duration: 54 Months

Deliverable Title:	**D5.2 Safety related features of the SCW-SMR concept**
Lead party:	BME
Author(s):	Ildikó Boros (BME), Attila Kiss (BME)
Participant(s):	BME: Csenge Antók, Boglárka Babcsány, Szabolcs Czifrus, Balázs Kiss, Tamás Varju; CIEMAT: Alberto Sáez Maderuelo; CNL: Chukwudi Azih, David Guzonas, Kittima Khumsa-Ang, Armando Nava-Dominguez, Hussam Zahlan; CVR: Monika Sipova; IPP: Yaroslav Dubyk, Vladislav Filonov, Yuliia Filonova; JRC: Radek Novotny, Oliver Martin; JSI: Andrej Prošek; KTH: Haipeng Li; RATEN: Manuela Fulger; UoP: Walter Ambrosini, Valerio Giusti; VTT: Aki Toivonen.
Due Date:	31/12/2023

Version Number: 3.0

Approved by: Leon Cizelj (JSI, WP5 leader)

Date: 02 February 2024

Dissemination level		
PU	Public	**X**
CO	Confidential: only for ECC-SMART Partners (including the Commission Services)	
EU-RES	Classified Information: RESTRAINT UE*	
EU-CON	Classified Information: CONFIDENTIEL UE*	
EU-SEC	Classified Information: SECRET UE*	

**Commission Decision 2005/444/EC*

Document History – Version Control

Version	Issue Date	Stage	Description of Changes	Contributor
0.0	14.12.2023	Draft Version	First integrated version	BME
1.0	10.01.2024	Draft Version	First revised version	BME
2.0	22.01.2024	Draft Version	Second revised version	BME
3.0	02.02.2024	Final Version	Third revised version	BME

Project information

Project acronym:	ECC-SMART
Project title:	Joint European Canadian Chinese development of Small Modular Reactor Technology
Grant Agreement n°	945234
Start date:	1.9.2020
Duration	54 months (Original 48 months, extended by 6 months)
Programme:	Framework Programme for Research and Innovation (2019-2020) under the call NFRP-2019-2020
Coordinator:	Centrum Výzkumu Řež, s.r.o. (Czech Republic)

Document information

Work package:	WP5 - Synthesis and Guidelines for Safety Standards
Contractual Date of Delivery:	31.12.2023
Actual Date of Delivery to the EC:	02.02.2024
Document type:	Report
Dissemination level:	Public (PU)
Number of pages:	73

Abstract:

This deliverable outlines the safety related features of supercritical water-cooled small modular reactor (SCW-SMR) design based on the review of the available literature, the conclusions available now from technical work packages and the expert opinion of WP participants.

The literature review covers the special safety features of supercritical water cooled reactors, small modular reactors and – because the similarity of designs – boiling water reactors. For the evaluation of WP2-4 results the deliverables and other reports of the working packages have been elaborated. The expert opinions about material-related, thermal hydraulics-related and neutronics-related safety features have been collected in a Phenomena Identification and Ranking Table (PIRT) analysis.

The aim of this deliverable is to provide answers to the potentially outstanding issues and gaps in knowledge regarding the safety related behaviour of supercritical water reactor. The outcome of this task are in a synthetic form summarized conclusions and lessons learned to facilitate the future work of designers and nuclear safety analysts.

The ECC-SMART project has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945234.

Executive Summary

In the framework of work package 5 (WP5) Task 5.2 the safety related features of Supercritical Water-cooled Small Modular Reactor (SCW-SMR) have been collected from the available results of work package (WP) 2, 3 and 4 and from a review of technology-related literature.

The related literature covers the safety documentation of supercritical water-cooled reactors (SCWRs) by the Generation IV International Forum (GIF)[5][6] and the design and analysis documentation of the High Performance Light Water Reactor (HPLWR) [10]. For special safety issues concerning Small Modular Reactors (SMRs), International Atomic Energy Agency (IAEA) documentation [12] and report by Canadian Nuclear Laboratories (CNL) [13] have been reviewed. The safety systems of Boiling Water Reactor (BWR) designs, especially Gen 3+ reactors have also been reviewed [11].

The deliverables and periodic reports of WPs 2, 3 and 4 have been reviewed to evaluate WP conclusions. The collection of a priori knowledge of the participating experts and the preliminary results has been started. For this qualitative analysis, the Phenomena Identification and Ranking Table (PIRT) method has been selected, which is a well-known systematic analysis tool for expert opinions and has been widely applied for advanced reactor designs in different development phases. The PIRT methodology consists of two main steps. In the first step significant safety-related phenomena are collected based on the previous knowledge of participants and on the recent results of WPs. In the second step, the participating experts rank these phenomena based on their safety importance and our level of knowledge related to the investigated phenomena. This method ensures a comprehensive overview of important phenomena and the identification of our knowledge gaps requiring further analysis.

The report summarizes the conclusions and lessons learned into a synthetic form to facilitate the future work of designers and nuclear safety analysts.

Table of contents

List of acronyms and abbreviations

AC	Alternating Current
ADS	Automatic Depressurization System
AFA	Alumina Forming Alloy
BME	Budapesti Műszaki és Gazdaságtudományi Egyetem (Budapest University of Technology and Economics)
BoC	Beginning of Cycle
BWR	Boiling Water Reactor
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (Centre for Energy, Environmental and Technological Research)
CIV	Containment Isolation Valve
CNL	Canadian Nuclear Laboratories Ltd.
CVR	Centrum výzkumu Řež s.r.o. (Research Centre Řež s.r.o.)
DBC	Design Basis Condition
DEC	Design Extension Condition
DHT	Deterioration of Heat Transfer
DiD	Defence in Depth
DPA	Deterministic and Phenomenological Analyses
EAC	Environmental Assisted Cracking
ECC-SMART	Joint European Canadian Chinese development of Small Modular Reactor Technology
ECCS	Emergency Core Cooling System
ELSMOR	Towards European Licencing of Small Modular Reactors
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
ENEN	European Nuclear Education Network
EoC	End of Cycle
EUR	European Utility Requirements
FA	Fuel Assembly
FoM	Figure-of-merit
FQT	Fuel Qualification Test
GDCS	Gravity-driven Emergency Injection System
GIF	Generation IV International Forum
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HPLWR	High-Performance Light Water Reactor
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IC	Isolation Condenser
IL	Importance Level
IPP	ТОВ ІПП-Центр (IPP-Centre LLC)
ISAM	Integrated Safety Assessment Methodology
IVR	In-vessel Retention
JRC	Europese Commissie - Joint Research Centre (Petten)
JSFR	Japan Sodium Fast Reactor
JSI	Institut "Jožef Stefan" (Jožef Stefan Institute)
KIT	Karlsruher Institut Für Technologie (Karlsruhe Institute of Technology)
KTH	Kungliga Tekniska Högskolan (Royal Institute of Technology)
KL	Knowledge Level
KR	Knowledge Ranking

LB-LOCA	Large Break Loss of Coolant Accident
LEI	Lithuanian Energy Institute
LW-SMR	Light water small modular reactors
LWR	Light water reactor
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MSIV	Main Steam Isolation Valve
OPT	Objective Provision Tree
PCV	Primary Containment Vessel
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
POLIMI	Polytechnic University of Milan (Politecnico di Milano)
PR	Phenomena Ranking
PSA	Probabilistic Safety Assessment
QSR	Qualitative Safety Features Review
RATEN	Regia Autonoma Tehnologii pentru Energia Nucleara (Institute for Nuclear Research Pitesti)
RCIC	Reactor Core Isolation Cooling System
RD	Relative Deviation
REA	Rod Ejection Accident
RPV	Reactor Pressure Vessel
RR	Relative Relevance
RSWG	Risk and Safety Working Group
SASS	Self-actuation shutdown system
SB-LOCA	Small Break Loss of Coolant Accident
SBO	Station Blackout
SCW	Supercritical Water
SCWR	Supercritical Water-Cooled Reactor
SCW-SMR	Supercritical Water-Cooled Small Modular Reactor
SEM	Scanning Electron Microscopy
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SMR	Small Modular Reactor
TEM	Transmission Electron Microscopy
TLFW	Total Loss of Feedwater
UoP	Università di Pisa (University of Pisa)
VTT	Teknologian tutkimuskeskus (Technical Research Centre of Finland)
WNA	World Nuclear Association
WP	Work Package

List of tables

List of figures

List of Annexes

Annex 4 – PIRT analysis results, WP4

Related documents

- Deliverable D2.1 Test matrix based on available materials, 04/07/2022, CIEMAT (leader),
- Deliverable D2.2 Report summarising basic characterization of materials and specimens machining, 30/7/2021, CVR (leader),
- Deliverable D3.1 Report summarizing the newly generated reference data for natural convection, forced and mixed convection and decay heat removal, 30/09/2023, KIT (leader),
- Deliverable D3.2 Report on the results of the benchmark exercise, 23/05/2023, NPIC (leader),
- Deliverable D4.1 Neutron physics code selection results, 15/11/2021, CNL (leader),
- Deliverable D5.1 Safety criteria and requirements for the SCW-SMR concept, JSI (leader), 31/08/2021,
- Work Package Periodic report M18 – Reports for WP2, WP3, WP4 and WP5,
- Initial List of Requirements for a Supercritical Water-Cooled Small Modular Reactor, CNL, 217-120200-REPT-002, 2022.

1 Introduction

This chapter introduces the project itself and its goals as well as the work done in the frame of WP5 Task 5.2 so far.

1.1 Project summary

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) concept.
The main objectives of the project are to define the design requirements for the future SCW-SMR technology, to develop the pre-licensing study and guidelines for the demonstration of the safety in the further development stages of the SCW-SMR concept including the methodologies and tools to be used and to identify the key obstacles for the future SMR licencing and propose strategy for this process. To reach these objectives, specific technical knowledge gaps were defined and assessed to achieve the future smooth licensing and implementation of the SCW-SMR technology (especially behaviour of materials in the SCW environment and irradiation, validation of the codes and design of the reactor core will be developed, evaluated by simulations and experimentally validated).

1.1.1 Project goals

The Work Package 5 (WP5 - Synthesis and Guidelines for Safety Standards) is aiming 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 4 reactors. The WP5 also synthesizes the main safety related findings and conclusions of the research work of the other work packages[1] (WP2, WP3, WP4), mainly based on the deliverables of the WPs. 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:

- WP5 Task 5.1 Generic and specific safety criteria and requirements,
- WP5 Task 5.2 Safety-related findings and conclusions of the WPs 2-4,
- WP5 Task 5.3 Pre-licensing study,
- WP5 Task 5.4 Guidelines for the demonstration of safety in the further development stages.

<u>WP5 Task 5.1 Generic and specific safety criteria and requirements (task leader: JSI)</u>
The main goal of the task group is to identify and overview the basic general safety criteria and requirements for advanced nuclear reactors developed by the IAEA (International Atomic Energy Agency) and GIF (Generation IV International Forum). The related documents are

[1] WP2 - Materials Testing; WP3 - Thermal Hydraulics and Safety of the SCW-SMR; WP4 - Neutron physics of SCW-SMR.

evaluated according to their applicability for the development of a set of requirements for SCW-SMR.

<u>WP5 Task 5.2 Safety-related findings and conclusions of the WPs 2-4 (task leader: BME)</u>
The aim of this task is to provide answers to the potentially outstanding issues and gaps in knowledge regarding the safety-related behaviour of supercritical water reactor. The task group identifies the safety features of the concept based on the results of the work packages. (WP2 aims to solve material challenges such as behaviour of cladding materials with supercritical water cooling; WP3 investigates the problems of flow and heat transfer in SCW reactor; WP4 deals with simulation issues of SCW-SMR neutronics.)

<u>WP5 Task 5.3 Pre-licensing study (task leader: JSI)</u>
Based on the safety criteria and requirements identified by Task 5.1 a pre-licensing study will be developed by the task group, taking the safety related behaviour and features of the SCW-SMR (from Task 5.2 results) into consideration. The study will define, whether the safety criteria and requirements are met by the design or fulfilment is probable but further research is needed or fulfilment is improbable therefore design changes are needed.

<u>WP5 Task 5.4 Guidelines for the demonstration of safety in the further development stages (task leader: CVR)</u>
The task group will draft guidelines and instructions for the future demonstration of the safety of the SCW-SMR concept based on the results of Task 5.3, which can be applied for the further design development stages of SCW-SMR.

1.1.2 Work done in the frame of WP5 Task 5.2

According to [1], the task of WP5 Task 5.2 is the following:

> *"WPs 2-4 will contribute to the resolution of basic conceptual issues involving material challenges of SCW technology (CIEMAT), problems of flow and heat transfer in SCW (KIT), as well as neutronic aspects of an active core with SCW as coolant and moderator (BME). The aim of this task is to provide answers to the potentially outstanding issues and gaps in knowledge regarding the safety-related behavior of supercritical water reactor. The outcome of this task will summarize the conclusions and lessons learned into a synthetic form to facilitate the future work of designers and nuclear safety analysts (JSI, CVR).*
> *Contributors: BME (leader), CIEMAT, KIT, ENEN, JRC, IPP"*

Although the original aim of the task group is to synthetize the safety related results of WP2-4, it is obvious that under real circumstances the work of the task group WP5 Task 5.2 cannot rely solely on this data. The research work of the work packages is still in progress, so the final results are not fully available yet. On the other hand, the very thorough research activity of the work packages covers only a part of the possible safety features of the reactor design.

In order to get a broader view of the safety-related behaviour of SCW-SMR, additional research has been performed in the form of Phenomena Identification and Ranking Table (PIRT) methodology. This methodology utilizes the common knowledge basis of the participant institutions and experts, consequently, it is able to identify safety issues not covered by WPs research activity.

Considering the previously mentioned factors, the following methodology has been used for D5.2

- Preliminaries
  - Overview of WP5 Task 5.1 results, based on D5.1 report;
  - Overview of Integrated Safety Assessment Methodology (ISAM) and especially PIRT methodology – applicability for WP5 Task 5.2 tasks;
- General knowledge about technology-related safety features
  - Safety features of SCWR technology – overview of earlier SCWR (large scale energetic reactor) designs from safety viewpoint;
  - Safety features of Boiling Water Reactors – overview of typical BWR safety systems (based on the similarities between SCWR and BWR designs), given a special emphasis to advanced boiling water SMR designs;
  - safety features of SMR reactors – overview of international literature to define the specific safety features of small size reactor designs.
- Preliminary safety evaluation of SCW-SMR based on the results of WP2-4
  - Overview of main result of work packages published in WP deliverables, interim reports and scientific publications, determination of safety-related features.
- PIRT analysis for identification of safety-related features other than the ones in the investigated by research projects of work packages.
- Conclusions – safety features of SCW-SMR.

The content of this report reflects the considerations described above.

2 Input data for safety evaluation

2.1 Preliminaries

2.1.1 Evaluation of previous work performed in the field of Generation IV SMR reactors

2.1.1.1 Overview of the results of WP5 Task 5.1

The main purpose of task group WP5 Task 5.1 was to identify generic safety principles, objectives and criteria and specific safety elements applicable to ECC-SMART. The determination of safety elements was based on identifying existing safety elements (based on a structured literature review and applicability analysis) and developing safety elements from identified gaps related to specific fields of safety where accommodation of existing safety elements is not feasible. To achieve this goal, the relevant document of international legal or guidance frameworks (such as IAEA standards and reports, GIF safety assessments and guidelines) and the national legislations of selected countries have been reviewed.

According to the task group report, all compatible safety elements of new Generation 3 reactors apply to SCW-SMR. A part of safety elements specific to small modular reactor designs apply to SCW-SMR depending on the design. The high-level safety requirements of GIF applies to SCW-SMR as well, in accordance with the GIF basic safety approach document[2].

The task group found that majority of the IAEA high level safety requirements apply to the ECC-SMART. IAEA SSR-2/1 standard for design could be used during the SCW-SMR development as a solid basis for the design criteria development, as several requirement are applicable for SCW-SMR. IAEA SRS-123[3] report gives a further guidance on application of IAEA SSR-2/1 for evolutionary and innovative reactor designs.

In the framework of WP5 Task 5.1, the following GIF documents have been reviewed and elaborated by the BME NTI:

- Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System (Rev. 1) (2017) [2];
- Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-cooled Fast Reactor Systems (Rev. 1) (2019) [3];
- An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems (Version 1.1) (2011) [4];
- Super-Critical Water-cooled Reactors (SCWR) Risk and Safety Assessment White Paper (Rev. 1) (2017) [5];
- Supercritical-water-cooled reactor system (SCWR) System Safety Assessment (2018) [6].

These documents have been processed with regard to a possible applicability for ECC-SMART project. According to our results the SFR-related documents cannot be directly applied as a

[2] Basis for the Safety Approach for Design & Assessment of Generation IV Nuclear Systems, Revision 2, July 2021, https://www.gen-4.org/

[3] IAEA Safety Reports Series No. 123: Applicability of IAEA Safety Standards to Non-Water Cooled Reactors and Small Modular Reactors, IAEA, 2023

basis for SCW-SMR safety requirement development. However, the SCWR-related documents can be applied for general technology-related safety evaluation of supercritical water reactors. These results are summarized in Chapter 2.2.1.

The review of ISAM methodology revealed the possibility of application of qualitative safety assessment methods for preliminary safety evaluation of Generation IV reactors. This review serves as basis for our PIRT analysis introduced in Chapter 2.3.3, while the review's main findings are summarized in the next Chapter (2.1.1.2).

2.1.1.2 Integrated Safety Assessment Methodology (ISAM)

The document 'An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems' – hereafter 'ISAM document' has been prepared by the Risk and Safety Working Group (RSWG) of the GIF and was published in 2010, with an update (version 1.1) in June 2011 [4]. In 2014 a supporting Guidance Document for ISAM (GDI) has been released by RSWG to provide the users with further help for the ISAM implementation [14].

The ISAM methodology has been developed in the last decades for a systematic evaluation of safety issues of different nuclear systems. The method is a PSA-based procedure that ensures the thorough safety evaluation of nuclear reactors starting from qualitative evaluation up to a detailed probabilistic and / or deterministic analysis of the affected systems. This systematic approach aims to prevent the necessity of design backfits (i.e. the subsequent modification of the reactor's design or the safety systems to meet the safety requirements).

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

In the following the description of ISAM is given, where majority of text is taken from D5.1 (which is based on information from [4]) to make this report self-standing for the readers. Namely, the ISAM analytical tools have been used, especially PIRT, to make analysis of safety related knowledge gaps on material-related, thermal hydraulics-related and neutronics-related safety features.

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

- *The ISAM is intended for use throughout the concept development and design phases with insights derived from the ISAM serving to influence the course of the design evolution. In this application of the methodology, the ISAM is used to develop a more detailed understanding of safety-related design vulnerabilities, and resulting contributions to risk. Based on this detailed understanding of safety vulnerabilities, new safety provisions or design improvements can be identified, developed, and implemented relatively early.*
- *Selected elements of the methodology will be applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins,*

effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision-makers.

- *The ISAM can be applied in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria. In this way, the ISAM will allow evaluation of a particular Gen IV concept or design relative to various potentially applicable safety metrics or "figures of merit." This post facto application of the ISAM will be especially useful for decision makers and regulators who require objective measures of safety for licensing purposes, or to support certain late-stage design selection decisions.*

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

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

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

ISAM, as a whole concept of an integrated safety evaluation (see *Figure 1*) is not really widely used yet, however some examples can be found in the literature (e.g. by the JAEA for an SFR design[4], or for DEMO fusion project in Korea[5]). However, the single parts of the methodology are quite well known, for example PIRT method is widely used for the safety evaluation of specific systems or phenomena in LWR reactors as well.

[4] Kenichi Kurisaka et al.: Application of integrated safety assessment methodology (ISAM) to Japanese sodium-cooled fast reactor (JSFR), Proceedings of 2010 International Congress on Advances in Nuclear Power Plants (ICAPP '10)

[5] Kyemin Oh et al.: Safety studies on Korean fusion DEMO plant using integrated safety assessment methodology, Fusion Engineering and Design, Volume 89, Issues 9–10, 2014.

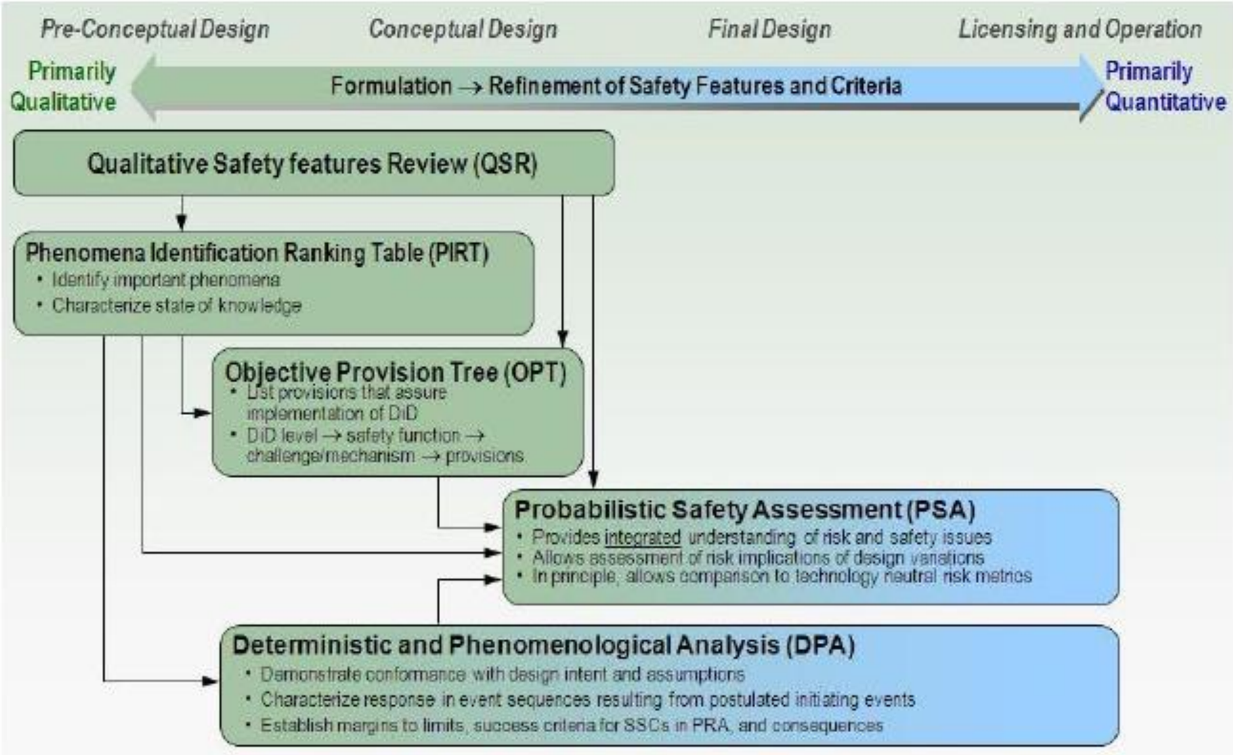

Figure 1: Proposed GIF Integrated Safety Assessment Methodology (ISAM) Task Flow [4]

2.1.1.3 Parts of Integrated Safety Assessment Methodology:

Different elements of ISAM methodology can be applied for different stages of design maturity (see *Figure 1* above). The detailed description of the elements can be found in D5.1 [15].

2.1.1.3.1 Qualitative Safety Features Review (QSR):

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

QSR provides a structured template for designers to help ensure that safety is "built-in, not added-onto" through the early phases of the design of Generation IV systems.

The Qualitative Safety Features Review groups the applicable requirements and recommendations for different defence-in-depth (DiD) levels and different safety functions into classes as it follows:

- *Class 1 – Generic & Technology neutral (i.e., applicable to all the technologies implemented by the innovative systems);*
- *Class 2 – Detailed & Technology neutral;*
- *Class 3 – Detailed & Technology neutral but applicable to a given safety function;*
- *Class 4 – Detailed, applicable to a given safety function, and technology specific, i.e., applicable to a given reactor technology.*

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

2.1.1.3.2 Phenomena Identification and Ranking Table (PIRT)

PIRT is a well-known qualitative process used to identify the importance of safety issues and determine their knowledge. In the framework of the ISAM it means a step of the integrated safety assessment, however, PIRT methodology can also be applied as a stand-alone qualitative assessment for given systems or plant conditions, independently from the other ISAM steps.

The method is based largely on expert elicitation. The process involves selecting hardware (i.e., a given design), selecting an accident scenario (or plant state), and then identifying all plausible phenomena impacting on the outcome of the accident. During the process important phenomena are identified and ranked based on their effect on a selected key parameter (called figure-of-merit, FoM), chosen by a panel of experts. According to [4] '*the "phenomena" can actually be the condition of a particular reactor/system/component, a physical or engineering approximation, a reactor parameter, or anything else that might influence the figure-of-merit.*'

The expert panel members then rank each phenomenon in order of relative importance and its state of knowledge. The ranking process consists of the classification of importance and knowledge level of the phenomena (see *Figure 2*). The typically used categories for importance are High, Medium, Low and Insignificant, but the definition for the categories is arbitrary. For the knowledge level evaluation the level of uncertainty is also evaluated.

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

Table 1: Most Often Used Phenomena Ranking Scales

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

Table 2: Most Often Used Knowledge Based Ranking Scales

Figure 2: Possible ranking of phenomena importance and knowledge level [4]

Based on the PIRT evaluation process, important knowledge gaps can be identified. The results can be used for the determination of further analytical and / or experimental investigations, determining a priority R&D effort (shown red in *Figure 3* below).

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

Figure 3: Results of PIRT process [4]

2.1.1.3.3 Objective Provision Tree (OPT):

The third step of the ISAM methodology is the OPT, which identifies provisions of the given design for the achievement of safety functions for each DiD levels. OPT is a relatively new analytical tool, developed mainly by the IAEA. Based on the results of the PIRT evaluation, it identifies the design provisions necessary for preventing, controlling or mitigating the consequences if the given phenomena occur.

OPT uses a tree structure for the visualization of safety structures. For the each DiD levels, safety objectives and barriers to be protected are identified, and for them the connected safety functions are set. After determination of the possible challenges and their damaging mechanism for a safety function, provisions that are able to cope with the given challenges are listed.

The result of the OPT gives a so-called line of protection (LOP), which includes the necessary provisions that together can cope with the given deterioration mechanism of the challenge.

2.1.1.3.4 Deterministic and Phenomenological Analyses (DPA)

The fourth step of ISAM, DPA consists of using “conventional” deterministic and phenomenological analyses, such as thermal-hydraulic, computational fluid dynamics (CFD), reactor physics analyses, structural models, etc. DPA helps to understand significant phenomena; it supplies input data for PSA and helps to determine the uncertainties.

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

2.1.1.3.5 Probabilistic Safety Analysis (PSA)

The basis of the ISAM methodology is the full-scope PSA. PSA can be applied rather in more matured design phases (from late pre-conceptual design), up to the licensing of the Generation IV system.

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

In WP5 Task 5.1, the applicability of Integrated Safety Assessment Methodology for SCW-SMR development was evaluated by BME. The evaluation results documented in the D5.1 [15] are for the convenience reasons provided below.

The advantages of the ISAM methodology are the followings:

- Applicable for whole design or only for specific systems or phenomena (see examples of JSFR SASS system);
- The method has been developed further since the release of ISAM document [4] in 2011, new guidance created by JRC is available, and some examples of the application are available;
- The method has been already applied for Generation IV reactors (e.g. SFRs). Canada even has an experience with ISAM concerning SCWR reactors;
- Parts of ISAM (as the PIRT method) are well established, with decades of experience with application for Generation III and IV reactors;
- The method can help to identify safety gaps of different designs and to evaluate the needs for further evaluations and research activities (such as the necessity of experimental facilities etc.);
- It can take advantage of number of international experts participating in the ECC-SMART project.

Disadvantages of the ISAM methodology are:

- For the application of the full ISAM methodology a preliminary design is needed in order to determine the phenomena and systems being subject of evaluation, thus some design decisions have to be made concerning the size and technology of the reactor (see Table 1);
- At the present pre-pre-conceptual design phase only the first and second (partly the third) steps can be used;
- The qualitative analysis is based on the judgement of an expert panel, so the adequacy of the result relies on the members of this expert panel – appropriate number and qualification of experts is necessary;
- The quantitative analysis (OPT, deterministic analysis of selected phenomena and PSA analysis) needs experts with large experience, mainly in field of PSA;
- The ISAM, as a whole approach has not an exhaustive literature, however, some parts of it (mainly the PIRT) has been used for a wide range of nuclear technologies.

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

Table 1: Applicability of different ISAM steps during the design evolution [4]

2.1.1.5 Application of PIRT methodology for Generation IV reactors and for SMR designs

Due to its flexibility the PIRT methodology offers wide range of possible applications. It can be used at different design phases with a different level of details, for general design processes or for the evaluation of a given safety system or a given operational condition or type of accident.

PIRT has been applied in some Generation III reactors and Generation IV concepts as well. An example is the application for the Japan Sodium Fast Reactor – JSFR, which evaluates the operation of the planned self-actuation shutdown system (SASS) (see [4]). For JSFR, the whole ISAM methodology has been applied – as a part of it, the necessary R&D activities have been determined with the help of the PIRT analysis. This application also showed a possibility for use of expert evaluation for the description of change of knowledge level: after the identification of key phenomena (specifically for the SASS system) and performance of the R&D activity, the expert ranking has been performed once again, showing a solid increase of the knowledge level concerning the key phenomena. For more details the reader can refer to [4].

The PIRT method has also been used in the EU-funded **ELSMOR** project (Towards European Licencing of Small Modular Reactors)[6]. ELSMOR aims to create methods and tools for the European stakeholders to assess and verify the safety of light water small modular reactors (LW-SMR) that would be deployed in Europe. Tasks of the project are:

- Collection, analysis, and dissemination of the information on the potential and challenges of Small Modular Reactors to various stakeholders, including the public, decision makers and regulators.
- Development of the high level methods to assess the safety of LW-SMRs.
- Improvement of the European experimental research infrastructure to assist in the evaluation of the novel safety features of the future LW-SMRs.
- Improvement of the European nuclear safety analysis codes to demonstrate the capability to assess the safety of the future LW-SMRs.

In the ELSMOR project – among other investigations – two sets of PIRT analysis have been performed. The first one describes the behaviour of the Safety Core Cooling safety function, while the other one deals with Passive Containment Cooling. [7][8]

These analyses do not consider the exact systems of one specific SMR reactor type, but the general passive safety systems of Generation 3 Light-Water Cooled SMRs has been taken into account, described based on different SMR designs (mainly on IRIS by Westinghouse and SMART by KAERI for core cooling systems and mainly on NUWARD and NuScale for passive containment systems).

The figures of merit (evaluation criteria) were defined as: *The importance of a phenomenon is proportional to the impact that it has on the risk of uncovering or overheating the nuclear fuel in the core.*

[6] https://cordis.europa.eu/project/id/847553

For the evaluation of the Safety Core Cooling safety function, an expert panel, i.e., a workgroup of twelve people has been formed (six of which from CEA, three from CIRTEN, one from ENEA, one from GRS and one from IRSN). The experts have been asked to evaluate the given phenomena based on their importance (regarding the figure of merit) and the scientific-technical knowledge level. Both rankings were performed on a 3-level scale (high / medium / low).

Figure 4 shows the ranking scale used by the experts.

Rank	Weight	Definition with respect to the importance	Definition with respect to the knowledge
High (H)	1.0	The phenomenon is judged important according to the evaluation criteria	The phenomenon is well known and understood. Experimental data is available. Models are validated.
Medium (M)	0.5	The phenomenon is judged moderately important according to the evaluation criteria.	The phenomenon is partially known and understood. Experimental data is available but in small quantity / quality.
Low (L)	0.0	The phenomenon is judged of little or no importance according to the evaluation criteria.	The phenomenon is poorly known and understood. Little or no experimental data is available. No validated models.

Figure 4: Evaluation scale used in ELSMOR PIRT [8]

It is worth noticing the weighting factors used to evaluate of expert rankings. The weight 1.0 / 0.5 / 0.0 was given for rankings of high / medium / low rankings, respectively.

Based on the weighting factors the importance level (IL), knowledge level (KL) and standard deviation for each category have been defined:

$$Importance\ level \stackrel{\text{def}}{=} IL \stackrel{\text{def}}{=} \frac{0.0\ N_L + 0.5\ N_M + 1.0\ N_H}{N_L + N_M + N_H}$$

where Nn_L, N_M and N_H represent respectively the number of 'low', 'medium' and 'high' votes assigned to the importance.

$$Knowledge\ level \stackrel{\text{def}}{=} KL \stackrel{\text{def}}{=} \frac{0.0\ n_L + 0.5\ n_M + 1.0\ n_H}{n_L + n_M + n_H}$$

where n_L, n_M and n_H represent respectively the number of 'low', 'medium' and 'high' votes assigned to the knowledge.

$$Standard\ deviation\ on\ importance \stackrel{\text{def}}{=} \sigma(IL) \stackrel{\text{def}}{=} \sqrt{\frac{N_L(0.0 - IL)^2 + N_M(0.5 - IL)^2 + N_H(1.0 - IL)^2}{N_L + N_M + N_H - 1}}$$

$$Standard\ deviation\ on\ knowledge \stackrel{\text{def}}{=} \sigma(KL) \stackrel{\text{def}}{=} \sqrt{\frac{n_L(0.0 - KL)^2 + n_M(0.5 - KL)^2 + n_H(1.0 - KL)^2}{n_L + n_M + n_H - 1}}$$

With the help of the weight factors, the importance level and knowledge level values are normalized (values can range between 0 and 1).

For the definition of the screening level of the most important issues the relative relevance and relative dispersion of the expert rankings has been defined, dividing the IL and KL values with the corresponding average values, showing the significance of a given phenomenon compared to the others.

In the PIRT analysis concerning the Safety Core Cooling, two main accident categories were analysed. The first one is a Loss of Offsite Power (LOOP) event without classified diesel generator support, and global passive systems used for the core-cooling situation. This event includes the operation of a hypothetical passive decay heat removal system through the secondary circuit (passive SG cooling). The second accident type is a Loss of Coolant Accident (LOCA) accompanied by a loss of electric power supply event, using a safety injection system with an autonomy time of 72 h. A variation of the latter transient is the assumption of long-term cooling with a closed-cycle system. Importance ranking has been performed for short-term and long-term as well.

Phenom.	Importance ranking								Knowledge ranking				Screening parameters			
	Phase 1				Phase 2								Phase 1		Phase 2	
	L	M	H	Importance level	L	M	H	Importance Level	L	M	H	Knowledge level	Relative relevance	Relative dispersion	Relative relevance	Relative dispersion
α	2	3	1	0.42	0	3	3	0.75	1	1	4	0.75	0.17	0.93	0.31	0.67

Figure 5: Example of PIRT results for a given phenomenon [8]

According to the PIRT analysis, the phenomena with the highest relative relevance (and the ones with highest relative dispersion) have been listed for the short-term and long-term behaviour as well, revealing the significance of fouling processes on the heat exchanger surfaces of the assumed safety condenser system.

The PIRT analysis for passive heat removal from the containment (as part of WP4 of ELSMOR project) [9] investigated the LWR SMR types with large water pools for passive containment cooling, such as the NUWARD and the NuScale designs. Both designs are integrated PWRs with relatively small steel PCV (primary containment vessel), which acts as a heat exchanger for transportation of decay heat to the water pool in which the PCV is submerged. The steel PCV also plays a role in severe accident management helping the outer flooding of RPV for the in-vessel retention (IVR).

An important consequence of integrating the whole primary circuit into the RPV is the elimination of LB LOCA (large break loss-of-coolant accidents) as no large diameter connecting pipes exist in the primary circuit. For this reason, only SB LOCA scenario has been considered in the PIRT analysis. SB LOCA has been investigated as DBC event and – assuming the failure of the active or passive emergency core cooling systems – also as DEC initiating event. Station blackout event has been investigated as well, assuming the loss of passive containment cooling for NUWARD.

The figure of merit for the containment analysis was "the assurance of containment integrity by keeping its pressure well below design limits".

For the evaluation of the Containment Cooling safety function, an expert panel, i.e., a workgroup of eight people has been formed (2 of which were from LEI, two from GRS, one from JRC and three from POLIMI and ENEA). The experts have been asked to evaluate the given phenomena based on their importance (regarding the figure of merit) and the scientific-technical knowledge level. Both rankings were performed on a 3-level scale (high / medium / low). The weighting of the expert votes is the same as for the core cooling PIRT analysis (see *Figure 4*).

An interesting approach for this PIRT analysis was further classifying knowledge level votes based on three knowledge categories (physical, experimental and modelling). The expert ranking was performed for all knowledge categories and averaged in order to get only one specific value for the given phenomenon's knowledge level.

The definition of knowledge level (KL), importance level (IL) and their standard deviation is the same as for the core cooling analysis. The relative relevance and dispersion of the values has also been calculated.

According to the SB LOCA analysis results, the most significant phenomena (with the highest relative relevance) regarding the containment cooling are the heat transfer from the containment atmosphere to the containment wall under condensation; the heat transfer from the inside of the RPV through the RPV wall after melt relocation; the heat transfer to the water pool under boiling conditions; the flow field and the water level of the water pool; the possible thermal stratification in the pool and the heat transfer from the water pool to the outer building structures.

For Station Blackout (SBO) analysis, the phenomena with the highest relative relevance were heat transfer through the RPV wall after melt relocation; the flow field in the water pool; the heat transfer from the containment atmosphere to the containment wall under condensation and to the water pool under boiling conditions; the water level of the water pool and the possible thermal stratification in the pool.

Based on the literature review for ISAM methodology[7] and especially for PIRT analysis method, **it was concluded that PIRT methodology in itself also can support the definition of safety-related features of SCW-SMR** (i.e. the Task 5.2). The involvement of a large group of SCWR experts may help to expand the range of considered behaviour of SCW-SMR.

Chapter 3.2.7 of Deliverable D5.1 already recommended that "*An exercise to apply the ISAM should be performed on the SCW-SMR concept as soon as a sufficiently detailed SCW-SMR*

[7] Some further PIRT analyses have been reviewed, but not presented here as they investigated non-SMR and non-SCWR specific issues, such as the risks of accident-tolerant fuel (ATF) in large LWR reactors (Geelhood, Luscher: *Degradation and Failure Phenomena of Accident Tolerant Fuel Concepts Chromium Coated Zirconium Alloy Cladding*, DOE, 2019) or severe accident phenomena in large Gen 3+ PWR reactors (Magallon et al.: *European expert network for the reduction of uncertainties in severe accident safety issues (EURSAFE)*, Nuclear Engineering and Design, February 2005). However, these analyses served as a good starting point for our work, because of being independent PIRT analysis, not part of global ISAM assessment. The methodology of this analysis is quite similar to ours, evaluating general failure / accident phenomena by a larger expert panel.

concept becomes available”[8]. However, it should be clear that the PIRT analysis presented in this report does not necessarily contribute to the ISAM assessment planned for the next SCW-SMR related project.

[8] Chapter 3.2.4 had a more detailed recommendation on the topic: „*Apply the ISAM methodology as early as possible in the design process of the SCW-SMR concept. Its application may be prepared during the lifetime of the ongoing ECC-SMART project then started in the next project by the application of first, second, third, fourth and fifth steps of ISAM.”*

2.2 General knowledge about technology-related safety features

2.2.1 Safety features of SCWR technology

As it is considered as one of the six promising Generation IV reactor design by the GIF, there is a quite extensive literature available about the Supercritical Water-cooled Reactor technology. The elaboration of this literature can give a general overview of the safety-related features of SCWR technology.

2.2.1.1 Literature review of SCWR reactor designs

Super-Critical Water-cooled Reactors (SCWR) Risk and Safety Assessment White Paper

The White Paper document has been developed by the GIF Risk and Safety Working Group (RSWG), the GIF Super-Critical Water-cooled Reactor (SCWR) System Steering Committee (SSC), and the GIF as a whole institution [5]. The White Paper provides an overview of activities conducted by participating members of the GIF in the application of the ISAM developed by the GIF RSWG. Based on the application of the ISAM, the GIF SCWR SSC has elaborated the future Research and Development (R&D) needs for the conceptual design of SCWR systems. Several areas have been identified which do require additional R&D and analysis in order to improve the risk and safety performance of the different SCWR concepts. The White Paper provided a summary of the identified areas for improvement [5]. This White Paper's objective was to assess of the adequacy of safety provisions incorporated in the conceptual design completed so far for the SCWR systems.

The White Paper focused on the safety of the Canadian SCWR concept, and only a supplementary description was provided on the High Performance Light Water Reactor (HPLWR or European SCWR).

Supercritical-water-cooled reactor system (SCWR) System Safety Assessment (SSA) [6]

This document summarized the lessons concluded from system safety assessments of SCWR relevant reactor developments in the past and the current ongoing SCWR developments (up to 2018). The Canadian SCWR concept represented the pressure tube type while the European HPLWR the pressure vessel type SCWR concepts among the six existing SCWR concepts.

According to the SSA document, 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. Basically, the design of the safety systems adheres to the "Defence-in-Depth" (DiD) safety principle [6].

Multiple fields where progress needed have been identified in SSA document which are summarized in the following:

- First of all, the application of novel manufacturing processes (for example the new fuel assembly concepts like the Canadian fuel assembly containing 64 fuel rods) carry a risk due to they have a first-of-its-kind design and may introduce new phenomena occurring in the new design itself (for example the re-entrant flow in case of new Canadian SCWR fuel assembly design).

- The aggressive chemical effects of SCW in the reactor core (on the fuel sheath, liners, pressure tube, and insulators in case of the Canadian SCWR / on the fuel cladding, wrapped wire, fuel assembly wall and moderator box wall and structural elements of lower and upper mixing chambers in case of the HPLWR) will require extensive experimental testing and demonstration.
- Furthermore, due to SCW as a coolant has not been used in such highly radiative environment like the reactor core of an SCWR, the required understanding to adopt the SCW requires significantly high level of experimental testing and demonstration similarly like at the previous point.
- Due to many simplifying assumptions and extensions of existing data and methods have been applied during the development works of the different SCWR design concepts, thus the correctness of assumptions and extensions do require confirmation.
- It seems that 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.
- It was explained that additional R&D data would be required in the field of “in-reactor” data despite there is reasonable confidence that in-reactor performance should be satisfactory during the relevant operation types of the reactor.
- Many material issues have been identified as well. For example, the maximum so-called “diametral strain” estimated for the pressure tube of the Canadian SCWR concept after 75 years of full power operation would require validation. The presented estimate in the SSA document has been based on a very limited amount of data, thus in-core irradiation experiments are required at SCWR conditions to validate the presented estimation.
- Last but not least, additional experimental tests which provide measured data on the thermal conductivity, fuel qualification, and performance of thorium-plutonium dioxide ($(Th, Pu)O_2$) fuel would be required in order to enable the implementation of this kind of fuel in different SCWR concepts.

<u>High Performance Light Water Reactor - Design and Analyses [10]</u>

The book *High Performance Light Water Reactor - Design and Analyses* [10] edited by Thomas Schulenberg and Jörg Starflinger is a basic literature for Supercritical Water Cooled Reactors. It describes the technological design, the safety systems, and the performed analysis for the HPLWR reactor design.

The book [10] describes the most general features of SCWR designs as:

- Direct, once through steam cycle is possible, so steam separators, steam generators and dryers can be eliminated.
- With proper design, even main coolant pumps can be omitted.
- Because of the lack of phase change, the possibility of boiling crisis can be eliminated.

According to historical data, high temperature boiling water reactor (Heissdampfreaktor - HDR) has been tested in Germany in 60s, proving the feasibility of the concept, but showing minimal cladding damage at the hottest places of the core.

HPLWR is a European design for Supercritical Water Cooled Reactor with a net electric power of 1000 MW, developed by a European consortium of more than 10 participating institutes. The high coolant parameters (at least 500°C core outlet temperature and 25 MPa coolant pressure) could result in a net plant efficiency of around 44%. Special design feature of the HPLWR is the unique coolant flow path with 3 vertically oriented heat-up steps (so called passes) - which is necessary because of the high enthalpy rise of the supercritical water - and the physical separation of moderator and coolant supercritical water. The flow direction of the coolant is vertically upward in the first, so called Evaporator pass, as well as in the third so called Superheater 2 pass, while it is downward in the second Superheater 1 pass.

An important feature of HPLWR is that – similarly to boiling water reactors – the core power distribution is heavily influenced by the coolant density distribution through the neutron moderation parameters. 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. The large pressure and temperature differences between the reactor inlet and outlet may cause higher stresses for in-vessel equipment and also increases the risk of moderator bypass flow into the coolant channels. Some of the original target parameters could not be met in the core design: high peak cladding temperature, low fuel burn-up and high hot channel factors require further analyses. The proposed use of stainless steel in fuel clusters has some disadvantage for the neutron balance because of high neutron absorption. Further analysis and tests are necessary for decreasing the large uncertainties of heat transfer predictions.

Concerning the development of the safety systems, the requirements known from the GIF initiative and EUR requirements have been used as general guidelines. The highest level of requirements considered were the three major GIF safety goals: (1) the high level of safety and reliability; (2) the low probability of reactor core damage; and (3) the elimination of the need for off-site emergency response. The concept of design in depth (DiD) is applied for all plant states. The preliminary targets for different design basic conditions can be seen in Table 2.

Further criteria for design extension conditions are the use of diversified systems and resistance against external events for DEC1 conditions, and the use of accident mitigation systems and prevention of early containment failure for DEC2 states.

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

Table 2: Plant states and preliminary set target parameters for design basis conditions [10]

Figure 6 shows a comparison on the general features of the simplified control systems of a BWR and a SCWR. As it can be seen on the left part of *Figure 6*, the so-called once-through system of a BWR has an important safety advantage in terms of general safety system requirements: it has a closed coolant loop inside the reactor during emergency situations. Thus in case of a BWR the residual heat can be removed by natural convection (driven by the rising steam in the core and above) and the safety system has to ensure sufficient amount of coolant inventory into the reactor pressure vessel (RPV) to keep the core covered with water. On the other hand, in case of a SCWR the residual heat can be removed only by forced convection inside the RPV, which forced convection may be driven by a natural convection loop outside the RPV. Thus the requirement for the safety system in case of a SCWR is to ensure sufficient coolant mass flow rate to remove the decay heat from the reactor core [5]. Despite this fundamental difference between the general features of the simplified control systems of a BWR and a SCWR, a BWR has numerous safety system requirements, which can be directly applied to a SCWR concept without significant modifications.

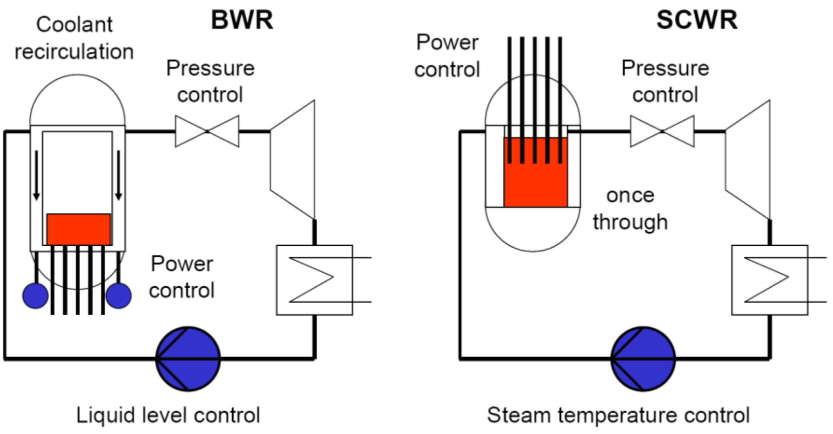

Figure 6: Comparison on general features of simplified control systems of a BWR and a SCWR [5]

Concerning the design of the safety systems, the similarities and differences between HPLWR and BWR reactors has been analysed in [10]. One major difference is the lack of recirculation loop in case of HPLWR as described above. This results not only in different control systems for normal operation condition (i.e. pressure, temperature, and power control) but means also a basic difference for safety systems. BWR reactors apply natural convection for decay heat removal after shutdown of the reactor, which is impossible for the HPLWR.

The main similarities concerning safety systems are the following:

- the reactor can be shut down safely with control rods or the diverse emergency boron injection system (however, it is worth to note that in supercritical water cooled reactors boric acid reactivity control is not used for normal operation - similar to BWRs -, which can make the use of emergency boron system somewhat complicate);
- because of the use of one coolant circuit, the isolation of containment (with active or passive containment isolation valves) is inevitable for accident conditions;
- the increasing coolant pressure after shutdown or in accident conditions can be controlled with the help of pressure relief valves;
- the automatic depressurization system (ADS) blows down steam into an in-containment pressure suppression water pool, similar to BWR wetwells;
- emergency injection system is necessary for supplying water after depressurization or in case of coolant leakage;
- a residual heat removal system is needed for the long-term heat transport into the environment.

The minimum set of safety systems for HPLWR can be seen in *Figure 7*.

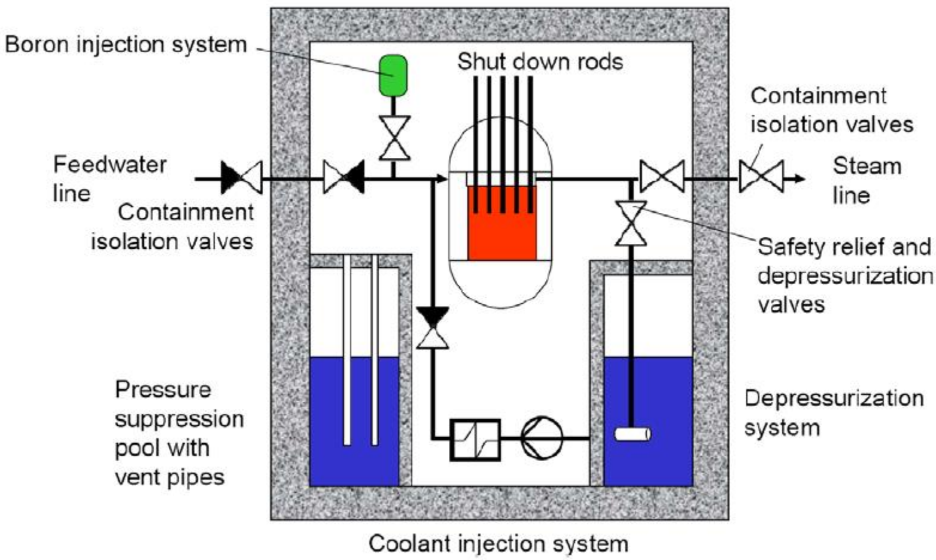

Figure 7: Scheme of necessary HPLWR safety systems [10]

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

According to [10] a general difficulty of the HPLWR core design is its rather high coolant pressure drop as well as the meandering coolant flow path in the core, which requires a higher pressure head of a passive coolant loop than for a BWR.

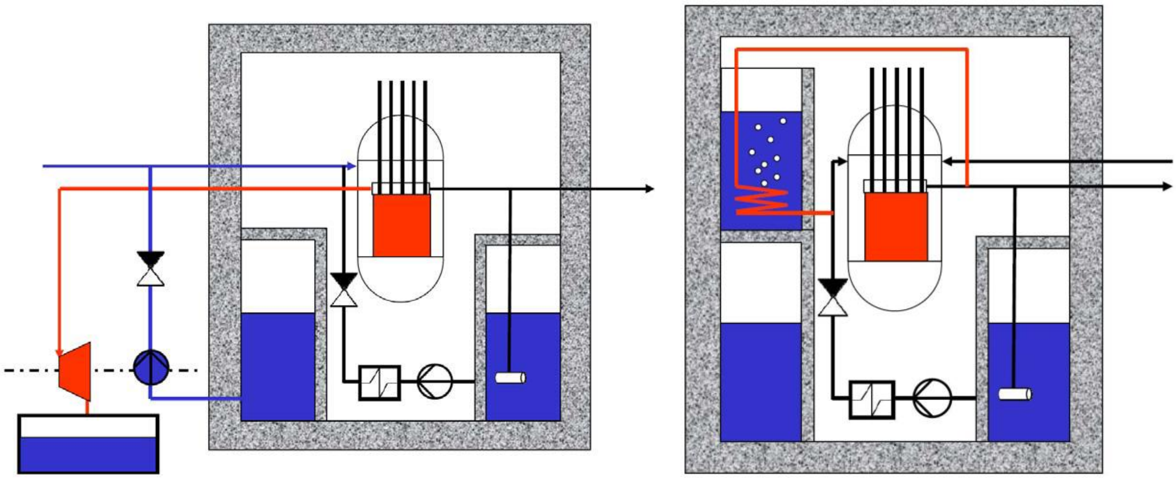

Figure 8: Depressurization through a steam turbine driven injection pump (left) and into an emergency condenser (right) [10]

As for the last engineering barrier, a pre-stressed cylindrical concrete containment has been designed for the HPLWR. The compact containment has an inner diameter of 20 m and inner height of 23.5 m, designed for an internal pressure of 0.5 MPa. On the top of the traditional BWR-like pressure suppression pool (wetwell) there are 4 upper pools located on the top of the containment building with a combined volume of 1100 m^3.

This upper pool is used for the automatic depressurization system and it contains the submerged emergency condensers as well. There are also 4 containment condensers mounted on the ceiling of the drywell, connected to the upper water pool. The containment systems can be seen on *Figure 9*.

The book [10] investigated the applicability of widely used system codes for supercritical water coolants. In most codes the properties of supercritical water can be added as a single-phase fluid. However, the description of potential supercritical to subcritical pressure can be problematic. This challenge can be solved by applying a pseudo two-phase flow with a 6-equation model, which means the numerical description of supercritical fluid as a two-phase medium.

Another heat transfer issue is the lower heat storage capacity of HPLWR compared to traditional light water reactors, which can cause faster progress of accidents.

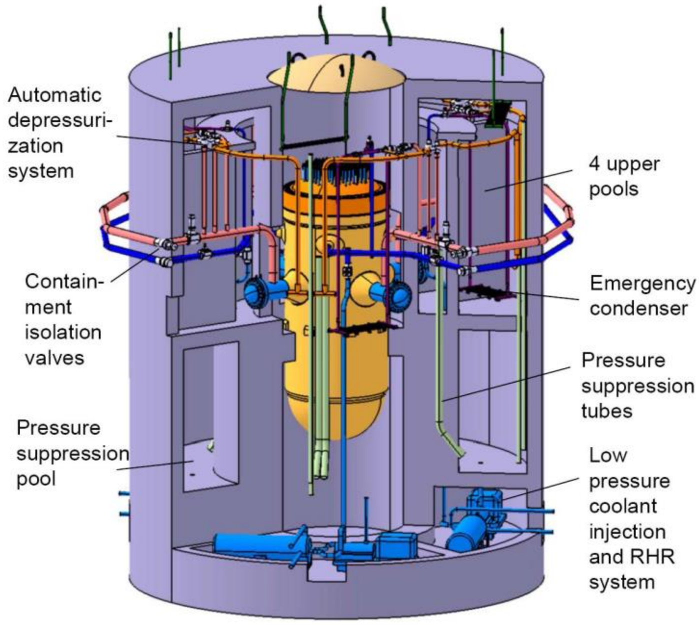

Figure 9: The containment passive systems of the HPLWR [10]

The book [10] included also the analysis of some transients (e.g. loss of feedwater, LOCA, control rod ejection etc.) using the aforementioned considerations. The calculations showed the compliance of the safety systems but also revealed some challenges (such as flow instabilities or high fuel temperatures in case of reactivity-induced transients).

It has been revealed by the simulations that in case of a total loss of feedwater (TLFW), combined with an anticipated transient without scram (ATWS) the cladding temperature distribution can be described only with 3D coupled codes. For a large break LOCA in the feedwater line, the first seconds seem to be critical concerning the cladding temperature.

2.2.1.2 Conclusion

In Chapter 2.2.1., some SCWR-related documents have been reviewed. Based on these, a number of safety-related special issues can be identified (see above). A part of these conclusions can be applied for the SCW-SMR as well. These include:

- Material issues (such as the chemical effects of SCW in the reactor core; the effect of radiation on SCW or the irradiation-induced changes in material properties of new cladding materials);
- Technology issues (such as the risks of novel manufacturing processes);
- Reactor physical issues (such as the strong dependence of core power distribution on the coolant density distribution through the neutron moderation parameters; possible Xenon oscillation instabilities or optimization of fuel burn-up);
- Thermal-hydraulic issues (such as coolant and moderator flow stability; optimization of peak cladding temperatures or large uncertainties of heat transfer predictions).

However, because of the differences between SCW-SMR and large SCWR designs (e.g. horizontal fuel assemblies or 7-step heat-up section) some of the gained experience cannot be applied for our case.

2.2.2 Safety features of BWR reactors

As it was shown above, the basic system arrangement of SCWRs is similar to boiling water reactors', so it is worth reviewing the safety features of BWRs.

Boling Water Reactors have had a great history in nuclear energy production, with a continuous development path since the 50's. The first commercial BWR unit (Dresden, USA, 200 MWe capacity) started its operation in 1960. The most popular BWR design was the BWR series of General Electric (GE), but also other designs were available on market (such as Siemens-KWU BWR, ABB Atom BWR, etc.). Present operating BWR designs (Generation 2) use a direct steam cycle with forced coolant flow. The boiling of the coolant occurs inside the reactor core, resulting in 12-15 m% steam content at core outlet. For the separation of the water droplets and dry steam 2-stage moisture separators are operating above the core.

The typical operating parameters for BWRs are about 70 bar coolant pressure and the corresponding saturation temperature (about 285˚C). The separated steam flows to the turbines, and after expansion and condensation it is forced back to the reactor vessel with the help of feedwater pumps. The feedwater and the remaining water from the separators is mixed and forced to the core inlet with the help of the so-called recirculation pumps, driving in-vessel jet pumps.

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.

Present BWR fuel assemblies are similar to PWR FAs, usually with a lower number of fuel rods and with an outer fuel shroud to prevent flow instabilities. The reactivity is controlled with the help of control rods and the recirculation pumps described above. The control rods are inserted from the bottom of reactor vessel (because of the higher reactivity worth in the water volume and the presence of moisture separator systems). Because of the different insertion, emergency shutdown cannot utilize gravity as a passive driving force as in case of PWRs. Instead, a special hydraulic control rod drive mechanism is widely used, applying pressure difference for moving the control rod main drive piston.

As the core cooling water is flowing directly on the turbines (outside the containment building), potentially contaminated coolant can get out of the containment building. The lack of separated primary and secondary circuit means also the lack of the primary pressure boundary as an engineering barrier. This special feature of BWRs also affects safety systems, as it is necessary to ensure the fast isolation of the containment building from the environment in case of an accident. For this purpose, the Main Steam Isolation Valves (MSIV) and containment isolation valves (CIV) are inevitable.

The isolation of the main steam line requires an efficient pressure suppression system being able to accept the produced steam in case of an isolation. For this purpose practically all the operating BWRs have a pressure suppression water pool, the so-called wetwell. This makes possible also having smaller containment free volumes (drywell), as it is possible to blowdown from the containment building as well in case of containment overpressure. For present BWRs the volume of the drywells is about 5000-7000 m^3, while the volume of the wetwell water pool is about 3000-4000 m^3.

For isolated states BWRs also have passive or active cooling systems for residual heat removal. Older BWRs have isolation condensers, i.e. heat exchangers in a large in-containment water pool, while other BWR designs apply the so-called Reactor Core Isolation Cooling System (RCIC), which is a steam-driven cooling system injecting external water into the reactor. The emergency core cooling systems (ECCS) usually consist of high-pressure and low-pressure systems.

In the 90's, Generation III BWRs have been developed. The only Gen. III BWR with real construction and operating experience is the Advanced Boiling Water Reactor (ABWR) by the GE-Hitachi-Toshiba. The first ABWR reactor was the Unit 6 of the Kashiwazaki Kariwa NPP in Japan. Altogether 4 units have been constructed and started in Japan, however, none of them have been restarted since the Fukushima Daiichi accident. ABWR introduced some advanced technology features (such as internal reactor pumps instead of external recirculation loops), as well as passive heat removal systems for reactor cooling and containment cooling for DEC conditions. The passive emergency condenser is able to remove heat from containment even after a reactor vessel melt-through.

The ESBWR (Economic Simplified BWR) developed by GE-Hitachi is an advanced Gen. III+ reactor design. The US NRC has issued a design certification for the ESBWR in 2015. The reactor has a natural convection cooling even in normal operation conditions, thus eliminating the need for recirculation pumps (see in
Figure 10).

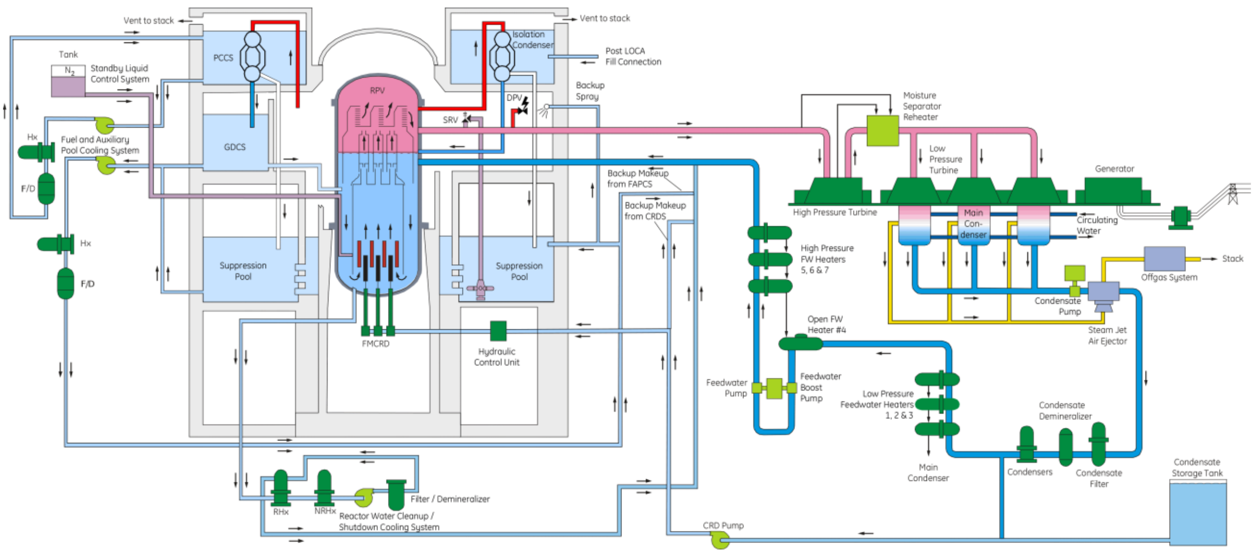

Figure 10: Safety systems of ESBWR. On the top of the containment building, the isolation condenser and the PCCS (passive containment cooling system) can be seen. On the left side, the extra water pool is used for gravity-driven emergency injection system (GDCS). [Source: GE Hitachi]

A specific BWR design is the BWRX-300 by the GE-Hitachi, which is under pre-licensing in Canada. The design has been selected by Ontario Power Generation for the new units of

Darlington NPP in Canada, potentially being the first SMR in North America. The most important feature of BWRX-300 is the simplification – the use of integral RPV isolation valves that mitigate the impacts of LOCAs, and large-capacity

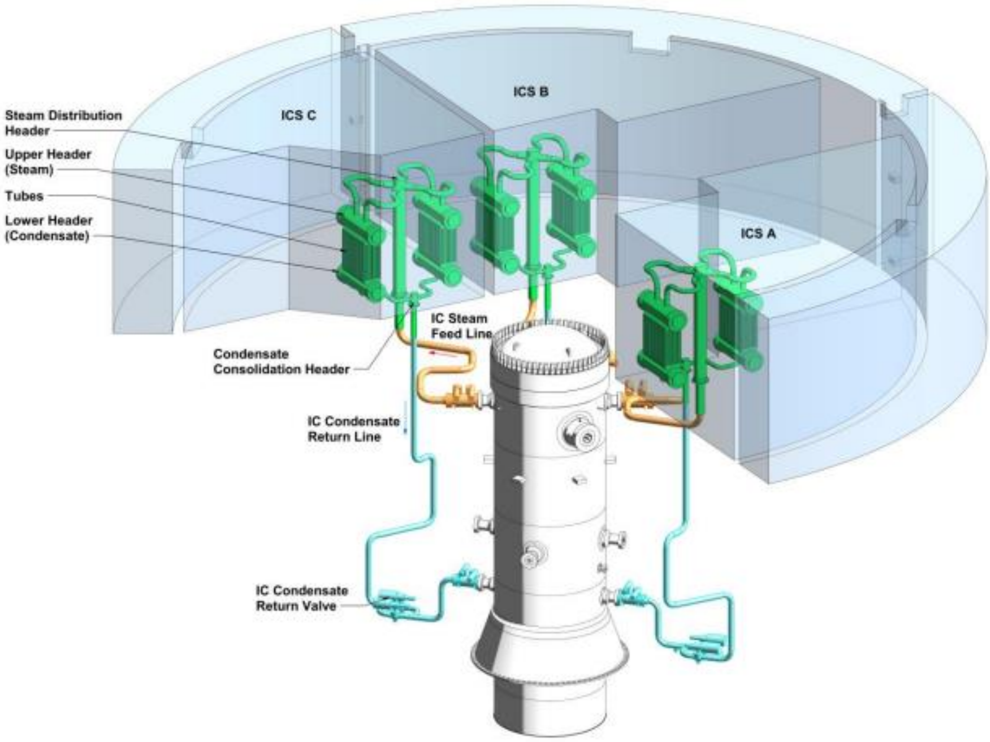

Figure 11: Isolation condenser system of BWRX-300 [11]

The isolation condensers (ICs) provide overpressure protection without the need for safety relieve valves. The ICs also act as the ECCS, utilize natural circulation, and require no Alternating Current (AC) power to perform their functions (see in *Figure 11*). The key elements of the safety systems are the reactor isolation valves, so any traditional non-isolable LB LOCA accidents can be eliminated as well. BWRX-300 has a dry containment, which can withstand to LOCA accidents. For accident conditions, the cooling of the containment is performed passively, into the equipment pool with the help of three independent heat exchangers.

2.2.3 Specific safety issues of small modular reactors (SMRs)

There has been a great interest in small modular reactors in the last decade, as this new technology is envisioned as one of the possible pathways of nuclear energy in the future. Considering their flexibility, possible mass production, modular construction, manoeuvring capability, SMRs can offer a feasible energy system also to newcomer countries and for expanding nuclear markets.

The World Nuclear Association (WNA) summarizes the special safety-related features of SMRs as follows[9]:

- small power, compact architecture and extensive employment of passive safety concepts;

---

[9] https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-power-reactors/small-nuclear-power-reactors.aspx

- modular, in factory fabrication, which also facilitates higher quality standards due to mass production and in-factory quality assurance;
- the smaller core size and the resulting low thermal power leads to reduction of source term, i.e. smaller possible releases in accident conditions;
- many of the SMR reactor designs can be located underground or underwater, providing more protection against natural and man-made external hazards;
- an important consideration is the possibility of multiple units on the same site, meaning a challenge for accident management.

Based on the available SMR designs, it can be seen that the smaller amount of fission products in the core - meaning lower decay heat production together with the large specific core surface - makes possible to apply natural convection for passive residual heat removal systems. In some cases after a given cooling time (e.g. 30 days for NuScale design) air cooling can be enough for decay heat removal, which ensures independence and makes an infinite grace period possible.

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

2.3 Project-specific knowledge about safety features

2.3.1 Summary of research area covered by WP2, WP3, WP4

According to [1], WP5.2 synthetizes the results of WP2-4 research work. “The outcome of this task will summarize the conclusions and lessons learned into a synthetic form to facilitate the future work of designers and nuclear safety analysts.”

For this work, the following deliverables and interim reports of WP2-4 have been elaborated:

- D2.1 Test matrix based on available materials;
- D2.2 Report summarising basic characterization of materials and specimens machining;
- D3.1 Report summarizing the newly generated reference data for natural convection, forced and mixed convection and decay heat removal;
- D3.2 Report on the results of the benchmark exercise;
- D4.1 Neutron physics code selection results;
- D4.2 Analytical investigation of neutron physics parameters relevant to the safety and feasibility of the SCW-SMR;
- M18 Periodic Reports of WP2, WP3 and WP4.

2.3.2 Results of WP2, WP3, WP4

2.3.2.1 Main findings of WP2

According to [1], the main objective of WP2 is to identify the licensing-related issues and knowledge gaps by gaining more in-depth knowledge of the corrosion behaviour of candidate materials for the Small Modular Reactor cooled by Supercritical Water under non-irradiated and irradiated conditions. Moreover, electrochemical measurements inside of the supercritical region will help to understand how changes in the physicochemical properties can affect the corrosion mechanisms that take place in this environment. In addition to this, this work package also studies radiolysis processes in supercritical water, essential for defining the chemistry of these reactors in the future.

In order to meet this objective, WP2 covers an extensive experimental program for the behaviour of potential cladding materials under simulated SCWR conditions with irradiated and non-irradiated materials (i.e. oxidative and radiation environment under SCW conditions) and for the investigation of SCW corrosion-related parameters (such as radiolysis). For these purposes the international cooperation of experimental facilities’ operators is necessary.

Subtasks of WP2 are:

- WP2 Task 2.1: Selection and characterization of materials
- WP2 Task 2.2: Study of the corrosion and EAC behaviour of selected alloys out-of-pile
- WP2 Task 2.3: Study of the corrosion and EAC behaviour of selected pre-irradiated alloys in SCW
- WP2 Task 2.4: Study of the effect of SCW-SMR chemistry and changes in the chemical properties of SCW in the behaviour of candidate materials

As this experimental program runs for the whole length of the ECC-SMART project, at the date of D5.2 unfortunately only preliminary results of WP2 are available. These include the set-up of the experiment matrix, the selection of the candidate materials and the characterization of materials.

For the collection of safety-related results of the ECC-SMART project the continuous data gathering has been initiated by BME in the form of a live Excel worksheet ("Safety-related findings and conclusions live Excel file"). However, the voluntary input to this data collection from WP2-4 partners proved to be quite limited and general.

In order to get a broader view about material-related safety features of SCW-SMR an expert evaluation (in the framework of a PIRT analysis) has been initiated for the project partners of WP2, utilizing the enormous knowledge and experiment of the participants in the field of material behaviour under supercritical condition. The results of WP2 PIRT analysis are introduced in Chapter 2.3.3.1.

Based on the aforementioned live Excel file, the following issues have been identified previously by the participants:

- <u>Oxidation behaviour at long time exposures</u> – Currently, there is not sufficient knowledge available in the literature for the project partners. Thus there is a gap in our knowledge regarding the oxidation behaviour at long time exposures of the candidate structural materials (SS310S, 800H and AFA). This "material issue" is under investigation during the project by the WP2 partners.
- <u>Effect of irradiation on cladding materials</u> – There is not sufficient knowledge available in the literature for the project partners in this topic. Thus there is a gap in our knowledge regarding the effect of irradiation on the candidate cladding materials (SS310S, 800H and AFA). This "material issue" is under investigation as well during the project by the WP2 partners. As D5.1/3.2.6.1. sub-section states: "It seems that 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."
- <u>Effect of the manufacture and shaping of tubes in the crack initiation behaviour of candidate materials</u> – It is valid here as well, that currently, there is neither sufficient knowledge publicly available in the literature nor suitable own experience on the effect of the manufacture of tubes in the crack initiation behaviour of candidate materials under SCW conditions (SS310S, 800H and AFA) at the project partners. Thus there is a gap in our knowledge regarding this issue. This "experimental issue" will be possibly solved during the project due to the WP2 partners may gain practical experience on the effect of the manufacture of tubes in the crack initiation behaviour of candidate materials.
- <u>Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature</u> – 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.

2.3.2.2 D2.1 - Test matrix based on available materials

The Deliverable 2.1 (D2.1) defines the possible candidate materials and the planned test matrix for material research for the project. According to the report the most important features of a cladding material are:

- the high neutron transparency;
- the good mechanical properties;
- the high toughness;
- and high resistance to creep and corrosion processes.

Supercritical water reactor environment means much higher temperatures beside the usual high radiation doses.

Conventional LWR cladding materials such as Zr alloys suffer from severe oxidation in SCW environment. Ni based alloys as candidate materials are excluded because of their low neutron transparency and embrittlement and swelling behaviour.

Stainless steel cladding materials show an optimal behaviour to degenerative processes like stress corrosion cracking (SCC) and do not suffer from swelling. However, austenitic SS are not ideal from the neutron transparency viewpoint.

The selected materials for further investigation were 310S and 800H stainless steels, characterized by high Cr content. A third option (Alumina Forming Alloy – AFA based on 310S stainless steel) is planned to be tested as well. There are still some knowledge gaps concerning their resistance against Irradiated Assisted Stress Corrosion Cracking (IASCC) and long-term behaviour under long-term oxidation. Another parameter that can affect the behaviour of the cladding material is the shaping of the material, for the tests tube shape has been selected where possible.

An open question for further investigation is the change of electrochemical behaviour of water around the critical point. It is well-known that the physicochemical properties of water (ionic product, dielectric constant and density) change sharply when the system crosses the critical point, which may also affect corrosion processes. There are signs of changing corrosion mechanism from electrochemical to chemical oxidation above the critical point. This change takes place at around 450ºC; at 380ºC the dissolution of materials is rather high.

The combination of neutron irradiation and oxidative environment needs also further evaluation as neutron radiation can modify not only the behaviour of the material but also the environment (with radiolysis process generating hydrogen, oxygen and hydrogen peroxide and free radicals, having big effect on corrosion).

A further challenge identified by D2.1 is the effect of the SCW inhomogeneity on the corrosion parameters as a consequence of large temperature and density changes at parts of the core.

The test matrix includes the following items for non-irradiated and irradiated specimens:

- Materials characterization and machining of specimens;
- Oxidation tests in SCW with non-irradiated and pre-irradiated materials;

- Slow Strain Rate Tensile (SSRT) Tests in SCW (only with non-irradiated materials);
- Radiolysis tests in SCW;
- Electrochemical measurements in SCW.

D2.1 also includes a detailed testing procedure for planning and performing measurements.

2.3.2.3 D2.2 - Study of the corrosion and EAC behaviour of selected alloys out-of-pile

WP2 Task 2.2 focuses on the basic pre-exposition characterization of the selected candidate materials.

The D2.2 describes the basic information regarding the candidate materials (310S, 800H and AFA). It also includes a cutting plan in order to prepare proper geometries of samples for further testing and the description of applied technology. The main part of the deliverable is the basic characterization of 310S and 800H materials as received in a tube and rod forms, including the description of their microstructure and mechanical properties.

The basic characterization contains the description of microstructure and results of selected materials properties. The microstructure was observed and analysed via light optical microscopy, scanning electron microscopy (SEM) and accompanied analyses and in the selected case via transmission electron microscopy (TEM). Among the mechanical testing, the uniaxial tensile tests, microhardness and nanoindentation were involved, and the surface roughness was characterized via measurement of roughness average (Ra) and SEM.

The results revealed that SS310S-tube possessed the most homogeneous microstructure with the presence of secondary phase enriched in Nb and Mo, this was confirmed by microhardness and nanoindentation analysis. In case of tensile tests, yield strength (Ys) and ultimate tensile strength (UTS) were slightly higher in comparison with that value stated in the material sheet.

In contrast, the microstructure of 800H-tube contained a mixture of coarse and finer grains with a higher presence close to the inner surface, confirmed by microhardness and nanoindentation analysis. The results of tensile tests of 800H-tube corresponded well with the values in the material sheet and they were lower in comparison with the parameters of SS310S-tube. The microstructure of both rods possessed residual deformation. The average grain size of the materials was determined based on Electron Back-Scatter Diffraction analysis (8 μm, 40 μm, 24 μm and 42 μm for 310S tube, 800H tube, 310S rod and 800H rod, respectively.) The secondary phase was present only in 800H-rod and identified as the nitrides and carbides of titanium (Ti). In both cases, the microstructure contained a mixture of coarse and smaller grains. The outer surface roughness was in the interval of 0.2 up to 0.8 μm in case of all materials.

The data resulting from WP2 Task 2.2 serve as a base for further experimental activities, however, no safety-related features of the planned SCW-SMR can be derived yet.

2.3.2.4 Main findings of WP3

According to [1], the main tasks of WP3 are the performance of design- and safety-related thermal-hydraulic investigations for the SCW-SMR. The main tasks are:

- Creation of an extended validation database;
- Model development and validation to assess turbulent heat and mass transfer and pressure drop along corroded surfaces;
- Development and validation of heat transfer correlations and models relevant for the SCW-SMR operation and safety;
- Implementation and validation of the developed correlations and models for applications in system codes;
- Improvement, implementation and validation of engineering models for turbulent heat transfer under supercritical conditions;
- Experimental, numerical and analytical investigations of key thermal hydraulic phenomena relevant for the SCW-SMR design concept;
- Safety and design analysis of the SCW-SMR concept;
- Derivation of European-Canadian-Chinese (ECC) joint design requirements for an ECC SCW-SMR design concept.

Tasks of WP3 are:

- Task 3.1: Conceptual design requirements of SCW-SMR;
- Task 3.2: Reference database;
- Task 3.3: Development and improvement of system-, subchannel- and CFD-codes for SCW-SMR;
- Task 3.4: Study of a pre-conceptual core layout and passive safety concept for SCW-SMR.

Based on *"Safety-related findings and conclusions live Excel file"*, WP3 listed the following important issues as thermal-hydraulics related and at least partly solved:

- Availability of practically useable models, methods and correlations for turbulent heat transfer at supercritical conditions;
- Modelling of the Deteriorated heat transfer (DHT) - The phenomenon of DHT is not fully known thus the possible occurrence in the core of SCW-SMR cannot be predicted in a trustable manner.

One issue has been listed as safety-related knowledge gap:

- Formulation of design and safety concept - 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;
2. For SCW-SMR design and software development an iterative procedure between WP3, WP4 and WP2 is successfully established;
3. Reference database has been set up and is available online;
4. System-, subchannel- and CFD-codes benchmarking activities are defined and are in progress;
5. First system thermal hydraulics analysis is performed for analysis of core layout and passive safety concept.

In order to collect the safety-related features, knowledge gaps and outstanding issues of SCW-SMR, PIRT analysis has been conducted for WP3 experts, see Chapter 2.3.3.2.

2.3.2.5 Main findings of WP4

The main objective of WP4 is the study of design- and safety-related neutronic parameters and reactor physics behaviour of the SCW-SMR conceptual design. In this work package, the selection of neutronics codes and parameters; preliminary core calculations concerning the safety-related parameters (such as reactivity coefficients) and first burn-up calculations are performed.

In D4.1 the results of the comparison of various neutron physics codes and cross-section sets based on the Gen-IV SCWR-FQT reactor physics computational benchmark are presented. In the study three Monte Carlo codes have been evaluated: MCNP6, OpenMC and Serpent 2. The calculated parameters in the benchmark project were: effective multiplication factor, axial power distribution, axial three-group neutron flux distribution in the water channels of the FQT assembly and energy deposition.

During the evaluation of preliminary results of the benchmark, results, significant discrepancies have been found mainly because of preliminary inconsistencies in the input files prepared by the participants, showing the need for a finalized and comprehensive, preliminary tested benchmark specification.

An important experience of the benchmark is, that – although the global parameters such as the effective multiplication factor – shows good agreement among the applied codes, achieving the agreement for neutron flux distribution is much more complicated. The benchmark project revealed the main differences among the applied Monte Carlo codes (e.g. different normalization), which should be considered in future works.

After a thorough model revision, the calculations showed a good agreement, but some lessons could be learned from the analysis of the results. For example: concerning the effective multiplication factors, the Serpent 2 code shows a larger deviation from the others, which may be caused by the specific nuclear data used by the code. Some systematic deviation has been observed during the comparison of axial volume-integrated neutron flux distributions, which shows the importance of the treatment of thermal neutron scattering.

Different models of the applied codes have been investigated, such as the energy deposition modes available in Serpent 2, or the effect of using different nuclear data libraries.

The most important conclusion of the analyses is that based on the experiences, the investigated <u>Monte Carlo codes can be applied for SCW-SMR modelling purposes, but their special features must be considered</u>. "MCNP6, being thoroughly verified and validated, is generally applied to provide reference Monte Carlo transport results, while the available broad range of input settings and the efficient burn-up calculations make Serpent 2 or OpenMC more practical for design calculations."

In D4.2 the reactivity coefficients calculated for normal operational conditions are demonstrated. For the analysis Serpent 2 Monte Carlo code has been applied.

The special feature of SCW reactors is the enormous change in coolant density around the pseudocritical point, having potentially a strong effect on neutronics of the reactor. In this work reactivity feedback coefficients – moderator temperature coefficient, coolant temperature coefficient, Doppler coefficient – were calculated both globally (reactor-wise) and locally (heat-up stage-wise). The main factors affecting reactivity in case of a power excursion are the broadening of fuel resonances (Doppler-effect), neutron spectrum hardening, and water density changes. Above these, the thermal expansion of structural elements can affect reactivity through the changing probability of nuclear reactions.

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

From the aspect of reactivity coefficients, the SCW-SMR design shares the features of pressure vessel and pressure tube type reactors as well – although the same medium is applied as moderator and coolant at same pressure, they are separated from each other in the vicinity of the fuel assemblies. This makes it obvious, that nor the pressure vessel, neither pressure tube features could be applied without further consideration.

According to D4.2 further model improvements, such as the definition of a fixed core arrangement, fuel enrichments and cladding material, should be carried out for additional reactivity coefficient calculations. Furthermore, the effect of reactivity feedbacks should also be investigated in cases different from normal operation as the importance of reactivity feedback is more significant in transient and incident scenarios.

The Work Package Periodic Report of WP4 gives an overview about the progress of the WP4 in the first 18 month of the project. Above the aforementioned results published in D4.1 and D4.2, the Periodic Report lists the following main results:

- Detailed MCNP and Serpent MC models of the SCW-SMR design concept suggested by WP3 have been developed;
- Extensive core design work has been started, with special focus on how the design targets of the concept can be achieved in terms of neutron physics. These targets are:
    - At least 2-year-long burnup cycle;
    - Use of stainless-steel material instead of zirconium alloys;
    - Power density distribution among the seven stages (in vertical direction) according to the thermal-hydraulics requirements formulated by WP3;
    - The least possible power peaking in horizontal direction (among fuel assemblies of a single row);
    - The least possible power peaking in horizontal direction along fuel assemblies;
    - The least power peaking ratios for fuel pins inside a fuel assembly.
- A large number of Monte Carlo simulations have been performed in order to find some feasible (in terms of neutronics) reactor core configurations;
- Several quantities, such as effective multiplication factor and reactivity feedback parameters have been determined for each of the investigated configurations.

According to the Periodic Report, based on the fuel cycle calculations, a proper reactivity reserve can only be achieved with higher uranium enrichments, and the usage of MOX fuel besides UO_2 is essential for longer cycles. "*The necessity of the application of higher* *enrichments may be reduced significantly only if either the target profile can be altered (linear heat rates can be increased on the higher stages) or if neutronically more compatible structural materials (or coatings like zirconium silicide for example) would be applied.*"

The power profile can be shaped with the implementation of a mixed fuel core and for the linear power profile the best results can be achieved by introducing several differently enriched assemblies (some with enrichments up to 8%), but further optimization and coupled neutronics-thermal-hydraulics calculations are needed. However, the application of mixed fuel core can raise feasibility concerns and requires further evaluation.

According to the report, two main criteria have been set up from the aspect of neutron physics:

a) the necessity of a sufficiently high beginning of cycle (BoC) effective multiplication factor (so that the end of cycle (EoC) operability can be ensured);
b) the decrease of peaking factor in the horizontal direction in between assemblies and alongside them.

From the viewpoint of moderation optimization, the report recommends to use a wider fuel assembly gap and a lower moderator temperature in order to increase BoC multiplication factor. Further solutions can be the implementation of different horizontal fuel assembly gaps or to consider alternative moderator box shapes.

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.

2.3.3 PIRT analysis of safety related knowledge gaps

As mentioned previously, although the work packages have progressed significantly since the start of the project, their work has not been finished yet. As the task of our group 5.2 is to derive the safety-related features of SCW-SMR based on the results or WP2-4, on the top of the evaluation of the available results of the working groups (see above), a general PIRT analysis has been initiated in order to collect the common knowledge of the WP members.

The results of this PIRT analysis are described in the following chapters.

2.3.3.1 WP2 - PIRT analysis of safety related knowledge gaps

As WP2 experimental investigation of selected potential cladding materials is still under way, the participants of WP2 have been invited to give an expert evaluation of possible significant safety features of SCW-SMR in connection with material issues.

For the members of WP2 an invitation (with a short description of PIRT methodology, see Annex 1) and a fill-in Excel worksheet has been sent out, with the following issues:

- Descriptive name of material phenomena;
- Short description of the item or notes;
- Reference for the item;
- Phenomena Ranking (PR);

- Rationale of PR by the proposer of the item;
- Knowledge Ranking (KR);
- Rationale of KR by the proposer of the item;
- Operating status.

From the participants – in more iterating rounds – altogether 7 expert opinions were received from 6 institutions. Originally 22 different phenomena have been identified, however, during the ranking process the original "oxide release from cladding" phenomenon (No. 3.) has been split into two based on the release process (by dissolution/evaporation and by spalling). Participants gave also a short description of the given phenomenon, together with some literature reference. The explanation for ranking has been also asked for, but the first explanation filled in has not been changed by later contributors.

Figure 12 shows a part of the PIRT table filled by WP2 participants, the whole table can be found in Annex 2.

#	Descriptive name of material phenomena:	Short description of the item or notes:	Phenomena Ranking (PR):	Rational of PR by the proposer of the item:	Comments/Reasons for Selected Ranking	Knowledge Ranking (KR):	Rational of KR by the proposer of the item:	Reasons for Selected Ranking	Operating status:
1	**Through wall penetrations produced by general corrosion**	The resistance of cladding matererial is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases	Phenomena Ranking Scales	Takes place more or less frequently in present LWRs. Results in increased radiation exposure of personnel in the vicinity of the reactor coolant circuit.Should be observed easily from the first cracks.	Please specify!	Knowledge Based Ranking Scales	Only knowledge is based on lab tests on small samples.	Please specify!	All operations
2	**Oxide build-up that impedes heat transfer**	The oxide thickness could affect the heat transfer and the efficience of the reactor	Phenomena Ranking Scales	May result in overheating of the fuel and cladding. Probably most pronounced where the temperature is close to Tc.	Please specify!	Knowledge Based Ranking Scales	Limited knowledge on the effect on heat transfer and cladding integrity. Possibility of accumulation depends on many unknown factors, e.g., nominal flow condition and properties of formed deposits/depositing oxide particles.	Please specify!	Normal operation
3	**Oxide release from the cladding surface by dissolution/evaporation**	The release of irradiated oxides can contaminate other reactor areas	Phenomena Ranking Scales	Results in increased radiation exposure of personnel in the vicinity of the reactor coolant circuit.	Please specify!	Knowledge Based Ranking Scales	Only knowledge is based on lab tests on small samples.	Please specify!	Normal operation

Figure 12: PIRT table filled by WP2 members

The 23 identified phenomena and their short explanation are the following:

1. **Through wall penetrations produced by general or localized corrosion** - The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases;
2. **Oxide build-up that impedes heat transfer** - The oxide thickness could affect the heat transfer and the efficiency of the reactor;
3. **Oxide release from the cladding surface** - The release of irradiated oxides can contaminate other reactor areas;

3a. Oxide release from the cladding surface by dissolution / evaporation - The release of irradiated oxides can contaminate other reactor areas;

4. **Pellet cladding interaction** - This interaction could produce hot-spots and harm the material resistance;
5. **Environmental Assisted cracking (EAC)** - Cracking of the cladding tube;
6. **Changes in the mechanical properties of the materials produced by ageing and/or irradiation** - The cladding must retain the mechanical properties and critical dimensions to contain the fuel and fission gases;
7. **Changes in the geometry of tubes produced by irradiation, creep** - The cladding must retain its mechanical properties and critical dimensions to contain the fuel and fission gases;

8. **Radiolysis processes** - Production of oxygen by radiolysis must be studied in order to reduce the corrosion of cladding and structural components;
9. **Physicochemical properties of water within the SC region** - Close to the critical and pseudocritical point the dissolution of materials is promoted. The corrosion behaviour of materials at these points must be studied;
10. **Resistance of cladding materials under LOCA conditions in SCWRs** - The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases;
11. **Impurity enrichment** - Enrichment of electrolytes (e.g., chlorides) in oxide deposits. This ties to the purification system during normal operation and transients;
12. **Oxide release from the cladding surface by spalling** - Transport of radioactivity to out-of-core surfaces. Damage to downstream corrosion by erosion. Blockage of piping or fuel assembly sub-channels;
13. **Irradiation embrittlement due to He** - He produced by transmutation reaction of Ni;
14. **IASCC** - Irradiation-Assisted Stress Corrosion Cracking;
15. **Hydriding** - Internal hydriding due to fabrication. External hydriding due to waterside corrosion;
16. **Cladding collapse** - If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential would exist for the cladding to collapse into a gap;
17. **Overheating of the Cladding** - fuel failure due to overheating;
18. **Overheating of Fuel Pellets** - fuel failure due to overheating;
19. **Cladding rupture** - Cladding rupture due to fast heating rate;
20. **Fuel Rod Mechanical Fracturing** - It refers to a fuel rod defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion;
21. **Strain Fatigue** - Fuel rod failure due to fatigue or corrosion fatigue;
22. **Fretting Wear** - It can occur on the fuel rod cladding surfaces in contact with the spacer grids.

It is worth to mention here that the identified phenomena are valid for different plant conditions, some of them apply only for LOCA condition (as it is mentioned in name of item No 10.), most of them apply for all conditions. The exact classification can be seen in Annex 2. The key parameter of cladding issues – the aspect of which has been taken into consideration – is the leaktightness of the cladding (i.e. cladding damage).

The expanding list of identified significant phenomena has been sent back to experts in more rounds for ranking of the newly emerged phenomena. This proved to be quite an inefficient solution, as multiple ranking process was not expectable from participants[10].

In order to keep the anonymity of the experts answering the invitation for ranking, the code names [Expert A] … [Expert] G are used. However, the list of the 6 cooperating institutions (in alphabetical order, not corresponding the expert code names) is the following:

- CIEMAT;
- CNL;
- CVR;

[10] Based on the lessons learned from this ranking process, for the other WPs a new approach was applied: first the common list of phenomena has been prepared on the Sharepoint surface of the project, and only after finalization of the phenomena list were the experts asked for ranking.

- JRC;
- RATEN;
- VTT.

Figure 13 shows the result of the expert ranking (detailed results and statistics on the results can be found in Annex 2). The figure shows the average importance level (IL) and knowledge level (KL) for each phenomenon identified by WP2 experts. The direction of the arrow shows the increase of significance (IL and KL) of the phenomena[11].

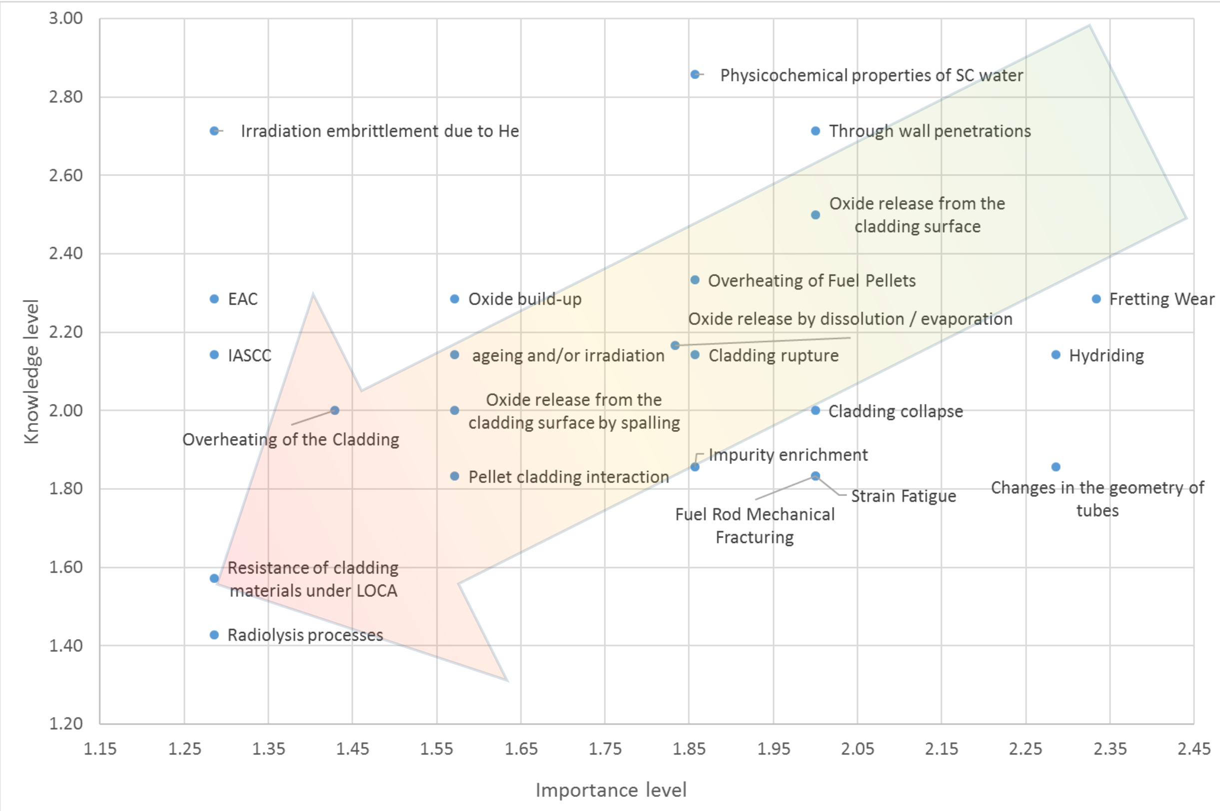

Figure 13: Ranking of identified phenomena in WP2

According to the results, some of the most important phenomena have also quite low knowledge level, revealing the necessity of further analysis / experimental evaluation. The most important issues with low knowledge level can be seen in the left lower part of the diagram.

Similarly to the method applied in ELSMOR project described in Chapter 2.1.1.5, the statistical evaluation of the results has been performed as well.

Concerning the deviation of the results, the average of the deviation of importance level (σIL) is 0.73; while the deviation of knowledge level (σKL) is 0.67. The highest deviation results for

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

phenomena with relative small number of votes (such as oxide release issue, which later has been divided into 2 separate effects). Otherwise usually good agreement could be observed among the experts' answers.

From the IL and KL values the relative relevance (RR) of a given i phenomenon has been calculated. First, the normalized value of IL and KL (nIL and nKL) has been calculated. From these normalized values (between 0 and 1 each), the RR was calculated as:

$$RR_i = \frac{(1 - nIL_i) * (1 - nKL_i)}{max_i((1 - nIL) * (1 - nKL))}$$

The relative dispersion (RD) has been calculated as well from the data as follows:

$$RD_i = \frac{\sigma IL_i * \sigma KL_i}{max_i(\sigma IL * \sigma KL)}$$

Based on this definition, the most relevant phenomena (having the highest relative relevance, which includes the importance level and knowledge level as well), can be seen together with their relative dispersion values in Table 3.

Table 3 shows that the most relevant phenomena are partly covered by the present experimental program of ECC-SMART WP2, however, there are some additional material-related issues, which need further examination, such as LOCA behaviour of cladding, or overheating of cladding material.

Phenomenon	Relative relevance	Relative dispersion
Radiolysis processes	1.00	0.40
Resistance of cladding materials under LOCA	0.90	0.27
Pellet cladding interaction	0.52	0.15
Overheating of the cladding	0.52	0.43
IASCC	0.50	0.29
Oxide release from the cladding surface by spalling	0.43	0.00
EAC	0.40	0.25
Mat. prop. changes due to ageing and/or irradiation	0.36	0.20
Impurity enrichment	0.32	0.42
Oxide build-up	0.29	0.18
Oxide release by dissolution / evaporation	0.23	0.21
Cladding rupture	0.23	0.23
Fuel Rod Mechanical Fracturing	0.23	0.45
Strain Fatigue	0.23	0,36
Cladding collapse	0.19	0.45
Overheating of Fuel Pellets	0.17	0.48
Irradiation embrittlement due to He	0.10	0.16
Oxide release from the cladding surface	0.08	1.00
Through wall penetrations	0.03	0.27
Changes in the geometry of tubes	0.03	0.19
Hydriding	0.02	0.29
Physicochemical properties of SC water	0.00	0.65
Fretting Wear	0.00	0.39

Table 3: Most relevant phenomena by expert ranking for WP2

2.3.3.2 WP3 - PIRT analysis of safety related knowledge gaps

The tasks of WP3, i.e. numerical and experimental investigation of heat transfer issues and passive core coolability are also in progress, so it is challenging to draw general safety-related issues for WP5 Task 5.2. In order to get a broader view about thermal hydraulics related safety issues, the participants of WP3 have been invited to give an expert evaluation of possible significant safety features of SCW-SMR considering thermal-hydraulic behaviour.

For the members of WP3 an invitation (with a short description of PIRT methodology, see Annex 1) and a web link to a fill-in Excel worksheet on SharePoint has been sent out, with the following list of information[12]:

- Descriptive name of TH phenomena;
- Short description of the item or notes;
- Rationale of the item;
- Reference for the item;
- Figure of merit;
- Operating status.

[12] The method of information collection has been changed as described in previous chapter.

From the participants altogether 6 expert opinions were received from 6 institutions, and 21 different phenomena have been identified. (The issue „decay heat“ has been added later to the phenomena list, but it was excluded from the analysis because the ranking had been already done by that time.)

The list of the 21 relevant thermal-hydraulics related safety issues (the detailed list with short explanation of the issues and with literature references can be found in Annex 3):

- Steep non-linear change of SCW fluid material properties;
- Heat transfer in water under supercritical pressure conditions (trusted prediction of heat transfer coefficient);
- Pressure drop (Δp) in water under supercritical pressure conditions in SCW-SMR (SCWR) relevant geometries;
- Turbulent heat and mass transfer in water under supercritical pressure conditions in SCW-SMR relevant (in horizontally installed tube but mainly rod bundle) geometries;
- Heat and mass transfer of coolant/moderator SCW along corroded and rough surfaces of the core structural element;
- Deterioration of heat transfer (DHT) or heat transfer deterioration (HTD);
- Transition from supercritical to subcritical pressure water state;
- Steam and liquid water two-phase flow in SCW-SMR primary loop;
- Natural circulation of water under super- or sub-critical pressure conditions in SCW-SMR primary loop;
- Strong coupling between the thermal hydraulics (e.g. SCW density field) and the reactor physics (e.g. heat production by nuclear fission in the core);
- Depressurisation of the primary loop and the travelling depressurisation wave and its path in case of a LOCA;
- The effect of the presence of large and hot structural components (large amount of structural material) in the reactor pressure vessel (RPV) during accidents;
- Flow instability under supercritical pressure conditions in SCW-SMR relevant (in horizontally installed tube but mainly rod bundle) geometries;
- Allowable maximum cladding temperature during normal operation (WP2-WP3) in the SCW-SMR;
- Flow stratification in horizontal channels, the lighter fluid will tend to move upwards in the fuel assembly;
- Flooding;
- Thermal hydraulics and neutronic coupled instabilities;
- Critical heat flux (CHF) near the critical point;
- Fluid induced vibration;
- (Decay heat);
- Mechanical deformation (coupled with thermal hydraulics);
- Pellet/cladding interaction (conductance).

Most of the identified phenomena are considered in all operational conditions, however, some of them are specifically defined for LOCA accidents. The figure of merit has been selected as the fuel cladding temperature. After the finalization of the phenomena list, the ranking table in a fill-in Excel worksheet was sent out to the participants. Most phenomena has been ranked by 5 participants, for some issues we got 6 expert votes (called [Expert A] ...[Expert F] in the worksheets).

The results of the expert ranking can be seen in *Figure 14*.

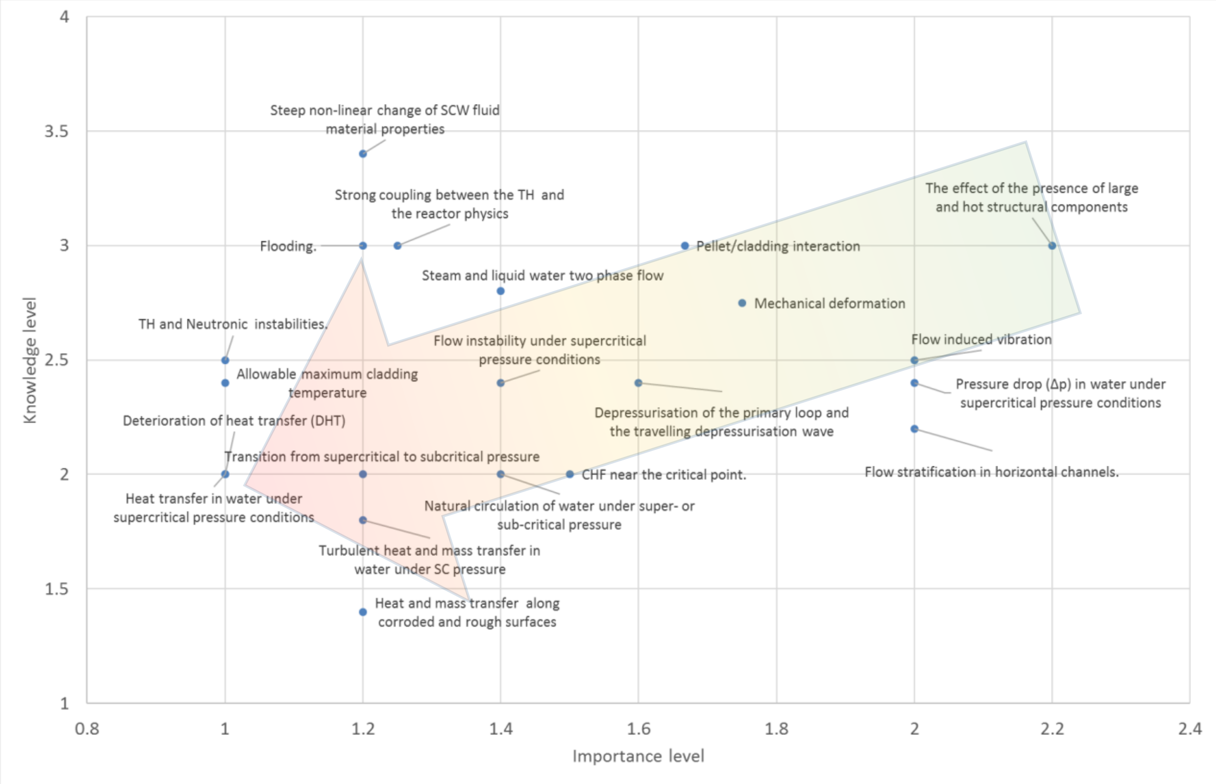

Figure 14: Ranking of identified phenomena in WP3

Similarly to WP2 PIRT analysis, the statistical evaluation of the results has been performed as well. Concerning the deviation of the results, the average of the deviation of importance level (σIL) is 0.39; while the deviation of knowledge level (σKL) is 0.52. Usually good agreement could be observed among the experts' answers, there are even some phenomena with 0 deviation.

From the IL and KL values the relative relevance (RR) of a given i phenomenon has been calculated, as shown in case of WP2. Based on this definition, the most relevant phenomena (having the highest relative relevance, which includes the importance level and knowledge level as well), can be seen together with their relative dispersion values in Table 4.

Table 4 shows that the most relevant phenomena are mostly covered by the present experimental program of ECC-SMART WP3, however, among the lower relevance phenomena there are additional issues to be investigated. Because of the small number of expert votes the relative dispersion is 0 in few cases, where all the expert rankings were the same for a phenomenon.

Phenomenon	Relative relevance	Relative dispersion
Heat and mass transfer along corroded and rough surfaces	1.00	0.42
Heat transfer in water under supercritical pressure conditions	0.84	0.00
Deterioration of heat transfer (DHT)	0.84	0.00
Turbulent heat and mass transfer in water under supercritical pressure conditions	0.80	0.34
Transition from supercritical to subcritical pressure	0.70	0.54
Allowable maximum cladding temperature	0.60	0.00
Natural circulation of water under super- or sub-critical pressure conditions	0.56	0.00
TH and Neutronic instabilities	0.54	0.00
CHF near the critical point	0.49	0.59
Flow instability under supercritical pressure conditions	0.40	0.51
Depressurisation of the primary loop and the travelling depressurisation wave	0.30	0.51
Steam and liquid water two phase flow	0.24	0.42
Flooding	0.20	0.54
Strong coupling between the thermal hydraulics and the reactor physics	0.19	0.69
Mechanical deformation	0.15	0.81
Flow stratification in horizontal channels	0.12	0.54
Pellet/cladding interaction	0.11	0.00
Pressure drop (Δp) in water under supercritical pressure conditions	0.10	0.00
Flow induced vibration	0.09	0.00
Steep non-linear change of SCW fluid material properties	0.00	0.42
The effect of the presence of large and hot structural components	0.00	1.00

Table 4: Most relevant phenomena by expert ranking for WP3

2.3.3.3 WP4 - PIRT analysis of safety related knowledge gaps

In order to get a general overview about safety-related reactor-physical features of SCW-SMR, the survey for collecting relevant phenomena has been sent to the members of WP4 as well. In the first round a set of phenomena identified by reactor physics experts of BME NTI has been disseminated for WP4 members. BME NTI experts identified 22 relevant phenomena for general plant conditions (Normal Operation / All Conditions) and 8 phenomena for control rod ejection transients (REA – Rod Ejection Accidents).

Unfortunately, no answer has been received for our announcement, so the list of original 22+8 phenomena has been circulated for expert ranking. For this second round only one contribution has been submitted (by UNIPI) on the top of the ranking of BME NTI experts. Having only these 2 expert rankings, it is not feasible to perform a statistical analysis of the results, but some general lessons can be drawn. The detailed list of identified phenomena can be observed in *Annex 4*.

Concerning the general phenomena, a further distinction can be made. The first group of identified phenomena is connected to the computation methodology of SCW-SMR neutronics. These include issues concerning deterministic and Monte Carlo calculations. Figures-of-merit can be the ability to determine the effective multiplication factor (FoM1) or neutron flux distribution (FoM2). Some relevant phenomena – based on the two expert ranking – are the determination of time discretization for deterministic calculations and the boundary conditions and group constants for the reflector. For Monte Carlo calculations the accurate material properties seem to be the most important issue with high influence on FoM and with limited knowledge. For less extent geometric parameters and material composition are also relevant.

Material and neutron physics related data were considered as a separate set of phenomena by the experts ranked as High or Medium for importance. Some of these issues have limited knowledge as well, playing a significant role in simulation uncertainties. Such less known parameters are the material composition of control rods and structural material temperatures and densities and parameters of burnable poison materials.

The third group of relevant phenomena are the design related data. According to the experts, there are significant uncertainties concerning the design of reactivity control systems and in-core detectors, which have also a High / Medium importance ranking, so they can influence the neutronics calculations.

For REA transients Figures-of-Merit were power history during REA transient (FoM1) and pin fuel enthalpy during REA transient (FoM2). FoM2 includes the peak cladding temperature for which an acceptance limit is defined. In this phenomenon group, some high-importance issues have been identified with very limited knowledge. These are the rod worth of the ejected control rod and the rate of reactivity insertion during the accident. The common cause of these phenomena is the lack of control rod system design.

3 Safety features of SCW-SMR

In this report, the available safety related features of SCW-SMR has been collected. Based on the literature review made by WP5 Task 5.1 and 5.2, the specific features of super-critical water cooled reactors [4][5][6], especially the HPLWR design [10]; special issues of BWR reactors and SMR-related features have been collected [11][12]. According to this literature review the most important safety-related features are the following:

<u>Safety-related features derived from SCWR technology</u> – challenges (for further details see Chapter 2.2.1):

- application of novel manufacturing processes;
- aggressive chemical effects of SCW in the reactor core;
- SCW has not been used in highly radiative environment;

- correctness of assumptions and extensions do require confirmation;
- 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;
- material issues identified (specific issues for the investigated reactor designs).

<u>Safety-related features derived from HPLWR technology</u> – challenges (for further details see Chapter 2.2.1):

- core power distribution is heavily influenced by the coolant density distribution through the neutron moderation parameters;
- another important issue is the problem of coolant and moderator flow stability;
- possible Xenon oscillation instabilities;
- possible larger stresses because of the high temperature difference;
- high peak cladding temperature, low fuel burn-up and high hot channel factors require further analyses;
- large uncertainties of heat transfer predictions;
- concerning the safety systems, they are quite similar to BWRs but „in case of a SCWR the residual heat can be removed only by forced convection inside the RPV”.

<u>Safety-related features derived from BWR technology</u> – challenges (for further details see Chapter 2.2.2):

- recirculation pump (not relevant for SCWRs);
- large changes in neutron flux, coolant density along the FAs;
- one-circuit design →radioactive steam arrives to turbines (less physical barriers available, no leaktight primary circuit) → steam line isolation valves are essential;
- Special containment arrangement (drywell / wetwell).

<u>SMR special issues</u> (for further details see Chapter 2.2.3):

- safety features of large reactors are mainly applicable, but more extensive use of passive safety systems;
- 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);
- differences in reactor physics parameters (higher enrichment, new fuel types, etc.).

For the evaluation of the WP work results, the review of WP deliverables was performed, complemented with WP experts’ opinion and with the results of the performed PIRT analysis.

<u>Safety-related features based on the results and expert opinion of WP2 (for further details see 2.3.2.1 and 2.3.3.1):</u>

Most important issues by expert opinion:

- Oxidation behaviour at long time exposures;
- Effect of irradiation on cladding materials;
- Effect of the manufacture of tubes/rods in the crack initiation behaviour of candidate materials;
- Effect of radiolysis in SCW and changes in electrochemistry with pressure and temperature.

<u>Safety-related features based on WP2 experiments:</u>

Most significant phenomena based on the results of PIRT analysis by WP2 members:

- Radiolysis processes;
- Resistance of cladding materials under LOCA;
- Pellet cladding interaction;
- Overheating of the cladding;
- Irradiation Assisted Stress Corrosion Cracking (IASCC).

Safety-related features based on the results and expert opinion of WP3 (for further details see 2.3.2.4 and 2.3.3.2):

Most important issues by expert opinion:

- Party solved issues are: models, methods and correlations for turbulent heat transfer at SC conditions; modelling of deterioration of heat transfer (DHT).
- Knowledge gap exists for: formulation of design and safety concept.

Safety-related features based on WP3 results:

Most significant phenomena based on the results of PIRT analysis by WP3 members:

- Heat and mass transfer along corroded and rough surfaces;
- Heat transfer in water under supercritical pressure conditions;
- Deterioration of heat transfer (DHT);
- Turbulent heat and mass transfer in water under supercritical pressure conditions;
- Transition from supercritical to subcritical pressure.

Safety-related features based on the results and expert opinion of WP4 (for further details see 2.3.2.5 and 2.3.3.3):

Safety-related features based on WP4 results:

- all temperature reactivity coefficients are negative, but differencies in the different stages of coolant flow;
- uncertainties related to ensuring the reactivity reserve (use of HA-LEU or MOX fuel);
- further problems to be solved:
  - shaping of power profile (large number of different FA would be necessary);
  - moderation optimization (wider FA gap and lower moderator temperature or design modification);
  - set of refuelling strategy.

Most significant phenomena based on the results of PIRT analysis by WP4 members:

- 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.

4 Summary

This deliverable outlines the safety-related features of SCW-SMR design based on the review of the available literature, the conclusions available at the time of the preparation of this report from technical work packages and the expert opinion of WP participants.

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 above mentioned conclusions are depicted in *Figure 15*.

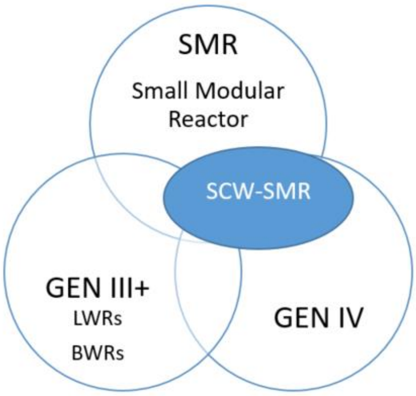

Figure 15: SCW-SMR technology compared to existing categories

The main goal of our work was to identify knowledge gaps (WP2, WP3), simulation gaps (WP4), and weak points of SCW-SMR design. For this purpose, an extensive literature review has been performed to identify the special safety features of supercritical water cooled reactors, boiling water reactors and small modular reactors.

Due to the fact that the experimental and simulation work of working packages 2, 3, 4 is still in progress, on the top of the review of the available reports of WPs, the collection of expert opinion has been performed in a systematic way, applying the widely used PIRT methodology.

The report presents the result of this work.

5 References

[1] ECC-SMART Annex 1., Description of Action

[2] Safety Design Criteria for Generation IV Sodium-cooled Fast Reactor System (Rev. 1) (2017)

[3] Safety Design Guidelines on Safety Approach and Design Conditions for Generation IV Sodium-cooled Fast Reactor Systems (Rev. 1) (2019)

[4] An Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems (Version 1.1) (2011)

[5] T. Nitheanandan, T. Schulenberg and L. Leung, Super-Critical Water-cooled Reactors (SCWR) Risk and Safety Assessment White Paper (Rev. 1), January 2017.

[6] T. Nitheanandan, T. Schulenberg and L. Leung, Supercritical-water-cooled reactor system (SCWR) System Safety Assessment, 2018.

[7] David J. Diamond: Experience Using Phenomena Identification and Ranking Technique (PIRT) for Nuclear Analysis, Brookhaven National Laboratory, 2006, BNL-76750-2006-CP

[8] ELSMOR – Towards European Licensing of Small Modular Reactors, Elaboration of simplified PIRT transients for SMR PWR, WP 3: Safety Core Cooling, WP 3.1 – PIRT Elaboration, 2020

[9] ELSMOR – Towards European Licensing of Small Modular Reactors, Improved Safety Analysis Methods and Tools for Containment Safety, WP 4: PIRT for heat removal from the containment with passive safety features, 2020

[10] T. Schulenberg, J. Starflinger (eds.): High Performance Light Water Reactor - Design and Analyses, KIT Scientific Publishing, 2012

[11] GE Hitachi Nuclear Energy – BWRX-300 General Description, 005N9751, Revision E, August 2023

[12] IAEA-TECDOC-1785, Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident, IAEA, Vienna, 2016

[13] Initial List of Requirements for a Supercritical Water-Cooled Small Modular Reactor, CNL, 217-120200-REPT-002, 2022

[14] European Commission, Guidance Document for Integrated Safety Assessment Methodology (ISAM) – (GDI), EC JRC report prepared for GIF Risk and Safety Working Group, 2014.

[15] Deliverable D5.1 Safety criteria and requirements for the SCW-SMR concept, JSI (leader), 10/2/2022

Annex 1 – Short description of PIRT methodology sent to the WP2-4 members

PIRT analysis of SCW-SMR reactor

PIRT (Phenomena Identification and Ranking Table) is a qualitative analysis tool in order to identify knowledge / analysis / validation gaps for a given technology. PIRT has been applied in the preliminary safety evaluation of different Gen IV reactor designs. A clear advantage of this method is that it can be applied at all design stages, even in our pre-design phase.

The analysis is based on the ranking of significance / knowledge level, performed by an expert panel.

The basic logic of the PIRT analysis is the following:

- **Step 1. Experts of a given field make a list of the identified phenomena. (These are design-specific or general problems, representing a challenge to the safety of the reactor.)**
- Step 2. All phenomena are ranked by further experts according to their significance and the knowledge level of the problem. The ranking is made based on a pre-determined scale, see Fig. A1-1. below.
- Step 3. The rankings of a given phenomenon are averaged, so finally there are two numbers describing the importance and the knowledge gaps concerning the problem.
- Step 4. The result can be seen in a PIRT table or matrix, based on which the most problematic phenomena can be identified, see red cells in Fig. A1-2. below.

In our process, experts of WPs evaluate these gaps, based on their prior knowledge on the topic or based on experiences gained during the recent work in ECC SMART.

With their help a preliminary phenomenon list has been prepared, we need your expert opinion for the ranking of these phenomena.

Phenomena Ranking Scales	Knowledge Based Ranking Scales
High (H)	4 - Fully known, small uncertainty
Medium (M)	3 - Known, moderate uncertainty
Low (L)	2 - Partially known, large uncertainty
Insignificant (I)	1 - Very limited knowledge, uncertainty can not be characterized

Figure A1-1: Typical phenomena ranking scales, applied for our analysis

Knowledge level / Importance	1	2	3	4
Insignificant				
Low				
Medium				
High				

Figure A1-2: PIRT matrix[13]

Colour codes: low importance, medium importance, high importance

[13] Note: In some versions of the description the colour code of PIRT matrix has been set erroneously.

Annex 2 – PIRT analysis results, WP2

#	Descriptive name of material phenomena:	Short description of the item or notes:	Reference for the item:	Phenomena Ranking (PR):	Rationale of PR by the proposer of the item:	Knowledge Ranking (KR):	Rationale of KR by the proposer of the item:	Operating status:
1	**Through wall penetrations produced by general corrosion**	The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases	Materials and Water Chemistry for Supercritical Water-cooled Reactors; Guzonas et al; Woodhead Publishing Series in Energy 2017; Irradiation Effects in cladding and structural materials; ASM 1965 and others	Phenomena Ranking Scales	Takes place more or less frequently in present LWRs. Results in increased radiation exposure of personnel in the vicinity of the reactor coolant circuit. Should be observed easily from the first cracks.	Knowledge Based Ranking Scales	Only knowledge is based on lab tests on small samples.	All operations
2	**Oxide build-up that impedes heat transfer**	The oxide thickness could affect the heat transfer and the efficiency of the reactor	Materials and Water Chemistry for Supercritical Water-cooled Reactors; Guzonas et al; Woodhead Publishing Series in Energy 2017; Irradiation Effects in cladding and structural materials; ASM 1965 and others	Phenomena Ranking Scales	May result in overheating of the fuel and cladding. Probably most pronounced where the temperature is close to critical temperature.	Knowledge Based Ranking Scales	Limited knowledge on the effect on heat transfer and cladding integrity. Possibility of accumulation depends on many unknown factors, e.g., nominal flow condition and properties of formed deposits/depositing oxide particles.	Normal operation
3	**Oxide release from the cladding surface by dissolution/evaporation**	The release of irradiated oxides can contaminate other reactor areas	Materials and Water Chemistry for Supercritical Water-cooled Reactors; Guzonas et al; Woodhead Publishing Series in Energy 2017; Irradiation Effects in cladding and structural materials; ASM 1965 and others	Phenomena Ranking Scales	Results in increased radiation exposure of personnel in the vicinity of the reactor coolant circuit.	Knowledge Based Ranking Scales	Only knowledge is based on lab tests on small samples.	Normal operation
4	**Pellet cladding interaction**	This interaction could produce hot-spots and harm the material resistance	Pellet-clad Interaction in Water Reactor Fuels (OECD/NEA)ISBN 92-64-01157-9 - No. 54167 2005; Progress on Pellet–Cladding Interaction and Stress Corrosion Cracking IAEA TEC-DOC ISBN 978-92-0-116421-6	Phenomena Ranking Scales	May result in cladding deterioration.	Knowledge Based Ranking Scales	Pellet - stainless steel cladding interaction in gen I/II LWRs is known, but is uncertain at SCWR temperatures	Normal operation
5	**Environmental Assisted cracking (EAC)**	Cracking of the cladding tube	Progress on Pellet–Cladding Interaction and Stress Corrosion Cracking	Phenomena Ranking Scales	Results in increased radiation exposure of personnel in the vicinity of the reactor coolant circuit. Should be observed easily from the first cracks.	Knowledge Based Ranking Scales	Lab tests indicate that there is a heat to heat differences in SCC susceptibility which can be taken into account, mostly.	Normal operation
6	**Changes in the mechanical properties of the materials produced by ageing and/or irradiation**	The cladding must retain the mechanical properties and critical dimensions to contain the fuel and fission gases	K. Fukumoto et al., Journal of Nuclear Science and Technology. 57, issue 1, 2020.	Phenomena Ranking Scales	May result in larger scale fuel cladding failure in accident conditions.	Knowledge Based Ranking Scales	Data on quenching tests on irradiated and aged specimens is lacking or at least limited.	All operations

7	**Changes in the geometry of tubes produced by irradiation, creep**	The cladding must retain its mechanical properties and critical dimensions to contain the fuel and fission gases	Open literature	Phenomena Ranking Scales	May have an effect on thermohydraulics/heat transfer?	Knowledge Based Ranking Scales	No data. However, effect is not that large?	All operations
8	**Radiolysis processes**	Production of oxygen by radiolysis must be studied in order to reduce the corrosion of cladding and structural components.	Recent trends in Radiation Chemistry. J.F. Wishart (ed.) World Scientific.	Phenomena Ranking Scales	May have an accelerating effect on all corrosion processes.	Knowledge Based Ranking Scales	Some modelled data exists on the radiolysis in SCW, but its effect on the corrosion phenomena in SCW is unknown.	Normal operation
9	**Physicochemical properties of water within the SC region**	Close to the critical and pseudocritical point the dissolution of materials is promoted. The corrosion behaviour of materials at these points must be studied.	Kritzer, P.Journal of supercritical fluids, vol.29, nº1-2, (2004), pp.1-29.	Phenomena Ranking Scales	May be a critical factor with regards to corrosion in all regions that are near the critical temperature.	Knowledge Based Ranking Scales	Some lab data is available.	Normal operation
10	**Resistance of cladding materials under LOCA conditions SCWR**	The resistance of cladding material is a key parameter for reactor safety. The cladding must contain the fuel and retain the fission gases	I. Idarraga-Trujillo, et al. TopFuel 2013, Charlotte, North Carolina, September 15-19, 2013	Phenomena Ranking Scales	Large scale LOCA and subsequent cooling (quenching) effects may be detrimental, in large scale, to the integrity of the irradiated and aged cladding.	Knowledge Based Ranking Scales	Studied on present LWR materials, but not so much on SS or Alloy 800H and even less on materials exposed for a long time in SCW + n-irradiation.	Accidental operation (e.g. LB-LOCA)
11	**Impurity enrichment**	Enrichment of electrolytes (e.g., chlorides) in oxide deposits. This ties to the purification system during normal operation and transients.	Materials and Water Chemistry for Supercritical Water-cooled Reactors; Guzonas et al; Woodhead Publishing Series in Energy	Phenomena Ranking Scales	Enrichment of impurities in oxide deposits may promote SCC.	Knowledge Based Ranking Scales	A known phenomenon in present day PWR steam generators. Very little knowledge in the case of SCW.	Normal operation
12	**Oxide release from the cladding surface by spalling**	Transport of radioactivity to out-of-core surfaces. Damage to downstream corrosion by erosion. Blockage of piping or fuel assembly sub-channels.	Materials and Water Chemistry for Supercritical Water-cooled Reactors; Guzonas et al; Woodhead Publishing Series in Energy 2017; Irradiation Effects in cladding and structural materials; ASM 1965 and others	Phenomena Ranking Scales	Transport of radioactivity to out-of-core surfaces. Damage to downstream corrosion by erosion. Blockage of piping or fuel assembly sub-channels.	Knowledge Based Ranking Scales	Known phenomenon from fossil SCW plants.	Normal operation
13	**Irradiation embrittlement due to He**	He produced by transmutation reaction of Ni	X. Guo et al., "Corrosion resistance of candidate cladding materials for supercritical water reactor," Ann. Nucl. Energy, vol. 127, pp. 351–363, 2019, doi: 10.1016/j.anucene.2018.12.007.	Phenomena Ranking Scales	The amount of He depends on the amount of Ni in alloy as well as on the neutron flux.	Knowledge Based Ranking Scales	Please specify!	Operating status:
14	**IASCC**	Irradiation induced stress corrosion cracking	R. Zhou, E. A. West, Z. Jiao, and G. S. Was, "Irradiation-assisted stress corrosion cracking of austenitic alloys in supercritical water," J. Nucl. Mater., vol. 395, no. 1–3, pp. 11–22, 2009, doi:	Phenomena Ranking Scales	Please specify!	Knowledge Based Ranking Scales	Please specify!	Operating status:

			10.1016/j.jnucmat.2009.09.010.					
15	**Hydriding**	Internal hydriding due to to fabrication. External hydriding due to waterside corrosion	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	Cladding failure mechanism this is however early-in-life failure mechanism.	Knowledge Based Ranking Scales	Please specify!	All operations
16	**Cladding collapse**	If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential would exist for the cladding to collapse into a gap.	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	It is design criterion that cladding collapse is precluded during the fuel rod design lifetime.	Knowledge Based Ranking Scales	Please specify!	All operations
17	**Overheating of the Cladding**	fuel failure due to to overheating	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	The design criterion (PWR) is that there will be at least 95 percent probability, at a 95 percent confidence level, (95/95) that DNB will not occur on a fuel rod during normal operation and AOOs	Knowledge Based Ranking Scales	Please specify!	All operations
18	**Overheating of Fuel Pellets**	fuel failure due to to overheating	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	The design criterion is that no fuel centreline melting is allowed for normal operation and AOOs	Knowledge Based Ranking Scales	Please specify!	All operations
19	**Cladding rupture**	Cladding rupture due to fast heating rates	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	A cladding rupture temperature correlation must be used in the LOCA emergency core cooling system (ECCS) analysis	Knowledge Based Ranking Scales	Please specify!	All operations
20	**Fuel Rod Mechanical Fracturing**	It refers to a fuel rod defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	The design criterion to prevent fracturing is that stresses due to postulated accidents in combination with the normal steady-state fuel rod stresses should not exceed the yield strength of the components in their fuel assemblies.	Knowledge Based Ranking Scales	Please specify!	All operations
21	**Strain Fatigue**	Fuel rod failure due to fatigue or corrosion fatigue	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	The design criterion for cladding strain fatigue is that the cumulative fatigue usage factor be less than 0.9 when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles	Knowledge Based Ranking Scales	Please specify!	All operations
22	**Fretting Wear**	It can occur on the fuel rod cladding surfaces in contact with the spacer grids	D.B. Mitchel, B. M. Dunn: Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Phenomena Ranking Scales	The design criterion is to provide sufficient support to limit fuel rod vibration and cladding fretting wear	Knowledge Based Ranking Scales	Please specify!	All operations

Table A2-1: Significant phenomena identified by WP2 experts

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

Table A2-2: Expert ranking by WP2 members

	Contributor / Phenomenon	Ranking		Statistics			
		Importance level	Knowledge level	Importance deviation	Knowledge deviation	Relative relevance	Relative dispersion
1	Through wall penetrations	**2.00**	**2.71**	0.82	0.49	1.00	0.27
2	Oxide build-up	**1.57**	**2.29**	0.53	0.49	0.66	0.18
3	Oxide release from the cladding surface (general)	**2.00**	**2.50**	1.15	1.29	0.92	1.00
3a	Oxide release by dissolution / evaporation	**1.83**	**2.17**	0.75	0.41	0.73	0.21
4	Pellet cladding interaction	**1.57**	**1.83**	0.53	0.41	0.53	0.15
5	Environment Assisted Cracking (EAC)	**1.29**	**2.29**	0.76	0.49	0.54	0.25
6	ageing and/or irradiation	**1.57**	**2.14**	0.79	0.38	0.62	0.20
7	Changes in the geometry of tubes	**2.29**	**1.86**	0.76	0.38	0.78	0.19
8	Radiolysis processes	**1.29**	**1.43**	0.76	0.79	0.34	0.40
9	Physicochemical properties of SC water	**1.86**	**2.86**	0.90	1.07	0.98	0.65
10	Resistance of cladding materials under LOCA	**1.29**	**1.57**	0.76	0.53	0.37	0.27
11	Impurity enrichment	**1.86**	**1.86**	0.90	0.69	0.64	0.42
12	Oxide release from the cladding surface by spalling	**1.57**	**2.00**	0.53	0.00	0.58	0.00
13	Irradiation embrittlement due to He	**1.29**	**2.71**	0.49	0.49	0.64	0.16
14	IASCC	**1.29**	**2.14**	0.49	0.90	0.51	0.29
15	Hydriding	**2.29**	**2.14**	0.49	0.90	0.90	0.29
16	Cladding collapse	**2.00**	**2.00**	0.82	0.82	0.74	0.45
17	Overheating of the Cladding	**1.43**	**2.00**	0.79	0.82	0.53	0.43
18	Overheating of Fuel Pellets	**1.86**	**2.33**	0.69	1.03	0.80	0.48
19	Cladding rupture	**1.86**	**2.14**	0.90	0.38	0.73	0.23
20	Fuel Rod Mechanical Fracturing	**2.00**	**1.83**	0.89	0.75	0.68	0.45
21	Strain Fatigue	**2.00**	**1.83**	0.71	0.75	0.68	0.36
22	Fretting Wear	**2.33**	**2.29**	0.52	1.11	0.98	0.39

Table A2-3: Statistics of WP2 ranking results

Annex 3 – PIRT analysis results, WP3

#	Descriptive name of TH phenomena:	Short description of the item or notes:	Rational of the item:	Reference for the item:	Figure of merit	Operating status:
1	Steep non-linear change of SCW fluid material properties	The thermal hydraulic properties (e.g.: thermal conductivity, density, dynamic viscosity etc.) can rapidly and non-linearly change according to the pressure and temperature inside the reactor.	A trusted database of SCW material properties is essential for the system-, sub-channel- or CFD modelling of SCW TH in SCW-SMR relevant geometries.	See the open literature (e.g.: Pioro-Duffey book: Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications, ASME Press, 2007)	Fuel cladding temperature (T_c)	All operation
2	Heat transfer in water under supercritical pressure conditions (trusted prediction of heat transfer coefficient)	Heat transfer (Nu) correlation which can predict not only the enhanced (EHT), the normal (NHT) but the deteriorated heat transfer (DHT) and onset of heat transfer regeneration (OHTR) heat transfer regimes in pipes and rod bundles as well.	Most of the heat transfer correlations are capable to predict the EHT and NHT but fail to predict the DHT & OHTR heat transfer regimes. Only a few, mostly very recent heat transfer correlations can predict DHT & OHTR correctly (uncertainty of correlation results depends on the main parameters of the investigated case). An accurate and trusted HT correlation is needed for system- and sub-channel code calculations.	See the open literature.	Fuel cladding temperature (T_c)	All operations
3	Pressure drop (Δp) in water under supercritical pressure conditions in SCW-SMR (SCWR) relevant geometries	The pressure drop correlations can only describe vertical and horizontal tube or duct geometries, but not fuel rod bundles and assemblies.	A pressure drop (Δp) correlation which can predict all of the pressure drop components (due to gravity, frictional resistance, acceleration of the flow and local flow obstruction) in SCW-SMR related geometries (not only in horizontally or vertically installed tube or duct but in horizontally or vertically installed rod bundle or fuel assembly geometries as well) is needed for thermal hydraulic calculations.	See the open literature (e.g.: Pioro-Duffey book: Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications, ASME Press, 2007)	Fuel cladding temperature (T_c)	All operations
4	Turbulent heat and mass transfer in water under supercritical pressure conditions in SCW-SMR relevant (in horizontally installed tube but mainly rod bundle) geometries	The RANS CFD approach uses turbulence models in order to model the turbulence of the flow. This modelling substantially influences the correctness of the CFD results. One of the main problems with the currently available turbulence models is that they uses wall functions (for high Reynolds number approaches) or damping functions (for low Reynolds number approaches) which were developed from constant thermophysical properties assumption originally from the classical boundary layer theory. Of course there are many other problematic aspects of current models.	It is very important for proper turbulence modelling at CFD analysis for SCW-SMR related geometries under supercritical water conditions. For obvious reasons there is an essential need for trusted and available experimental data in order to validate the set of different (system-, sub-channel, CFD, etc.) codes and their models for SCW-SMR application. For example we do need experimental data on the thermal hydraulics in horizontally installed rod bundle geometry.	See the open literature.	Fuel cladding temperature (T_c)	All operations

5	Heat and mass transfer of coolant/moderator SCW along corroded and rough surfaces of the core structural element	As it is well known the SCW is chemically very aggressive and thus the (metallic) structural surfaces which are in contact with SCW suffer corrosion damage especially during long term operation in an SCW-SMR or other type of SCWR. This corrosion can cause changes in the thermal hydraulic (heat-transfer, mass-transfer, pressure-drop) properties of the affected geometries.	It is very important to know how the thermal hydraulic properties will change on the corroded geometries in order to properly model the heat and mass transfer of the reactor. Since the knowledge is so limited it is possible that the corrosion can cause significant safety issues.	See the open literature. + Dr. Ivan Otic (KIT, Germany) and his team investigates this topic.	Fuel cladding temperature (T_c)	All operations
6	Deterioration of heat transfer (DHT) or heat transfer deterioration (HTD)	The DHT in supercritical state believed to be a counterpart phenomenon with respect to the DNB in sub-critical state but less sudden and less dangerous.	This phenomenon is still partially known and there is a lack of trusted heat transfer correlation or turbulence model for RANS approach which is able to predict DHT in every investigated case.	See the open literature (e.g.: Pioro-Duffey book: Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications, ASME Press, 2007)	Fuel cladding temperature (T_c)	All operations
7	Transition from supercritical to subcritical pressure water state	During accident scenarios when a pressure loss occurs the supercritical water will undergo a short but important transient phase before it becomes a stable two-phase fluid. This item is in a close relation to the TH11, but it is reasonable to handle them separately during the PIRT analysis due to its importance in system and other type of code modelling.	The system codes have to simulate the transition from supercritical to subcritical pressure water state in order to predict for example LOCA events. Currently these codes are not universally able to simulate the transition (for example the Apros system code can predict the transition but the simulation slows down at the transition and the results become relatively uncertain from this part of the simulation). The relevant models of the system codes need to be upgraded or changed completely in order to be able to simulate the transition phase.	See the open literature.	Fuel cladding temperature (T_c)	Accidental operation (e.g. LB-LOCA)
8	Steam and liquid water two phase flow in SCW-SMR primary loop	After depressurizations the single-phase supercritical fluid will become a two-phased steam-liquid fluid mixture. The two-phase flow has significantly different thermal hydraulic properties and modelling requirements as its single-phase counterpart.	The modelling of steam and liquid water two phase flow under sub-critical pressure condition in SCW-SMR primary loop after a fast depressurization still seems to be a challenge.	See the open literature.	Fuel cladding temperature (T_c)	Accidental operation (e.g. LB-LOCA)
9	Natural circulation of water under super- or sub-critical pressure conditions in SCW-SMR primary loop	The reactor is being designed in a way that the coolant water flow is always horizontal or upward. Ideally this will mean that a natural circulation can occur inside the primary loop.	The possible role of natural circulation in SCW-SMR primary loop is questionable and cannot be determined until a first draft of the primary loop will be designed.	See the open literature.	Fuel cladding temperature (T_c)	All operations
10	Strong coupling between the thermal hydraulics (e.g. SCW density field) and the reactor physics (e.g. heat production by nuclear fission in the core)	As it is well known there is a strong coupling between the thermal hydraulics (e.g. SCW density field) and the reactor physics (e.g. heat production by nuclear fission in the core) in SCWR type concepts which has to be handled with great care. This is the reason why multi-physical approach should be used to predict this complex behaviour of the SCW-SMR core	The strong coupling can cause safety issues if not modelled properly. This is the reason why multi-physical approach should be used to predict this complex behaviour of the SCW-SMR core.	See the open literature.	Fuel cladding temperature (T_c)	All operations

11	Depressurisation of the primary loop and the travelling depressurisation wave and its path in case of a LOCA	A sudden and steep decrease in the system pressure of the primary loop which occurs in case of a LOCA accident (or in case of operation of automatic depressurisation devices), when the depressurisation wave travels from the place of the break towards the far locations of the primary loop.	It is crucial for accident scenario calculations to understand the different events that happen during depressurization.	See the open literature.	Fuel cladding temperature (T_c)	Accidental operation (e.g. LB-LOCA)
12	The effect of the presence of large and hot structural components (large amount of structural material) in the reactor pressure vessel (RPV) during accidents	The large amount of structural material will have the same temperature as the supercritical water coolant. This will create a significant heat storage effect.	The presence of large amount of structural material in the reactor pressure vessel (RPV) could cause cooling problems in accident scenarios due to these large components store huge amount of heat and make more difficult the cooling of the core	See the open literature.	Fuel cladding temperature (T_c)	Accidental operation (e.g. LB-LOCA)
13	Flow instability under supercritical pressure conditions in SCW-SMR relevant (in horizontally installed tube but mainly rod bundle) geometries	The flow could become unstable under forced but mainly mixed and natural circulation parameters which could cause thermal hydraulic problems and could lead to damages on structural materials.	The relatively large temperature differences between heated surfaces and bulk fluid can cause high differences in densities. This leads to relatively strong buoyancy effects. Experimental results are usually available for horizontal pipes, not for rod bundles	See the open literature.	Fuel cladding temperature (T_c)	All operations
14	Allowable maximum cladding temperature during normal operation (WP2-WP3) in the SCW-SMR	Based on previous coupled TH-Reactor physics calculations, the maximum cladding temperature might reach unacceptably high values from cladding material point of view (WP2) in normal steady-state operation.	Alternative core configurations are under investigation with TH-Reactor physics models to reduce the maximum cladding temperature. Continuous negotiation between WP2-WP3 is needed to find an optimum solution.	See the open literature.	Fuel cladding temperature (T_c)	Normal operation
15	Flow stratification in horizontal channels. The lighter fluid will tend to move upwards in the fuel assembly	Flow stratification in horizontal channels where the lighter fluid will tend to move upwards in the fuel assembly.	Under mixed and mainly natural convection circumstances this could lead to thermal hydraulic problems.	See the open literature.	Fuel cladding temperature (T_c)	All operations
16	Flooding	Effect of safety injection on the thermal hydraulic parameters during LOCA accidents	Injection of low temperature emergency coolant may have an adverse effect on safety (higher original fuel temperatures, etc.)	See the open literature.	Fuel cladding temperature (T_c)	Accidental operation (e.g. LB-LOCA)
17	Thermal hydraulics and Neutronic coupled instabilities	Please, specify	Please, specify	See the open literature.	Fuel cladding temperature (T_c)	All operations
18	CHF near the critical point.	Near the critical point and in subcritical state, mainly under certain accident conditions, CHF may cause heat transfer problems	Please, specify	See the open literature.	Fuel cladding temperature (T_c)	All operations

19	Flow induced vibration	Flow induced vibration (and subsequent damage of fuel cladding) may cause the loss of integrity of fuel cladding	Please, specify	See the open literature, e.g. "Experimental Investigation on Flow-Induced Vibration of Fuel Rods in Supercritical Water Loop" by Licun Wu	Fuel integrity (release of volatile FPs)	All operations
20	Mechanical deformation (coupled with thermal hydraulics)	Please, specify	Please, specify	See the open literature.	Fuel cladding temperature (T_c)	All operations
21	Pellet/cladding interaction (conductance)	Please, specify	Please, specify	See the open literature.	Fuel cladding temperature (T_c)	All operations

Table A3-1: Significant phenomena identified by WP3 members

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

Table A3-2: Expert ranking by WP3 members

		Ranking		Statistics			
#	phenomenon / contributor	Importance level	Knowledge level	Importance deviation	Knowledge deviation	Relative relevance	Relative dispersion
1	Steep non-linear change of SCW fluid material properties	1.2	3.4	0.45	0.55	0.42	0.42
2	Heat transfer in water under supercritical pressure conditions	1	2	0.00	0.00	0.00	0.00
3	Pressure drop (Δp) in water under supercritical pressure conditions	2	2.4	0.00	0.55	0.00	0.00
4	Turbulent heat and mass transfer in water under supercritical pressure conditions	1.2	1.8	0.45	0.45	0.34	0.34
5	Heat and mass transfer along corroded and rough surfaces	1.2	1.4	0.45	0.55	0.42	0.42
6	Deterioration of heat transfer (DHT)	1	2	0.00	0.00	0.00	0.00
7	Transition from supercritical to subcritical pressure	1.2	2	0.45	0.71	0.54	0.54
8	Steam and liquid water two phase flow	1.4	2.8	0.55	0.45	0.42	0.42
9	Natural circulation of water under super- or sub-critical pressure conditions	1.4	2	0.55	0.00	0.00	0.00
10	Strong coupling between the thermal hydraulics and the reactor physics	1.25	3	0.50	0.82	0.69	0.69
11	Depressurisation of the primary loop and the travelling depressurisation wave	1.6	2.4	0.55	0.55	0.51	0.51
12	The effect of the presence of large and hot structural components	2.2	3	0.84	0.71	1.00	1.00
13	Flow instability under supercritical pressure conditions	1.4	2.4	0.55	0.55	0.51	0.51
14	Allowable maximum cladding temperature	1	2.4	0.00	0.55	0.00	0.00
15	Flow stratification in horizontal channels	2	2.2	0.71	0.45	0.54	0.54
16	Flooding	1.2	3	0.45	0.71	0.54	0.54
17	TH and Neutronic instabilities	1	2.5	0.00	1.29	0.00	0.00
18	CHF near the critical point	1.5	2	0.55	0.63	0.59	0.59
19	Flow induced vibration	2	2.5	0.00	0.58	0.00	0.00
20	Mechanical deformation	1.75	2.75	0.50	0.96	0.81	0.81
21	Pellet/cladding interaction	1.67	3	0.58	0.00	0.00	0.00

Table A3-3: Statistics of WP3 ranking results

Annex 4 – PIRT analysis results, WP4

RP PIRT: normal and all operations				
#	**Descriptive name of reactor physical phenomena:**	**Short description of the item or notes:**	**Reference for the item:**	**Operating status:**
Computation methodology 1: Deterministic calculations (e.g. for TH coupling) **FoMs: ability to determine the effective multiplication factor (FoM1) or neutron flux distribution (FoM2)**				
1	Required transport approximation	Diffusion or higher-order transport approximations (e.g. SP3) are necessary to accurately model the reactor power distribution.	n.a.	All operations
2	Number of necessary energy groups and energy group structure	Due to higher temperatures, and harder spectrum, it is relevant to investigate the number of necessary energy groups and specific energy group structure for the deterministic simulations (especially the thermal energy range).	W. Shen: Assessment of the traditional neutron-diffusion core-analysis method for the analysis of the Super Critical Water Reactor, Annals of Nuclear Energy 45 1–7, 2012. & A. Moghrabi, D.R. Novog: Determination of the optimal few-energy group structure for the Canadian Super Critical Water-cooled Reactor, Annals of Nuclear Energy 115 27–38, 2018.	All operations
3	Spatial discretization	It is necessary to determine the appropriate spatial discretisation, e.g. FA node level or pin level calculations are necessary, what is the required number of axial nodes. Besides, in the light of the applied numerical method, the numerical spatial discretization level can also have an influence.	n.a.	All operations
4	Time discretisation	It is necessary to determine the appropriate temporal discretization applied in time-dependent calculations.	n.a.	All operations
5	Reflector boundary and group constants	For deterministic codes the modelling of reflector region is not as straightforward as for neutron transport codes.	n.a.	All operations
Computation methodology 2: Monte Carlo calculations **FoMs: ability to determine the effective multiplication factor (FoM1) or neutron flux distribution (FoM2)**				
6	Geometric parameters	Appropriate modeling of geometric parameters is necessary	n.a.	Normal operation
7	Material composition, also including burn-up	Appropriate modeling of material composition is necessary	n.a.	Normal operation
8	Material properties (temperature, density, etc.)	For accurate simulations, the material properties need to be known.	n.a.	Normal operation
9	Operational history for burn-up calculations	For accurate burn-up calculations, the operational history of the reactor is necessary.	n.a.	Normal operation
Material and neutron physics related data **FoMs: impact on the effective multiplication factor (FoM1) or neutron flux distribution (FoM2)**				

	Material compositions	> For group constant generation, the accurate material composition of the components is needed. > Appropriate modeling of material composition is also necessary for the Monte Carlo calculations.	*See below.*	All operations
10	> *Composition of structural materials*	For precise calculations, the composition of structural materials must be determined	WP2 D2.1 report	All operations
11	> *Composition of fuel materials*	For precise calculations, the composition of fuel materials must be determined.	n.a.	All operations
12	> *Composition of control rods*	For precise calculations, the material composition of control rods must be determined.	n.a.	All operations
	Material properties	> For group constant generation, the accurate material properties of the components are needed. > For accurate simulations, the material properties need to be known.	*See below.*	All operations
13	> *Temperature and density of structural materials*	Structural material temperatures and densities need to be determined precisely for the applied neutronic (and TH) codes.	n.a.	All operations
14	> *Temperature and density of fuel materials*	Fuel temperatures and densities need to be determined precisely for the applied neutronic (and TH) codes.	n.a.	All operations
15	> *Temperature and density of the coolant and moderator*	*	n.a.	All operations
	Cross sectional data	Neutron transport codes must rely upon the cross sectional data of the different materials assigned to the geometric models. Therefore, these cross sectional lists must be determined precisely in the relevant temperature and energy regions.	*See below.*	All operations
16	> *Fuel material cross sections*	⮅	F. Rahnema, Ch. Edgar, D. Zhang, and B. Petrovic: Phenomena Identification and Ranking Tables (PIRT) Report for Fluoride High-Temperature Reactor (FHR) Neutronics. Georgia Institute of Technology, CRMP-2016-08-001.	All operations
17	> *Coolant and moderator cross sections*	⮅	n.a.	All operations
17	> *Structural material cross sections*	⮅	n.a.	All operations
18	> *Control rod cross sections*	⮅	n.a.	All operations
18	> *Burnable absorber cross sections*	⮅	n.a.	All operations

19	Applied burnable absorbers	⮅	Jordan A. Evans, Mark D. DeHart, Kevan D. Weaver, Dennis D. Keiser: Burnable absorbers in nuclear reactors – A review, Nuclear Engineering and Design, 391, 2022, 111726.	Normal operation
Design related data **FoMs: impact on the effective multiplication factor (FoM1) or neutron flux distribution (FoM2)**				
20	Reactivity control system	The design of reactivity control system and its operation/impact during burnup as well as the applied materials have a strong effect on the global neutron flux and the reactivity.	n.a.	All operations
21	Geometric parameters	> For group constant generation, the accurate dimensions of the fuel are needed. > Appropriate modeling of geometric parameters is also necessary.	n.a.	All operations
22	In-core detectors	The design of the in-core detector system is crucial for monitoring neutronic and TH parameters and for reactor operation in both normal and accidental scenarios.	n.a.	All operations

Table A4-1: Significant phenomena identified by WP4 members

RP PIRT: REA				
#	**Descriptive name of reactor physical phenomena:**	**Short description of the item or notes:**	**Reference for the item:**	**Operating status:**
FoMs: power history during REA transient (FoM1) and pin fuel enthalpy during REA transient (FoM2) ***(FoM2 includes the peak cladding temperature for which an acceptance limit is defined)***				
1	Ejected control rod worth	During the REA, the worth of the ejected control rod will determine the amount of reactivity that is inserted into the system. The inserted reactivity will result in power increase which will be limited by the Doppler-feedback.	D.J. Diamond, B.P. Bromley, and A.L. Aronson: Studies of the Rod Ejection Accident in a PWR. Energy Sciences & Technology Dept., Brookhaven National Laboratory, Technical report W-6382 1/22/02 David J. Diamond: Experience Using Phenomena Identification and Ranking Technique (PIRT) for Nuclear Analysis. ES&T Department / NEIS Division Brookhaven National Laboratory, BNL-76750-2006-CP	REA
2	Rate of reactivity insertion	The reactivity insertion rate will mainly depend on the ejected control rod worth.	↑↑	REA
3	Fuel temperature coefficient (FTC)	The reactivity decreasing effect of the Doppler-feedback will depend on the FTC (which is mainly determined by the Doppler coefficient (DTC)).	↑↑	REA
4	Moderator temperature coefficient (MTC)	As an immediate result of the rod ejection, the fuel temperature will increase due to the increased fission rate which in turn increases the moderator temperature.	↑↑	REA
5	Delayed-neutron fraction	Delayed-neutron fraction is another important parameter aside from the ejected rod worth in REA scenarios.	↑↑	REA
6	Fuel pellet radial power distribution	The radial power distribution of the fuel pellets will affect the increase in pin fuel enthalpy during the REA transient.	↑↑	REA
7	Pin peaking factors	The pin peaking factors will affect the increase in pin fuel enthalpy during the REA transient.	↑↑	REA
8	Thermal-hydraulic parameters	There are certain thermal-hydraulic parameters which are required in order to calculate FoMs or parameters related to the calculation of FoMs. These thermal-hydraulic parameters are: > the heat resistances in fuel (at different burnup levels), fuel gap and cladding > the cladding-to-coolant heat transfer coefficient during the transient > the heat capacities of the fuel and the cladding.	↑↑	REA

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