ELSMOR
Towards European Licensing of
Small Modular Reactors

Improved safety features of LW-SMR

WP 1: Identification of improved safety features of LW-SMRs

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Summary

The report describes the final results of WP1 (Identification of improved features of LW-SMRs) of the European project ELSMOR. Within the work package, European nuclear safety directives and good practices on safety assessment of LWR have been reviewed in terms of special regulations regarding LW-SMR.

Furthermore, a number of SMR on an advanced development stage have been reviewed by using publicly available materials like conference or journal paper, documents of IAEA and data provided by project partners. The resulting SMR descriptions obtained within the work package include a general technical description of the SMR concept as well as descriptions regarding the safety systems. The different systems and safety features were classified into the defence in depth concept for each design.

For further consideration in the project, potential challenges for safety demonstration have been identified and are proposed as an input for the second work package of ELSMOR (Development of safety analysis methodology for LW-SMR).

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1. Introduction

As within ELSMOR the goal is to create methods and tools for the European stakeholders to assess and verify the safety of light water small modular reactors (LW-SMR) which would be deployed in Europe, the first step is to determine the status quo regarding the current European nuclear safety directives and good practices on safety assessment of LWR. Furthermore, it is important to identify, how the SMR designers try to comply with the existing regulations in the already existing or currently developed SMR designs.

Several design decisions can be seen as direct consequence of the need for cost reduction, especially regarding cost overruns during the construction of large nuclear power plant (NPP). One example is the integration of the whole primary circuit into the reactor pressure vessel (RPV). In connection with a small containment, housing the RPV facilitates the construction of individual reactor modules within a factory with all its opportunities like standardisation and high-quality production. From the safety demonstration point of view, large break loss of coolant accidents (LBLOCA) might not be relevant, since the lines connected to the RPV are small compared to the hot and cold leg lines in a large reactor. Further simplification of the whole design is often achieved by implementation of passive safety systems instead of active ones. It is claimed that passive systems are more reliable and safer than their active counterparts. But the safety demonstration of such systems is highly challenging since their functional principle is often based on small driving forces (e.g. density differences between small height differences). Due to the low power of a SMR module, it is envisaged to couple a number of modules into a multi-unit side, in which the different modules could share systems (e.g. a large water pool providing a large heat sink for decay power) or even the control room. Also here questions arise, e.g., how the design deals with the impact of an accident within one module on the neighbouring modules. Such challenges for established safety demonstration methodologies need to be identified and dealt with.

The above-mentioned work has been performed in WP1 of the ELSMOR project. As written in /ELS 18/ primarily, the European nuclear safety directives and good practices on safety assessment have been reviewed (T1.1). These directives need to be fulfilled by all SMR designs, planned to be deployed in the EU. In T1.2, several SMR concepts have been screened regarding their specific design feature in light of the regulations. Evaluated references were mainly publicly available material like conference proceedings, journal papers or IAEA documents, but also information provided by project partners. The information is gathered and compiled into text descriptions of the designs regarding general data and descriptions of the reactor itself as well as of functionality of safety systems and the safety concept. It was also tried to classify the different safety systems and special features into the defence in depth concept if it was not already described in the references. Potential challenges for safety demonstration have been finally identified in T1.3 providing an input for the further work in ELSMOR, especially for WP2.

2. Review of safety directives

The review of the European safety directives and good practices on safety assessment of LWR has been performed based on the following information:

- European safety directives
- IAEA guidance
- WENRA and ENSREG guidance

Furthermore, national rules and regulations on selected EU countries needed to be reviewed. This task was extended also to the non-EU countries Canada, Russia and USA, since they are currently in a licensing process or have even deployed the first reactors and was performed first on the experiences of the different ELSMOR partners involved in this task. Additionally, a questionnaire has been created and sent to different regulatory bodies.

2.1 Review of European safety directives with respect to SMRs

This chapter focuses on the four main European safety directives: 2009/71/Euratom /EUR 09/, amended by 2014/87/Euratom /EUR 14/ establishing a “framework for the nuclear safety of nuclear installations, 2011/70/Euratom /EUR 11/ on spent fuel and radioactive waste management and 2013/59/EURATOM /EUR 19/ on radiation protection issues.

By nature, the articles of these directives are very high-level and technology neutral, in accordance to article 2 of 2014/87/Euratom /EUR 14/: “this Directive shall apply to any civilian nuclear installation subject to a licence”. All articles appear to be fully applicable to any SMR design, regardless of its technology.

An interesting update was performed between 2009/71/Euratom /EUR 09/ and 2014/87/Euratom /EUR 14/ in its articles 8a through 8c. These articles reflect:

- An overarching safety objective to be applied “to nuclear installations for which a construction licence is granted for the first time after 14 August 2014”, meaning it applies to every SMR undergoing a licensing process in the European Union;
- The principle of defence-in-depth in line with WENRA practical elimination of accidental releases and on-site preparedness: fully applicable for SMRs;
- The promotion of “an effective nuclear safety culture”, including management systems - this is critical for SMRs, since every step of the SMR licensing process (development, manufacturing, operation, maintenance, decommissioning, review of the safety assessment reports) may involve newcomers who might have different levels of safety culture;
- The need for periodic safety reviews (PSRs) as well as continuous improvement of the nuclear facility - expectations for PSRs and improvement should not be different than for large scale plants.

Directive 2011/70/Euratom /EUR 11/ widely acknowledges for research reactors, for example, allowing for manufacturing of the fuel outside the country where the fuel is burnt or reprocessing of the spent fuel in another country (articles 3 and 4). The management of spent fuel and radioactive waste for SMRs is expected to be a combination of what is done for large scale plants (notable quantities of spent fuel and wastes, including high activity wastes) and what is done for research reactors (smaller quantities of fuel coming from several different entities and locations). To that extent, Directive 2011/70/Euratom /EUR 11/ seems broadly applicable to the management of spent fuel and wastes for SMRs. Importantly, as LW-SMR use fuel designs that are broadly similar to existing LWR, this Directive imposes no notable additional restraints on new SMR designs.

Finally, Council Directive 2013/59/EURATOM /EUR 19/, which replaced Directive 96/29/Euratom, sets out the framework on radiation protection for nuclear facilities, including dose limits for occupational exposure, emergency exposure and public exposure. Importantly, it sets limits relevant to the definition of evacuation or exclusion zones from accidental situations, with acute emergency exposure limits between 20 mSv and 100 mSv and limits below 20 mSv per year for transition into an existing exposure situation. This is immediately relevant to the definition of exclusions zones for SMR as well as the definition of accidental level releases to be practically eliminated.

2.2 Review of IAEA guidance with respect to SMRs

There is no IAEA safety guidance specifically for SMRs. All IAEA guidance documents on reactor safety should apply to SMRs.

The IAEA Fundamental Safety Principles /IAEA 06a/ states the fundamental safety objective ("Protect people and the environment from harmful effects of ionizing radiation") and the ten safety principles. These apply to any country that operates nuclear facilities including SMRs.

In 2015 in response to the Fukushima Daiichi accidents, the diplomatic conference on the conventions of nuclear safety promulgated the Vienna Declaration /IAEA 15/. It specifically underlines and reinforces that new NPP shall be designed and operated such as to prevent accidents and that accidental level releases shall not require long-term protective measures. As it is binding to all EU countries, it is immediately relevant to SMRs.

IAEA guide "Safety of Nuclear Power Plants: Design" (IAEA Safety Standards Series no. SSR-2/1) /IAEA 16a/ establishes design requirements for the structures, systems and components of NPPs, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur. The document contains in total 82 design requirements covering the following areas: safety management in design, principal technical requirements (e.g. fundamental safety functions, radiation protection in design, design of a NPP, application of DiD, etc.), general plant design (e.g. plant state categories, postulated initiating events, internal & external hazards, DBA, DEC, common cause failures, etc.) and design of specific plant systems (e.g. reactor core control, design of reactor coolant systems, overpressure protection of the reactor coolant pressure boundary, heat transfer to ultimate heat sink, containment system, etc.). In Section 1.6 of the document it is stated that the publication will be used primarily for land based stationary NPPs with water cooled reactors designed for electricity generation or for other heat production applications (such as district heating or desalination), which includes water-cooled SMRs.

IAEA guide "Safety of Nuclear Power Plants: Commissioning and Operation" (IAEA Safety Standards Series no. SSR-2/2) /IAEA 16b/ establishes the requirements which, in the light of experience and the present state of technology, must be satisfied to ensure the safe commissioning and operation of NPPs. These requirements are governed by the safety objective and safety principles that are established in the IAEA Fundamental Safety Principles /IAEA 06a/. The publication contains in total 33 requirements for the safe commissioning and operation of a NPP covering commissioning and operation up to the removal of nuclear fuel from the plant, including maintenance and modifications made throughout the lifetime of the plant and preparation for decommissioning (but not the decommissioning phase itself). The publication also establishes additional requirements relating only to commissioning. Normal operation and anticipated operational occurrences as well as accident conditions are taken into account /IAEA 16b/. The requirements described in the document should apply to any NPP including water-cooled SMRs.

Beside the above three more general guidance documents the IAEA has published a significant number of safety guides providing considerations on the design of specific SSC, e.g. IAEA Safety Guide no. SSG-56 “Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants” /IAEA 20/, IAEA Safety Guide no. SSG-53 “Design of Reactor Containment Systems for Nuclear Power Plants” /IAEA 19a/, IAEA Safety Guide no. SSG-52 “Design of the Reactor Core for Nuclear Power Plants” /IAEA 19/, etc. and these apply to a large extent also to water-cooled SMRs.

Regarding emergency preparedness, IAEA GSR Part 7 “Preparedness and Response for a Nuclear or Radiological Emergency” /IAEA 15a/ formulates high-level requirements regarding accidental release situations from nuclear installations and associated roles and responsibilities. It includes dose limits regarding accidental exposure and transition to an existing exposure situation consistent to EU directive 2013/59/EURATOM /EUR 19/ (see above), which is immediately relevant for the definition of exclusion zones and practical elimination. Further IAEA guide like GSG-2 “Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency” /IAEA 11a/ or GSG-11 “Arrangements for the Termination of a Nuclear or Radiological Emergency” /IAEA 18a/ give further more detailed guidance. Regarding radiation protection, IAEA GSR Part 3 “Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards” /IAEA 14/ entails the high-level requirements by the IAEA, again generally consistent with EU directive 2013/59/EURATOM. All of this guidance is fully applicable to SMRs, however it does not impose qualitative new or different challenges compared to larger reactor designs.

Finally, IAEA GSR Part 5 “Predisposal Management of Radioactive Waste” /IAEA 09/ and SSR-5 “Disposal of Radioactive Waste” /IAEA 11/ formulate requirements on how to manage radioactive wastes. These are also generally applicable to SMR. However, there should be no major additional constraints for SMR compared to larger scale reactors.

2.3 Review of WENRA guidance with respect to SMRs

An international regulatory framework within Europe is the Western European Nuclear Regulators' Association (WENRA), a non-governmental organisation set up in 1999. WENRA is an organisation of EU countries operating or having operated NPPs, as well as Switzerland and Ukraine. The main objectives of WENRA are to develop a common approach to nuclear safety, to provide an independent capability to examine nuclear safety in applicant countries and to be a network of chief nuclear safety regulators in Europe exchanging experience and discussing significant safety issues. One of the first major achievements to this end was the publication in 2006 of a set of Safety Reference Levels (SRLs) for operating NPPs. After the TEPCO Fukushima Daiichi nuclear accident, these SRLs – already revised in 2007 and 2008 – have been further updated to take into account the lessons learned, including the insight from the EU stress tests. As a result, a new issue on natural hazards was developed and significant changes made to several existing issues /WEN 14/. WENRA SRLs are the basis for the harmonization of national safety requirements for existing operating NPPs. These guidelines are already used in Europe and they are incorporated stepwise into the regulatory frameworks of member countries. Notably, the WENRA reference levels are broadly technology neutral and could be applied to SMRs. Furthermore, /WEN 14/ specify the defense-in-depth concept endorsed by WENRA and for design extension conditions (see also IAEA SSR 2/1 Rev. 1 /IAEA 16a/) define two under-categories: DEC-A for severe fuel damage and DEC-B for an accidental level large or early release.

The Reactor Harmonization Working Group (RHWG) within WENRA has developed Safety objectives for new NPPs in 2009 /WEN 09/ and WENRA has issued a Statement on safety objectives for new NPPs in 2010 /WEN 10/. The Statement /WEN 10/ has been supplemented by a report issued by the RHWG in 2013 /WEN 13/, which aimed at clarifying the objectives by developing the rational and common positions on selected key safety issues for the design of new NPPs. It is to be noted that the RHWG report /WEN 13/ included also the lessons learnt from the Fukushima accident. The second Working Group on Waste and Decommissioning (WGWD) within of WENRA covers different safety aspects related to waste treatment and decommissioning. WGWD has issued comprehensive set of SRLs on decommissioning, storage of waste and spent fuel, processing of radioactive waste and disposal that, to large extent, are applicable to SMR.

These WENRA documents relative to the safety objectives of new NPPs were reviewed focusing on the specific features for SMR application. It should be noted that they describe basic, high level procedures but do not regulate details. The WENRA documents do not include recommendations on or analysis of safety assessment methodologies.

WENRA safety objectives for new power reactors (/WEN 09/, /WEN 10/) have been defined on the basis of a systematic analysis of the Fundamental Safety Principles /IAEA 06/. Each Fundamental Safety Principle has been investigated to check whether, on the basis of the review of the existing documentation, safety objectives related to this principle needed to be further expressed.

In /WEN 09/ it is recommended, that the designers of new reactors should endeavour to improve the safety of their new reactors, as compared with existing reactors:

“In line with fundamental safety principle n°5 “optimization of protection”, the safety of new reactors will have to be improved as far as reasonably achievable starting from the design stage with due consideration given to insights gained from:

- Experience feedback from existing reactors;
- Deterministic and probabilistic safety assessments;
- State of the art technologies, analysis methodologies and techniques;
- Results of safety research

For new reactors, more significant improvements in the design over what has been done before become now reasonably achievable, in particular concerning prevention and mitigation of severe accidents, including in the long term phase” /WEN 09/.

The proposed safety objectives for new reactors have been selected to further improve the protection of people and of the environment. However, since nuclear safety and what is considered adequate protection are not static entities, the safety objectives that are proposed in this report may be subject to further evolution. They have been formulated in a qualitative way. Besides, European citizens can easily understand WENRA’s expectations in terms of protection of the public (consequences of accidents) and workers (radiation protection); protection of the environment (discharges) and future generations (waste and dismantling).

In developing WENRA’s Safety Objectives, the following key considerations have been taken into account, among others:

- For new reactors, significant improvements in the design become now reasonably achievable, in particular concerning prevention and mitigation of severe accidents, including in the long-term phase.
- PSA shall be used as part of the design process.
- The defence-in-depth (DiD) concept remains the key safety approach for new reactors. Strengthening of its implementation has to be aimed for reinforcement of each level of the concept and improvement of the independence between the levels of defence-in-depth.
- Security features for new reactors should also be considered consistently with safety ones.

- Quality assurance and management of safety are key elements of the prevention of accidents.

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

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

In this frame, WENRA proposed seven safety objectives, which are applicable to the design, siting, constructions, commissioning and operation, entitled as follow /WEN 10/:

O1: Normal operation, abnormal events and prevention of accidents

O2: Accidents without core melt

O3: Accidents with core melt

O4: Independence between all levels of Defence-in-Depth

O5: Safety and security interfaces

O6: Radiation protection and waste management

O7: Leadership and management of safety

The above listed objectives have been elaborated in 2009 for new PWRs that were underway or planned. At that time, they reflected the common vision and expectations within WENRA concerning the safety objectives for NPPs to be built in the near term. However, regulators had identified several areas for safety enhancement, essentially from the feedback of the assessments of NPPs in operation and under construction.

O1. Normal operation, abnormal events and prevention of accidents

- Reducing the frequencies of abnormal events by enhancing plant capability to stay within normal operation;

- Reducing the potential for escalation to accident situations by enhancing plant capability to control abnormal events.

In other words, this objective prescribes for better control of NPP operation and better prevention of accidents. It is evident that the prevention of accidents is based primarily on careful selection of materials, use of qualified fabrication processes together with extensive testing programmes.

The RHWG considers that there is merit for countries to use quantitative safety targets along with the proposed qualitative safety objectives. As safety targets, these values are useful to drive in-depth technical discussions with the applicants aimed at identifying real safety improvements, rather than being used as stand-alone acceptance criteria. /WEN 09/ gives some tracks from which “compliance with the qualitative safety objective O1 is expected to be appreciated through:

- Demonstration that all operational experience feedback has been used to identify the safety issues of existing plants that could be relevant for the envisaged new design;

- Verification that appropriately validated means have been designed to address these issues;

- Implementation of extended operational margins.”

Before applying for a license, the licensee should analyse the relevant operational feedback of NPP to identify some safety issues relevant for the new reactor, i.e. screening of abnormal events occurrence, experience of accident management. Then the licensee shall demonstrate in the safety report that provisions are taken to cope with these issues. Provisions could be of design nature or could rely on the plant operating control or protection system.

Regarding “the implementation of extended operational margins” the main operational margins are shaped by the SMR design because operational parameters are self-limited through the inherent core characteristics. Thus, the applicant shall demonstrate the design margins are sufficient to limit the occurrence of abnormal events, and should they occur, to prevent escalation to accident conditions. On the other hand, an enhanced plant monitoring would probably be needed, and the control of operational parameters has to be completed by custom measurements able to detect any abnormal event occurrence that could jeopardize the safety functions.

O2. Accidents without core melt

- Ensuring that accidents without core melt induce no off-site radiological impact or only minor radiological impact (in particular, no necessity of iodine prophylaxis, sheltering nor evacuation).

- Reducing, as far as reasonably achievable,

    - Core damage frequency taking into account all types of credible hazards and failures and credible combinations of events;

    - Releases of radioactive material from all sources.

- Providing due consideration to siting and design to reduce the impact of external hazards and malevolent acts.”

O2 objectives could be interpreted as those objectives applicable to all the accidents which are enveloped by the accidental scenarios initiated by single initiating events or a relevant set of plausible multiple failures considered within the design basis or as design extension conditions. O2 states that accidents without core melt should not have significant NPP off-site impact. “Core melt” might not be a relevant terminology for some technologies (typically molten salts reactors). However, safety design for all reactors is oriented by the need for a highly reliable prevention of core degradation (core meltdown), and this remains true for SMRs.

O3. Accidents with core melt

- Reducing potential radioactive releases to the environment from accidents with core melt, also in the long term, by following the qualitative criteria below:

    - Accidents with core melt, which would lead to early or large releases have to be practically eliminated;

    - For accidents with core melt that have not been practically eliminated, design provisions have to be taken so that only limited protective measures in area and time are needed for the public (no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption) and that sufficient time is available to implement these measures.”

It is required that the likelihood of severe accidents with core melt (or severe core degradation) which would lead to early or large releases is demonstrated to be very low with high confidence or that such fault sequences are physically impossible. The practical elimination of a condition cannot be claimed solely based on compliance with a general cut-off probabilistic value. The justification should include demonstration that the conditions to be practically eliminated are duly identified and that there is sufficient knowledge of the accident condition analysed and of the phenomena involved.

To achieve the objective O3, it is expected that the off-site radiological impact of accidents with core melt only leads to limited protective measures in area and time (only limited sheltering for the public and restrictions in land use have to be imposed) for accident sequences which are not practically eliminated. Design targets should be set on application of limited measures.

The position of the WENRA (RHWG) on the concept and application of "practical elimination" is partially developed in /WEN 13/, notably: *"Accident sequences that are practically eliminated have a very specific position in the Defence-in-Depth approach because provisions ensure that they are extremely unlikely to arise so that the mitigation of their consequences does not need to be included in the design. The justification of the "practical elimination" should be primarily based on design provisions where possible strengthened by operational provisions (e.g. adequately frequent inspections). All accident sequences which may lead to early or large radioactive releases must be practically eliminated."*

Based on the considerations above, O3 is applicable to SMRs with reasonable interpretations depending on the SMR technology. Notably, objective O3 is consistent with the practical elimination requirements in the EU safety directive 2014/87/EURATOM /EUR 14/ and with the Vienna Declaration on Nuclear Safety of the IAEA /IAEA 15/.

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

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

Safety objective urges the applicant for an enhanced independence of defence-in-depth (DiD) levels.

Safety objective O4 is a priori technology neutral, thus directly applicable to SMRs as far as reasonably practicable.

RHWG discussed that whether, a new level of defence should be defined for the control of multiple failure events, because safety systems which are needed to control postulated single initiating events are postulated to fail and thus another level of defence should take over. However, the single initiating events and multiple failure events are two complementary approaches that share the same objective: controlling accidents to prevent their escalation to core melt conditions. Hence, in WENRA it has been proposed to treat the multiple failure events as part of the 3rd level of DiD, but with a clear distinction between means and conditions /WEN 13/. It is required that provisions to control failure events on different DiD levels should be independent from each other as far as possible in NPP designs. It is important to note that this also applies to essential support systems to such provisions, and that independence needs to be demonstrated for specific fault sequences.

Table 1 Defence in Depth approach by WENRA (RHWG) /WEN 13/

DiD Levels	Objective	Essential means	Radiological consequences	Associated plant condition categories
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of	No off-site radiological impact (bounded by regulatory	Normal operation

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

O5. Safety and security interfaces

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

Safety objective O5 is technology neutral, thus directly applicable to SMRs.

O6. Radiation protection and waste management

- Reducing, as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities:
    - Individual and collective doses for workers;
    - Radioactive discharge to the environment; and
    - Quantity and activity of radioactive waste.

Regarding individual radiation dose limits for workers and the public, such limits are introduced in the EU regulations for radiation protection. While the WENRA document refers to Directive 96/29/Euratom, it has been replaced by a new Directive 2013/59/Euratom /EUR 19/. For design purposes, WENRA RHWG considered that lower design targets should be set up at the design stage /WEN 09/.

Thus, safety objective O6 is directly applicable to SMRs.

O7. Leadership and management for safety

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

Safety objective O7 is technology neutral, thus appears to be applicable to SMRs. However, the organizational capabilities of designers, vendors, and operators of SMRs might be different to what we are accustomed to. There might be a much wider range of entities involved in the whole regulatory process (designer, vendor, manufacturer, outsourced companies, newcomer utilities, etc.). This may lead to new challenges for achieving the effectiveness of the management for safety pursued by O7 and is an aspect worth considering for SMRs.

These safety objectives call for an extension of the safety demonstration for new reactors, in consistence with the reinforcement of the defence-in-depth. Some situations that has been considered as “beyond design basis” for existing reactors, such as multiple failures conditions and core melt accidents, are considered within the design process for new plants, hence the name “design extension conditions”.

RHWG considers useful that the Safety Authorities define quantitative “safety targets” in order to drive in depth technical discussions with the applicants /WEN 10/. RHWG stated that safety targets are not to be used as “standalone acceptance criteria”.

2.4 Review of ENSREG position with respect to SMRs

European Nuclear Safety Regulators Group (ENSREG), established by a Commission Decision in July 2007, is an expert advisory group of senior representatives from each Member State with competence in the areas covered by ENSREG. ENSREG’s main goal is to improve continuously nuclear safety in the operation of nuclear facilities and to promote and adopt the relevant regulatory framework in order to deploy an adequate safety policy.

All Member States that operate or will operate nuclear installations follow the basic international principles for ensuring nuclear safety and the safe management of radioactive waste and spent fuel. These principles are established in the EU Conventions and Nuclear Safety Directives. Thus, ENSREG’s role is to help to establish the conditions for continuous improvement and to reach a common understanding in these areas based on EU legislation.

Furthermore, ENSREG is working with WENRA (especially with RHWG). WENRA provided extensive technical support to ENSREG, developing different technical specifications (for instance, for the stress test after the Fukushima disaster), papers, statements, guidance, etc. that are worthwhile for ENSREG.

Finally it could be concluded, that ENSREG activities to supervise a nuclear regime in both the Member States and EU neighbouring countries are based on the international legally binding safety conventions and EU legislation, as well as on international guidance and regulatory methodologies developed under the auspices of international bodies such as the IAEA (safety standards, codes), OECD's Nuclear Energy Agency and WENRA (safety reference levels). Regarding new reactors it is endorsed by the statement in the ENSREG Position paper on the Instrument for Nuclear Safety Cooperation /ENS 14/:

ENSREG "Considers that the INSC should as a priority support third countries to:

- …
- Adopt periodic safety review mechanism and benchmarking of their practises against the WENRA reference levels and WENRA safety objectives and positions for new reactors, as appropriate, …"

Any statements and/or special requirements regarding SMRs have not been found in the ENSREG documents.

2.5 French nuclear legislation with respect to SMRs

The Order of 7th February 2012 setting the general rules relative to basic nuclear installations – with complementary guidance provided in a guide on the Design of Pressurized Water Reactors /ASN 17/[1] – requires that the licensee applies the principle of defence in depth, which consists in deploying sufficiently independent levels of defence aiming at:

- First level: preventing incidents;
- Second level: detecting incidents and applying measures that will firstly prevent them from leading to an accident, and secondly restore a situation of normal operation or, failing this, place and maintain the installation in a safe condition;
- Third level: controlling accidents that could not be avoided or, failing this, limit their aggravation by regaining control of the installation in order to return it to and maintain it in a safe condition; this level is split between level 3a - prevention of fuel meltdown in the "design reference envelope" - and level 3b - prevention of fuel meltdown in the "design extension envelope", consisting in more complex sequences of events;
- Fourth level: managing accident situations that could not be controlled so as to mitigate the consequences, especially for humans and the environment.

Furthermore, a fifth level of defence in depth targeting emergency management by the public authorities aims at limiting the radiological consequences of radioactive releases that could result from accident conditions. Specific design measures shall be planned for in this respect. These levels of defence shall be sufficiently independent to meet the installation objectives.

It is worth noting that the principle of defence in depth is formulated in such a way that it is applicable as is to SMRs, except for the "fuel meltdown" part which might not be relevant for specific technologies. However, understanding "fuel meltdown" as "severe core degradation" makes it applicable.

[1] Only published in French as of September 2019; translation is in progress.

Regarding NPPs, the guide /ASN 17/ recommends the adoption of safety objectives consistent with the Council Directive 2014/87/Euratom /EUR 14/ and WENRA ones /WEN 09/. First, the number of incidents and the possibilities of accidents occurring shall be minimised and, in the event of incidents or accidents, the releases of radioactive or hazardous substances or the hazardous effects, and their impacts on human and the environment, shall be limited to levels that are as low as practicable.

Moreover:

- for accidents without fuel meltdown, the radiological consequences shall not lead to the need to implement population protection measures;
- accident situations with fuel meltdown which could lead to significant radioactive releases that develop too rapidly to allow deployment of the necessary population protection measures in due time shall be rendered physically impossible or, failing this, extremely unlikely with a high degree of confidence
- the population protection measures that would be necessary in the event of the other accidents with fuel meltdown shall be very limited in terms of extent and duration.

Considering the safety objectives and the principle of DiD mentioned above, large releases and early releases should be avoided:

- by defining provisions allowing to significantly limit the consequences of severe accident situations when feasible;
- and by “practically eliminating” severe accident situations where it appears to be impossible to define such provisions or to demonstrate their adequacy with the knowledge and techniques available at the time of the design orientations.

The justification of “practical elimination” shall preferably rely on the physical impossibility of the situation. Where this is not possible, the applicant shall demonstrate with a high degree of confidence that the situation is extremely unlikely.

This regulatory context is fully applicable to SMRs; as of today, there is no intent to modify or develop any new regulation.

2.6 German nuclear legislation with respect to SMRs

The national rules and regulations regarding safety directives and good practices on safety assessment of nuclear reactors in Germany (here: especially PWR and BWR) are based on the “Sicherheitsanforderungen an Kernkraftwerke” /SAK 12/ (“Safety Requirements for Nuclear Power Plants”) from 22.11.2012 in its revised version from 03.03.2015 promulgated by the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety. An English translation can be found in /BFS 15/.

All NPPs within Germany must fulfil these requirements. No special regulations are made for SMRs. On a high level, the German Safety requirements are consistent with EU Safety Directives and the WENRA (Objectives O1 to O7) and ENSREG positions detailed above. In particular, German regulation requires demonstration of practical elimination, continuous improvement of the safety of the plant as far as is reasonably practicable and independence of levels of defence in depth.

Considering the German phase-out for nuclear power for the production of electrical power, new-build projects of (most) SMRs in Germany are forbidden by law. Consequently, no plans to modify the regulatory framework for SMRS are known. However, in the following we provide some information about passive safety systems as well as multi-unit plants within the German regulatory framework. Both issues are of high relevance to SMR developers (some SMR concepts are planned as multi-unit plants and many designs use passive safety systems intensively).

Within the German safety requirements, “a system part is passive if there will be no change in its positioning in case of challenge (e.g. pipes, vessels, heat exchangers). Self-acting system parts (functioning without external power or remote control) shall be considered as passive if the position of the system part under consideration (e.g. safety valve or check valve) is not changed in the course of fulfilling its intended function.” /BFS 15/ This definition can be found in Annex 1 of the German safety requirements /SAK 12/. Passive safety systems should be preferred to active systems to ensure a sufficient reliability of systems, which are associated to Level 3 of the defence in depth concept (see 3.1 (3) /SAK 12/). Furthermore, passive systems should be preferred to cope with external and internal hazards or emergency conditions which could influence the intended functions of other safety systems (see 2.4 (2) and Annex 3 Section 2 (2) /SAK 12/).

Notably, the passive systems must also fulfil the single failure concept (Section 3.1 (7) and Annex 4 Section 2.1 (1) /SAK 12/). The following exceptions are mentioned within Annex 4 Section 2.5 (1) /SAK 12/: A single failure shall not be assumed, if it is demonstrated that the passive system is designed to withstand the maximum stresses to be expected in all cases of requirement to be assumed for them, taking into account the foreseeable changes in the material properties during the operating period with sufficient safety margins, are manufactured from a material suitable for the intended purpose and are “manufactured, assembled, erected, tested and operated based on a comprehensive quality assurance” /BFS 15/ so that sufficient reliability is ensured. The measures to be applied and the safety surcharges shall also be determined in accordance with the safety significance of the safety equipment.

Furthermore, in Annex 3 Section 3.2.8 (1) /SAK 12/, it is mentioned that internal hazards occurring in one unit of a multi-unit plant shall not lead to an inadmissible impairment of the safety of the other units.

2.7 Lithuanian nuclear legislation with respect to SMRs

Ignalina Nuclear Power Plant (INPP) is the only existing nuclear installation in Lithuania. It consists of two Units with RBMK-1500 type reactors which are water-cooled, thermal neutron with graphite moderators, pressure-tube type boiling-water reactors. Both Units are permanently shutdown since 2004 and 2009 respectively. The remaining nuclear fuel is stored in the spent fuel storage pools of both units. Unit 1 and 2 are defueled since December 2009 and February 2018 respectively. It is planned to remove all spent fuel from storage pools of both units till 2022. In overall, safety of the INPP is being ensured and maintained in accordance with design and regulatory requirements.

The advisory referendum on supporting the construction of the new Visaginas NPP in Lithuania was held in the middle of October 2012. The results of the referendum revealed the prevailing negative public opinion on the construction of a new NPP. The revised National Energy Independence Strategy does not foresee the development of nuclear power in Lithuania.

In accordance with national legislation and international conventions and treaties, the Republic of Lithuania undertakes appropriate measures to ensure the safety of nuclear installations under its jurisdiction through the establishment of legal framework and infrastructure necessary to maintain an effective nuclear safety regulatory system.

2.7.1 Primary legislative framework

The main laws regulating nuclear energy are: Law on Nuclear Energy /LIT 19/, Law on Nuclear Safety /LIT 19a/, Law on Radiation Protection /LIT 19b/ and Law on the Management of Radioactive Waste /LIT 19c/.

The Law on Nuclear Energy sets the general legal basis for activities involving nuclear materials, for other area of nuclear power involving sources of ionising radiation and for management of nuclear fuel cycle materials, including radioactive waste, managed at a nuclear installation. Regulation and supervision of nuclear safety, radiation protection and safety of radioactive waste management in the area of nuclear energy is carried out under this Law, the Law on Nuclear Safety, the Law on Radiation Protection and the Law on the Management of Radioactive Waste.

The Law on Nuclear Safety, among other provisions, establishes a procedure for issuing licenses, permits and other types of authorization, including main documents required and conditions to be fulfilled for granting authorization. This law also establishes the main principles for safety assessment and provides for different types of enforcement measures, including economic sanctions (penalties) for the most severe cases of noncompliance with safety requirements.

Both the Law on Nuclear Energy and the Law on Nuclear Safety were amended in September 2017 in order to transpose Council Directive 2014/87/Euratom of 8 July 2014 /EUR 14/. The amendments:

1) Set nuclear safety objectives, established by the Amendment to the Nuclear Safety Directive /EUR 14/, which are mandatory for the design of new nuclear facilities and are set as an endeavour for nuclear facilities already under construction or in operation;

2) Improved requirements for the periodic safety review of nuclear facilities and set procedural aspects of the regulatory review of periodic safety review report;

3) Extended regulation related to public communication and public participation in key decisions on nuclear power;

4) Extended regulation of the organization of international peer reviews.

The Law on Radiation Protection establishes the legal basis for radiation protection, enabling protection of people, subject to occupational, medical and public exposure, and the environment from the harmful effects of ionizing radiation. The law establishes an authorization system for the use of radioactive materials and radiation sources and prescribes general rules for their use. The law also provides powers and responsibilities to the authorities in this field.

The Law on the Management of Radioactive Waste establishes the rights, duties and functions of the state executive and supervisory authorities and of persons and legal entities involved in radioactive waste management.

These Laws were amended recently in order to transpose updated statements of EU Council Directives and Directives.

2.7.2 Secondary legislation for nuclear safety

Resolution on the approval of Rules of Procedure for Review of National Nuclear Safety Regulation System and Evaluation of Nuclear Installations' Safety /LIT 17/ establishes the procedures for review of national nuclear safety regulation system in order to ensure streamlined implementation of peer review responsibilities set forth by the Amendment to the Nuclear Safety Directive /EUR 14/.

The Law on Nuclear Energy /LIT 19/ provides the mandate to VATESI (State Nuclear Power Safety Inspectorate of Lithuania) to draft and approve the requirements and rules for nuclear safety, radiation protection in the area of nuclear energy, accounting for and control of the nuclear materials, physical protection of nuclear materials the quantity of which exceeds the well-defined quantity. These requirements and rules are mandatory to all the state and municipal authorities, also to all the persons engaged in such activities.

Requirements or regulations only related to or applicable to the deployment of light water SMR are presented below. Other VATESI documents are applicable for RBMK reactor and the management/treatment of generated radwaste.

Nuclear Safety Requirements BSR-2.1.3-2010 “General requirements on site evaluation for nuclear power plants” /VAT 10/ based on IAEA Safety Requirements No. NS-R-3 “Site Evaluation for Nuclear Installations” and best international practice. The regulation sets the main requirements for site evaluation, as well as proposals to use IAEA standards and guides for more detailed site’s hazards analysis.

New Nuclear Safety Requirements BSR-1.8.5-2018 “Commissioning of Nuclear Facility” /VAT 18/ for commissioning of all types of nuclear facilities were adopted by the Head of the VATESI in 2018. The main goal of the Nuclear Safety Requirements was to streamline provisions on commissioning gathering them in one document. These Nuclear Safety Requirements include content of the commissioning programme and requirements for its implementation, organization and management of commissioning of nuclear facilities, requirements for commissioning tests, requirements for verification of operation procedures, including emergency preparedness, during implementation of commissioning programme, requirements for commissioning tests and commissioning programme reports. It also includes provisions on operating, emergency operating and emergency preparedness procedures, applicable for nuclear power plants with pressurized or boiling light water reactors and pressurized heavy water reactors, as required by Amendment to the Nuclear Safety Directive /EUR 14/.

Nuclear Safety Requirements BSR-1.4.4-2019 “Use of the Experience of the Individuals Operating in the Nuclear Energy Sector” /VAT 19/ were approved by the Head of VATESI in 2019, two old documents. These Requirements BSR-1.4.4-2019 establish provisions on the use of experience of the individuals operating in the nuclear energy sector and reporting of unusual events as well as their analysis, updated pursuant changes in legislation and VATESI’s regulatory experience.

Nuclear Safety Requirements BSR-1.8.6-2019 “Maintenance, Surveillance and In-service Inspection of Nuclear Facility’s Structures, Systems and Components Important to Safety” /VAT 19a/ were approved by the Head of VATESI in 2019. These Nuclear Safety Requirements gather requirements for maintenance, surveillance and in-service inspection within one document and is applicable to all nuclear facilities.

Amendment of Nuclear Safety Requirements BSR-1.4.2-2014 “Management of Construction of Nuclear Facility” /VAT 14/ was approved in January 2017 by the Head of VATESI. It introduces more detailed requirements for program of tests and inspections of safety important structures, systems and components performed during construction of nuclear facilities, and for transfer of safety important structures, systems and components from construction to commissioning stage.

Nuclear Safety Requirements BSR-1.8.3-2017 “Technical Specification of Nuclear Facilities” /VAT 17/ were adopted by the Head of VATESI in 2017. The main objective of the requirements is to establish regulation measures allowing transfer of safety important information from site evaluation stage to design stage. Requirements set the content of the technical specification of nuclear facilities and specify requirements and information that must be implemented in the design of nuclear facilities.

Regulatory requirements in the area of operating experience feedback are established in the VATESI`s Nuclear Safety Requirements BSR-1.8.1-2010 “Requirements of notification on unusual events in nuclear power plants” /VAT 10a/ and P-2009-04 “Requirements on Operating Experience Feedback in the Field of Nuclear Energy” /VAT 04/.

In 2018, new Nuclear Safety Requirements BSR-2.1.6-2018 “Design of Nuclear Power Plant” /VAT 18a/ was approved by Head of VATESI. The regulation set the main requirements for design and design principles of NPPs and their separate buildings, parts of buildings and structures, systems and components, which are used for energy production, management of nuclear incidents, nuclear and radiological accidents, emergency preparedness. Based mainly on IAEA requirements SSR 2/1 (Rev. 1) /IAEA 16/, includes provisions from WENRA documents /WEN 14/, /WEN 09/, ENSREG "stress tests" specification, prepared reacting to Fukushima Daiichi accident, and includes provisions from Nuclear Safety Directive and its amendment /EUR 14/. In accordance with regulation, in designing NPP shall be assessed all external natural and human induced events which have been identified as having potential impact on NPP safety during site evaluation. Moreover, regulation includes provisions for large commercial aircraft crash impact assessment which are based on WENRA position on Intentional crash of a commercial aircraft – despite low probability of large commercial aircraft (maximum take-off weight 200 t) crash this event shall be considered in the design of NPP. Nevertheless, regarding SMRs the following is stated in BSR-2.1.6-2018: "*These Requirements* ***do not apply*** *to the design of nuclear power plants with integrated pressurized-water reactors and* ***small modular reactors***".

The common requirement for application of defence-in-depth in design, commissioning and operation of nuclear installation stages are introduced in the amendment to the Law on Nuclear Safety /LIT 19a/ (Article 35, #1). In accordance with the law, the specific requirements shall be set by VATESI. And the issues for implementation of the defence-in-depth concept at all stages of safety related activities (including design and construction) for new NPPs comprehensively introduced in regulation Nuclear Safety Requirements BSR-2.1.6-2018 "Nuclear Power Plant Design" /VAT 18a/ (again, not for PWR and SMR). The corresponding provisions are set according to WENRA safety objectives for new NPP /WEN 09/, taking into account Amendment to Nuclear Safety Directive /EUR 14/, IAEA specific safety requirements SSR-2/1 "Safety of Nuclear Power Plants. Design" /IAEA 16/ and other IAEA recommendations and best international practice. The principle of defence-in-depth should be applied in all stages of safety-related activities. During normal operation, all barriers and all means designed to protect them must be in proper operating condition. If any of the barriers provided in the plant design or any of the means intended to protect those barriers (in the frames of justified conditions of safe operation) are found to be out of order, operation at power is not permitted. The extent, to which the various safety functions are to be implemented, is specified in norms and technical requirements, and for each individual plant shall be stated and justified in the technical design.

The common requirement for application of proven engineering practices in design and construction of nuclear installation stages are introduced in the amendment to the Law on Nuclear Safety /LIT 19a/ (Article 35, #2).

Finally, it should be concluded that there are no specific detailed requirements for design, commissioning and operation of SMRs in the Lithuanian legislation framework, but common principles, primary legal basis and provisions for nuclear safety are in line with international approved regulatory frameworks and approaches within Europe.

2.7.3 Assessment of SMR application

The Lithuanian government is seeking to attract investment in the development of new cogeneration power plant projects in most populated areas, as it looks to deliver affordable heat to residents of both cities. These projects are recognised as economic projects of national significance. It should be noted, that these plans are related to the employment of combined heat and power plants. Although heat supply sector is large and consisting of many district-heating systems, the Lithuanian district heating network is well developed and a very attractive option could be the deployment of Small (Medium) type nuclear reactors at new sites close to cities with large heat demand, like Vilnius or Kaunas, where district heating supplies heat for 80% of the buildings. In general, SMRs could be considered for small countries like Lithuania, as an alternative to large nuclear units due to limitations by the grid size and available financial resources. Such analysis was performed during execution of the IRIS project in which LEI took part.

Different scenarios of future energy system development were analysed in this study from 2008 to 2012 /ALZ 12/. Economic modelling and optimisation were concentrating on evaluation of possibilities to construct a new energy source. In this study, the introduced approach was applied focussing on SMRs, which could be one of the future energy source options in Lithuania. As example SMR, the IRIS (International Reactor Innovative and Secure) advanced modular nuclear reactor was chosen for the analysis.

The IRIS is a pressurised light water cooled, medium-power (335 MWe) reactor /CAR 03/, /CAR 05/. The IRIS design supports the idea of licensing the power plant with reduced or even without the need for off-site emergency response planning /ALZ 08/. This would allow IRIS to be treated as any other industrial facility, located close to population centres. This allows better implementation of cogeneration option and reduction of transmission costs.

Possible introduction of IRIS nuclear units into Vilnius' and Kaunas' district heating systems was analysed in the study. According to study /ALZ 12/ IRIS reactor could be built only for electricity generation at the existing site of Ignalina NPP in order to utilise existing infrastructure and to lower construction costs. The heat demand in this region is very small (population of Visaginas town near Ignalina NPP site is 20 000 only, no large industry is located here).

The results of the study showed that construction of SMR (IRIS) units is very attractive option (looking from economical point of view) for the future electricity generation in Lithuania. The scenario with IRIS cogeneration units in the two biggest cities (Vilnius and Kaunas) caused lowest total discounted cost compared to the case without nuclear energy source. Depending on initial conditions, up to five IRIS units could be built in the Lithuanian energy system in the period up to 2025, and total installed capacity of IRIS units could be up to 1.7 GW. IRIS reactors can cover up to 50 % Lithuania's electricity market and 51-75 % of district heating markets in these two cities. In case IRIS cogeneration units are installed outside existing district heating networks (due to Emergency Planning Zones), the attractiveness of these units is decreasing gradually with distance, because of investment cost and heat losses in additional district heating pipelines.

2.8 Other nuclear legislation with respect to SMRs

2.8.1 Canada

The regulatory framework in Canada is well established and based on decades of operating experience. It uses a combination of performance-based requirements and guidance.

More than 10 SMRs designs are currently being reviewed by the Canadian Nuclear Safety Commission (CNSC) through an optional process called Vendor Design Review (VDR). This process, detailed in REGDOC-3.5.4 /REG 18/, consists of three phases, from assessing the vendor's proper understanding of the Canadian regulatory framework to assessing a preliminary safety report with a focus on selected topics. It is worth noting that the VDR is not specific to SMRs and flexible enough to be applied to any reactor.

This optional phase is followed by the licensing and compliance process, described in several REGDOCs. This process is divided in several steps, each ending in delivering a licence for site preparation, construction, operation and decommissioning. However, a new document, REGDOC-1.1.5 /REG 19/, was published in August 2019 to give additional guidance for SMR applicants. It promotes the usefulness of the VDR described above and proposes a wide range of considerations that may be useful for SMRs. Some of these considerations are the management system, emergency management and fire protection, waste management, etc.

Finally, the central document for reviewing the safety case is REGDOC-2.5.2, Design of Reactor Facilities /REG 14/. This is where the Canadian regulation defines its safety approach, from the global safety objectives and concepts down to the engineering principles and system-specific requirements. This REGDOC is applicable to SMRs for two main reasons:

- It allows the use of a graded approach through requirements commensurate with the relative risks;
- It allows the applicant to propose “alternative approaches … where:
    - the alternative approach would result in an equivalent or superior level of safety;
    - the application of the requirements in this document conflicts with other rules or requirements;
    - the application of the requirements in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose.”

2.8.2 United States

The top-level US regulations are broad enough to encompass all reactor designs. However, the existing regulatory guidance propose prescriptive methods and processes that may not be fully adapted to SMRs, especially non-light water-cooled designs. In order to complete their existing guidance to this extent, the US NRC is currently developing an “Enhanced Safety Focused Review Approach” (ESFRA) aiming at aligning the review focus and resources to risk significant SSC and other aspects of the design that contribute most to safety. The ESFRA is detailed in a chapter of the US Standard Review Plan that has been updated with a SMR focus /NUR 14/.

To complement this top-level development, the US NRC had to develop a Design Specific Review Standard for NuScale /NUS 17/, the light water SMR currently under review. This document outlines the sections of the safety review process that are different because of design specificities of NuScale compared to a large-scale reactor. There is a combination of areas of reduced scope review (auxiliary systems, offsite power, etc.) and areas where the review is augmented (containment integrity, reactor systems, etc.).

However, the development of design specific reviews on a broader scale might be impractical. For this reason, the US NRC is currently developing a new framework for regulatory processes for advanced reactors, through what they call a Licensing Modernization Project (LMP). The main objective of this LMP, explained in NEI 18-04 /NEI 19/, is the development of a systematic process for identifying “licensing-basis” events and classifying SSC. The US NRC is currently applying this process on several pilots, all non-light-water technology. However, it is worth noting that the proposed approach is technology neutral.

2.8.3 United Kingdom

The UK regulatory philosophy is based on a risk-informed framework with mostly non-prescriptive high-level goals /ONR 17/. This approach is technology-neutral and does not need to be changed in order to regulate SMRs, even of new technologies. The regulatory basis is laid out by the UK Office for Nuclear Regulation (ONR) in its Safety Assessment Principles for Nuclear Facilities /ONR 14/ (SAPs). The SAPs have been updated in 2014, benchmarked against the WENRA and IAEA positions and take into account the applicable EU safety directives. On a high level, the WENRA objectives O1 to O7 are reflected in the SAPs and practical elimination of severe accidents should be demonstrated.

Importantly, the SAPs define numerical targets for on- and offsite risk based on dose limits and frequency thresholds /ONR 14/. For each regulatory requirement, a (higher-risk) basic safety level and a (lower-risk) basic safety objective is defined. It is incumbent on a vendor or operator to demonstrate the basic safety levels are met and that all reasonably practicable measures have been implemented to achieve the basic safety objectives. At the same time, core principles like leadership and management for safety, optimisation of protection, prevention of accidents, inherent safety, defence in depth have to be fulfilled /ONR 14/.

With regard to SMRs, the overarching approach that all reasonably practicable measures need to be implemented gives some lee-way for alternative solutions for specific SMR designs. At the same time, relevant good practice needs to be taken into account and such practice comes from existing (larger) NPPs and also research reactors, so changes might be gradual.

Some principles in the extant version of the SAPs are a bit more specific, and the ONR are in the process of reviewing them in order to update them and develop new ones to broaden their scopes.

Unfortunately, no additional information is publicly available as of today.

2.8.4 Russian Federation

The Russian Federation legal and regulatory framework allows for the use of a graded approach for SMRs, making it broadly applicable. To our knowledge, there is no intent to develop specific guidance for SMRs.

However, it is worth noting that there are some differences in the regulatory approach for floating reactors due to their specificities. Since the Russian Federation already has some experience with this kind of reactors, and with the Akademik Lomonosov one facility in operation since 2019, their regulatory framework for floating SMRs is well established through a complete set of rules and guidance documents that tackle the specificities of such reactors.

2.8.5 Additional Information/Answers to questionnaire

Within the frame of T1.1 of ELSMOR, also a questionnaire was started. Regulators of different countries were asked the following questions:

1. Do you have any activities regarding licensing of SMR in your country and/or your organisation? If so, please provide some information (e.g. design name, planned location, how many modules, etc.).

2. Do you have any SMR specific requirements regarding licensing or assessment in your national rules? If yes, please summarise these requirements and/or give some references.

3. Do you intend to change your national rules in order to license or assess SMR more easily? If yes, please summarise these changes.

All in all, three organisations have answered the questions in time: STUK (Finland), ONR (UK) and SUJB/SURO (Czech Republic). The answers are given in the following:

1.	Do you have any activities regarding licensing of SMR in your country and/or your organisation? If so, please provide some information (e.g. design name, planned location, how many modules, etc.).
STUK	No.
ONR	Since October 2017, ONR has implemented a programme to further develop its capability and capacity of to support and regulate the development of Advanced Nuclear Technologies (ANTs), including Small Modular Reactors (SMRs), in preparation for potential future developments in the UK (at this stage there are not specific ANT designs underdoing licensing/ regulatory processes). The programme of work has the following objectives: • Develop ONR capability and capacity to regulate ANTs.

	• Review ONR's guidance and processes to ensure that they are fit for regulating ANTs. • Provide advice to BEIS' Advanced Modular Reactor (AMR) feasibility and development programme. • Increase engagement with international regulators. • Engage with the ANT industry
SUJB/ SURO	At present we do not carry out specific concrete activities regarding licensing of SMR in the Czech Republic.
2.	Do you have any SMR specific requirements regarding licensing or assessment in your national rules? If yes, please summarise these requirements and/or give some references.
STUK	No. In general, the safety requirements in the regulation and the licensing process are applied to all facilities.
ONR	Our regulatory requirements (e.g. regulatory guidance such as SAPs, SyAPs & TAGs) are in general technology neutral and non-prescriptive and therefore applicable to SMRs.
SUJB/ SURO	We have a new complete nuclear legislation package (2017) for licensing nuclear power plants and a legal possibility to apply graded approach in assessing nuclear safety features of nuclear facilities. We also discuss the level of applicably of the currently valid legislation ot SMRs.
3.	Do you intend to change your national rules in order to license or assess SMR more easily? If yes, please summarise these changes.
STUK	Yes. The changes are not defined yet, but we intend to check the safety requirements and make the necessary changes so that the requirements will be technology neutral and suitable to SMRs also concerning other aspects than technology (e.g. organisation, emergency preparedness etc.).
ONR	ONR is undertaking focused reviews of our guidance (SAPs, SyAPs & TAGs) to check its adequacy and sufficiency for regulating ANTs. In the first stage of this work we considered safety analysis, engineering and security of water cooled SMRs. In general, we have concluded that our SAPs, SyAPs and TAGs are suitable to regulate these types of reactors; However, some recommendations to expand our assessment guidance providing enhanced focus on the following topics have been raised from this work and are under consideration: • passive safety • thermal-hydraulic phenomena that may be relevant to the assessment of natural circulation within passive systems

	• exclusion of faults from design basis considerations on the basis of natural physical phenomena • multi-module PSA • consideration of relevant good practice for advanced reactors • methods and assurance requirements of new materials used for the construction • categorisation and classification of structures, systems and components in order to cover the challenges associated with indirect consequences of failure and limited access • design of cooling pools which are used by multiple reactors. The design considerations of sharing the cooling water between the fuel pool and the reactors pool will also be included in the guidance. • novel mechanical engineering features (e.g. Internal CRDM, Primary coolant pump design, Valve design Mechanical Handling) and the design may require enhanced attention • manufacture and supply chain, construction programmes (hold points), site justification (e.g. construction adjacent to existing nuclear sites) and phased deployment • Nuclear Safety Culture incorporating unique features of the SMR model (e.g. knowledge transfer from Vendor to the future Licensee) Similar reviews will consider the applicability of our regulatory guidance to Advanced Modular Reactors (AMRs). AMRs are a sub-set of ANTs, which cover a wide range of potential nuclear reactor technologies within the scope of the Generation IV Forum (GIF) technology roadmap. They involve molten metal or salt, high temperature gas or supercritical water as coolants, and are generally very different to the reactors we currently regulate.
SUJB/ SURO	At present we discuss the level of applicably of the currently valid nuclear legislation for the eventual future licensing of the SMRs.

2.9 Summary

The review of the safety directives given by the European Union, IAEA, WENRA, ENSREG as well as by selected countries (EU and non-EU) showed, that the regulation can be basically applied also for LW-SMR. The EU safety directives establishes a high-level framework, in which the member states can create their own regulations. This framework is technological neutral. Within the IAEA regulations, no explicit guidance for SMR is given up to now, but the current documents can be used for SMR. This is also the case for ENSREG and WENRA. The objectives given in /WEN 10/ apply also for SMR: low frequencies for accident without core melt, practical elimination for accident with core melt (or implementation of measures to limit consequences), independency of DiD levels and radiation protection under the concept of ALARP (as low as reasonable possible).

The nuclear regulations in the related countries are mainly also applicable for LW-SMR. Some countries have non-prescriptive approaches (like UK or Canada). In the US, the regulations are more prescriptive. This is a problem for licensing a SMR. In this case, design specific reviews can be performed, which is unpractical when a larger number of designs need to be assessed. Therefore, the regulations will be modified in the future, which other countries plan also to do.

3. Review of current LW-SMR designs

A large number of light water small modular reactor (LW-SMR) designs are currently under development. Within T1.2 of the ELSMOR project, the most advanced concepts have been screened to get an overview about their general designs and more specifically about their safety features. The designs have been screened regarding the following items:

- Reactivity control
- Decay heat removal
- Containment integrity
- Decommissioning
- Refuelling, spent fuel management, transport and disposal
- Multi-unit site and sharing of systems issues
- Severe accident management and emergency planning
- Operation and human factors

Depending on the information available the descriptions of the above-mentioned items might differ in their specific information content. The gathered data has further been used to classify the different safety systems into a defence-in-depth concept, even if it was not explicitly given in the available materials.

Used safety systems and features have been summarized in tables regarding their intended function to see which safety features are provided by which concepts. These tables are arranged in section 3.2 of this report. In the appendix of the report the different design descriptions can be found.

The data in this report is intended to be used for task T1.3 in which specific potential challenges for safety demonstration methodologies as well as for simulation tools are identified, but it can also be of value for WP3 and WP4 for creating the PIRTs.

3.1 Overview about considered SMR designs

In the following Table 3.1 in ELSMOR considered SMR designs are listed. Only light water SMRs have been screened. Besides on-shore SMR, also designs considered for deployment on or below the sea (on a barge, icebreaker, subsea level) were reviewed.

Table 3.1 Screened SMR designs, alphabetically ordered

Name	**Developer**	**Country**	**Power [MW_{el}]**	**Planned site**
ACP100	NPIC, CNNC, CNPE	China	125	Zhangzhou, Shangrao, Ganzhou, Hunan, Jilin
ACPR100 ACPR50S	CGN	China	140 (ACPR100) 60 (ACPR50S)	n/a
BWRX-300	GE Hitachi Nuclear Energy	USA	300	n/a
CAP200/150/50	SNERDI	China	> 200	n/a

Name	Developer	Country	Power [MW_{el}]	Planned site
CAREM	CNEA	Argentina	31	Atucha
DHR-400	CNNC	China	- (400 MW_{th})	Xudapu, Liaoning
Flexblue	DCNS	France	160	Cancelled
IMR	MHI	Japan	350	n/a
IRIS	ENEA, SIET, CIRTEN	Italy	335	n/a
KLT-40S	OKBM Afrikantov	Russia	35	Akademik Lomonosov (Barge)
mPower	B&W	USA	195	Cancelled
NuScale	NuScale Power, Fluor	USA	50	n/a
NUWARD	EdF, Technicatome, CEA, Naval Group	France	170	n/a
RITM-200 RITM-200M	OKBM Afrikantov	Russia	50	n/a
Rolls-Royce SMR	Rolls-Royce	UK	443	n/a
SMART	KAERI	Korea	100	Saudi Arabia
SMR-160	Holtec International	USA	160	n/a
SNP350	SNERDI	China	350	n/a
VBER-300 / RP	OKBM Afrikantov	Russia	325	n/a
VK-300	NIKIET	Russia	250	n/a
Westinghouse SMR	Westinghouse	USA	> 225	Suspended

3.2 Special features and safety systems implemented into different SMR designs

In the following, the special features and safety systems are summarised. Specific topics are general core properties, shut down systems, systems related to emergency core cooling (ECC), systems for depressurisation of the primary system, systems for pressure control of the containment, systems for decay heat removal, measures to mitigate external hazards, measures and systems to mitigate severe accidents, aspects regarding human factors, decommissioning and refuelling and spent fuel management.

3.2.1 Core features

In Table 3.2 the related core data based on thermal power, use of boron acid and/or burnable absorber, planned period of fuel element cycle, planned burnup as well as the mean power density.

It can be seen, that some of the concepts do without a boron system for compensation of excess reactivity, although the planned fuel cycle is long (up to 48 months in the IRIS case and 72 months in VBER-300 and VK-300 cases). Instead of boron, many concepts provide burnable absorber rods with absorber materials like Gd_2O_3, B_4C, IFBA or Er. When forgoing the boron system, space and therefore costs are saved. Additionally, the temperature reactivity coefficient of the coolant is improved.

Table 3.2 Properties of the cores of the different SMR concepts ("x"-mark means system is provided, "-"-mark means system is not provided)

Name	Power [MW_{th}]	Boron Acid	Burnable Absorber	Planned FE-Cycle [Month]	Planned Burnup [MWd/kg_U]	Mean Power density [kW/dm^3]
ACP100	385	x	x	24	< 52	n/s
ACPR50S	200	x	x	30	< 52	n/s
BWRX-300	300 MW_{el}	unknown	unknown	unknown	unknown	unknown
CAP200/150/50	660	x	x	24	37	n/a
CAREM	100	x	x	18	24	n/s
DHR-400	400	-	x	10	30	n/a
Flexblue	530	-	x	38	38	70
IMR	1000	unknown	x	26	> 46	40
IRIS	1000	x	x	48	65	36.7
KLT-40S	150	-	x	30-36	45.4	117,8
mPower	575	-	x	24	< 40	unknown
NuScale	160	X	x	24	30 – 50	unknown
NUWARD	540	-	x	24	unknown	70
RITM-200	175	unknown	unknown	54-84 M	68 – 51	unknown
Rolls Royce SMR	1276	-	x	18 – 24 M	55 – 60	unknown
SMART	330	X	x	36 M	< 60	62.6
SMR-160	525	-	x	18 – 24 M	45	unknown
SNP350	1035	x	unknown	18 M	unknown	unknown
VBER-300	917	x	x	72 M	47	21,3
VK-300	750	-	x	72 M	41.4	unknown
Westinghouse SMR	800	x	x	24 M	> 62	unknown

3.2.2 Systems and features related to emergency core cooling

Besides active systems for replenishing the coolant in the primary circuit also passive systems are used (see Table 3.3). These are mainly accumulators, core make up tanks (CMT) and elevated water tanks. The latter is mainly used at low pressures, since the pressure head of these systems is limited to the hydrostatic head. Injection at mid-pressures can be performed by accumulators, due to a pressurized gas cushion on top of the vessel. The pressure within the CMT is the primary pressure, due to connecting lines to the primary circuit. Furthermore, they are also elevated with respect to the primary system and can therefore inject coolant at high pressures. These systems are mainly used for short term replenishment during a loss of coolant accident (LOCA).

In the long term, the coolant leaked out of the primary system accumulates in the so-called sump or pit of the containment. This water needs to be fed back into the primary system. In the different SMR designs, this is performed either actively by pumps or passively. In the passive case, the sump or pit surrounds the RPV in such a way that only a small gap filled with water is formed. By opening recirculation valves implemented in the lower RPV wall and using the geostatic head of the water, a recirculation of the coolant is achieved.

In NUWARD, the main passive system RRP is used all along the transient. In addition, there are accumulators used in the short term and active water make-up system in the long term (over 3 days). Basically, in NUWARD, the systems could be classified in ”long term core cooling” and ”passively from water-wall” (cold source) with use of accumulators in addition.

Table 3.3 Systems related to emergency core cooling (see text)*

Emergency Core Cooling		Design
Accumulators		ACP100, ACPR50S, CAREM, Flexblue, IMR, IRIS, KLT-40S, mPower, NUWARD, RITM-200, VBER-300, VK-300
Active Systems		CAREM, Flexblue, KLT-40S, VBER-300, VK-300, SMR-160, RITM-200, SNP350
Make-Up-Tanks		ACP100, Flexblue, IRIS, KLT-40S, RITM-200, SMART, VBER-300, VK-300, Westinghouse SMR, CAP200
Elevated Tanks	Inside Containment	ACP100, ACPR50S, CAREM, Flexblue, IRIS mPower, SMR-160, VK-300, Westinghouse SMR, SNP350
	Outside Containment	RITM-200, SMART
Long Term Core Cooling	Passively from sump / pit	ACP100, ACPR50S, CAREM, Flexblue, IRIS, RITM-200 SMART, SMR-160, VK-300, Westinghouse SMR, NuScale, NUWARD*, DHR-400 (pool-type reactor), CAP200
	Actively from sump / pit	KLT-40S, RITM-200, VBER-300
No system for circulation needed		IMR

3.2.3 Systems and features related to primary depressurisation

The following table lists the specific systems and features to depressurise the primary circuit, besides the use of the secondary side. The primary coolant can be vented into a water pool, similar to BWR venting into the wetwell. Some concepts just vent into the containment. Steam entering it can condense on the containment structures. If the containment is immersed within a pool, the heat is transferred into the water inventory of an external pool (e.g. NuScale), otherwise transferred to the air outside.

Table 3.4 Systems related to primary depressurisation

Primary Depressurisation	Design
Venting in Pool (e.g. wetwell, etc.)	ACP100, ACPR50S, CAREM, Flexblue, IRIS, NUWARD, SMR-160, VK-300
Venting in containment	ACP100, IMR, mPower, NuScale, RITM-200, SMART, VBER-300, Westinghouse SMR
Purification and cooldown system	KLT-40S, VBER-300

3.2.4 Systems and features for depressurisation and pressure control of the containment

Pressurisation of the containment is normally caused by steam release from the primary system. To ensure containment wall integrity the pressure needs to be controlled. In the reviewed SMR designs this is done by the measures listed in Table 3.5. Some of the designs condense the steam inside a large water pool similar to a wetwell concept of a BWR. Such a pool can be located out- or inside the containment. Here, an overpressure inside the containment / drywell is needed to force the steam into the water pool.

Another possibility is to condense the steam with heat exchangers (called containment condenser in Table 3.5), e.g. tube bundles. The tube side is normally connected to a water tank. For this system, accumulation of non-condensable gases in the compartment of the heat exchangers needs to be taken into account.

The steam can also be condensed due to a spray of cold water into the containment atmosphere. Finally, condensation on cold containment walls can ensure long-term heat removal. The latter is achieved by immersion (fully or partly) of the containment into a large water pool or even the ocean (e.g. in the Flexblue case).

Table 3.5 Systems related to depressurisation and pressure control of the containment

Depressurisation of the containment	Design
Wetwell/Pool	Flexblue, CAREM, KLT-40S, VK-300
Containment condenser	ACP100, IRIS, KLT-40S, RITM-200, SMART, VBER-300
Spray into containment atmosphere	CAREM, SMART, SNP350
Condensation on containment inner wall	ACP100[2], ACPR50S, Flexblue, IMR, mPower, NuScale, SMR-160, Westinghouse SMR, CAP200, NUWARD

3.2.5 Systems and features for decay heat removal

Decay heat removal in the different designs is provided by different approaches. The first possibility is to use the secondary side to cool down the primary side. This can be done by all concepts except the VK-300, since this is a BWR. In all other cases, the main heat sink could be used. If it is not

[2] Depends on reference

available, the steam can be routed to dedicated heat exchangers, in which the steam can condense. The heat exchangers can be immersed within large water pools, the ocean or cooled by air.

The second possibility is the direct cooling of the primary system, without any use of the secondary side, either directly by a passive heat exchanger to a water pool, using an intermediate circuit (e.g. in the Westinghouse SMR concept, an intermediate circuit connects the CMT with a water pool) or other active systems.

Table 3.6 Systems related to decay heat removal

Decay Heat Removal		Design
Cooled by Secondary Side	Passively within water pool	ACPR50S, CAREM, IMR, IRIS, KLT-40S, NuScale, RITM-200, SMART, VBER-300, VK-300
	Passively on air	IMR, mPower, NuScale, RITM-200
	Passively into the ocean	Flexblue
	Passively to the condenser	mPower
	Actively by main heat sink	All (except VK-300)
Primary Side	Passively within water pool	ACP100, Flexblue, mPower[3], DHR-400
	Passively with extra circuit	SMR-160, Westinghouse SMR, NUWARD
	Actively by purification loop	RITM-200
Other active systems		Flexblue, IMR, Rolls Royce SMR

3.2.6 Systems and features to shut down the core

Table 3.7 Systems related to shut down the core

Core shut down		Design
Control rods (gravitation)		All (except ACPR50S with magnetic force)
Control rods (magnetic force)		ACPR50S
Liquid Absorber Injection	From core make up tank	IRIS, SMART, SMR-160, Westinghouse SMR
	From accumulator	CAREM, IMR, NUWARD, VK-300, VBER-300
	Other active injection	ACP100, ACPR50S, CAP200, CAREM, KLT-40S, NUWARD, Rolls Royce SMR, VBER-300, Westinghouse SMR

[3] Information taken from pictures

The measures for core shut down of the reviewed SMRs do not differ from the measures of current large LWR. All concepts are using control rods as primary shut down system which works with gravity (except for the ACPR50S). For a second line of defence, liquid absorber can be injected passively from the CMT, accumulator and/or other, but active systems.

For the time being, the system of NUWARD is composed of accumulators for short term absorber injection and an active system for medium/long term injection.

3.2.7 Systems and Measures to mitigate external hazards

The systems and measures provided by the different SMR designs to mitigate external hazards (tsunami, flooding, wind, etc.) are listed in Table 3.8. Some modules are located underground, mitigating e.g. airplane crashes, etc. When located on a ship or barge, the module might be safe from earthquakes and could be brought to safe places during storm, tsunami, etc. Finally, a location on the sea ground potentially exclude hazards like tsunami, storm, draught or flooding.

By special architectonical design of the site is meant that, for instance in the Rolls-Royce case, support buildings are located around the reactor facilitating a so-called “hazard protection barrier” /IAEA 18/ protecting the site against tsunami, airplane crashes, etc.

Table 3.8 Systems/Measures related to mitigate external hazards

Prevention from external hazards		Design
Containment	No special location	Rolls Royce SMR, Ground-based VBER-300, VK-300, SNP350
	Located underground	ACP100, DHR400, CAP200, mPower, NuScale, NUWARD, SMR-160 (partially embedded), Westinghouse SMR,
	Located onboard ship	KLT-40S, VBER-300 on FPU
	Located at the sea ground	Flexblue
Special architectonical design of the site		Rolls Royce SMR, Land based concept RITM-200

3.2.8 Systems and Measures to mitigate severe accidents

Besides other measures, some SMR concepts provide in-vessel retention of the molten corium by ex-vessel cooling of the RPV. This is mainly done by flooding of the lower cavity with water. The SMR designs providing such systems are listed in Table 3.9. Additionally, also a pressure control of the containment is needed in order to ensure the containment barrier integrity.

Table 3.9 Systems/Measures related to mitigate severe accidents

Mitigation of severe accidents	Design
Corium in-vessel retention by ex-vessel cooling	ACP100, ACPR50S, CAREM, Flexblue, IMR, NUWARD, KLT-40S, RITM-200, SMART, VBER-300

3.2.9 Human factor and shared systems aspects

Some human factor aspects could be found in the related references of the SMR designs. The first one is the use of one control room (or centre) to control multiple modules. Here specific issues arise for safety demonstration. For instance, how to avoid operator confusion during normal operation or accident conditions? How many people are needed in the control room?

Regarding the asterisk for NUWARD in Table 3.10: there is one general control room for normal operation and for accident situation in the short time and a dedicated control room for each reactor for outage situation and accidental situation in the long term. Regarding operation, the reactors are independent with dedicated control panels and dedicated operators (no crosslink of I&Cs and/or operators) in the general control room. Only support people (safety, maintenance, team leader, etc.) are shared.

Besides the control room also support systems might be shared by the modules. In NUWARD for instance, each reactor within its metallic containment is immersed in its own cooling water wall (large heat sink in accidental situation). It has its own nuclear auxiliary, safety and control systems and delivers steam to a dedicated secondary loop outside the nuclear island. Within the nuclear island, support systems are shared only if it is demonstrated that their failure has no adverse effect on safety (maintenance hall covering the two reactors, common fuel pool, rooms and means associated to the NI support systems, etc.).

A second point mentioned in Table 3.10 is the so-called “walk-away safe” design of the SMR-160, meaning that basically the reactor is maintained in a safe configuration during all DBC and DEC without any operator actions, due to reliance on passive safety systems only. Also this aspect must be carefully demonstrated.

Table 3.10 Human factor aspects (see text)*

Human Factor aspects	Design
Multiple module operation by one control centre	ACPR50S, Flexblue, KLT-40S, NUWARD*
“Walk-away safe” design	SMR-160

3.2.10 Decommissioning, refuelling and spent fuel management aspects

Only little is published regarding decommissioning, refuelling and fuel management of the different SMR concepts, meaning that even if some of the designs are not mentioned in the following tables, they might be classified to some of the items.

The floating SMRs KLT-40S and VBER-300 as well as the Flexblue are supposed to be transported to a specialised facility for decommissioning after their lifetime.

Most of the SMRs are refuelled at the site. Only in the case of Flexblue, the module is transported to a specific facility, refuelled and transported back to the site. If refuelling is performed at the site, the spent fuel is generally stored on the site.

Table 3.11 Decommissioning Aspects

Decommissioning aspects	Design
Specialized service base	Flexblue, KLT-40S, VBER-300 on FPU
Easy assembly and disassembly	SMR-160

Table 3.12 Refuelling aspects

Refuelling aspects	Design
Refuelled at site	ACP100, ACPR50S (on ship), KLT-40S, mPower, NuScale, NUWARD, SMR-160, VBER-300, VK-300, Westinghouse SMR, SNP350
Refuelled in facility	Flexblue

Table 3.13 Spent fuel management

Spent Fuel Management	Design
Fuel pool located underground	ACP100, NUWARD
Fuel stored on board	KLT-40S, VBER-300 on FPU
Fuel pool located in containment	VBER-300, SMR-160, VK-300, SNP350
Underground fuel dry storage	SMR-160

3.3 Summary

All the above mentioned design decisions for different aspects of the reactor need to be carefully investigated and the safety of the plant must be demonstrated considering the respective regulations of the specific countries and established guidelines of WENRA and IAEA mentioned in chapter 2. During the screening of the different SMR designs, differences to current large LWR occurred, which were summarised in the respective tables of this chapter.

Some of the specific design decisions can be challenging for the currently established safety demonstration methods. For instance, some of the SMR designs do without a boron system for compensation of excess reactivity. Reasons are saving space and therefore costs but also improving the coolant density reactivity coefficient. Instead of boron, burnable absorbers are extensively used. The question is now, how does this design decision impact the safety approach and the safety demonstration methods? A second point is the extensive use of passive safety systems for decay heat removal. Since they generally rely on small driving forces, more attention must be given to reliability demonstration of the respective systems.

Some ideas where to look at are given in the following chapter 4, in which potential challenges identified by experts involved in this work are summarised.

4. Potential challenges on safety demonstration methods

The features of the specific SMR designs, shown in chapter 3, need to comply with the regulatory requirements, briefly summarized in chapter 2. Safety methodologies to demonstrate the compliance are developed in ELSMOR WP2. To support this work, in T1.3 potential challenges to safety demonstration arising from the different SMR designs are identified to ensure that a homogeneous assessment will be performed in WP2. For this purpose, it was proposed to perform a qualitative and preliminary assessment of the relevance of a list of safety challenges regarding each WP2 issue, which cover the following topics:

- Reactivity control
- Decay heat removal
- Containment
- Refuelling, spent fuel management, transport and disposal and decommissioning
- Multi-unit and system sharing
- Severe accident management
- Emergency planning
- Human factors

Basically, PWR SMRs may differ from large PWR in the following major features:

- Low power
- Several modules grouped together and possibly sharing some major features as a cooling pool, a control room and possibly some support systems
- Several modules built in a sequential time frame, with units operating while some others are still under construction
- Extensive use of passive safety systems
- Innovative equipment (steam generators, control rods, main coolant pumps, etc.) or plant management process (no boron)

All these features may either simplify or bring new challenges to the safety demonstration. The next part aims at providing a kind of checklist of the question that were reviewed for each feature in WP2 tasks in order that all features are screened with a homogeneous approach.

4.1 Brief description of the potential challenges

For each of the above listed topics, general SMR specific issues were identified, which may challenge the respective item:

- Low power: does the limited power of an SMR and especially the reduced residual heat (~1/10 ratio compared to large PWR) change the way the safety demonstration can be performed?
    - Reduced power density in the core, both during operation and as decay heat
    - Reduction of source term for radiological releases
      → Potential for reduction of the emergency planning zone
    - Simplification of the safety cooling chain

- Fuel: are there fuel specificities that may affect the safety demonstration related to the feature under consideration?
    - Fuel enrichment
    - Fuel cycle length and high burnups
    - Maximum linear power
    - Small core dimensions and steep local power gradients
- Passive systems: are passive systems credited in the safety demonstration related to the feature under consideration?
    - Is there industrial experience of these systems? How is their reliability assessed?
    - How are they qualified?
    - Do they involve complex physical behaviour like two-phase flow?
    - Is it possible to test their function or inspect them on site (periodic tests)?
    - What kind of failures are expected to be considered in the safety demonstration (single failure, passive failure, functional failure, common cause failure, etc.)?
    - How to consider uncertainties in the safety demonstration and reliability assessment?
    - Could spurious actuation cause new kinds of Postulated Initiating Events (PIE)?
- Innovative equipment/material: does innovative equipment or material bring new challenges or suppress risks in the safety demonstration related to the feature under consideration?
    - Do we need to add or can we delete a specific initiating event (PIE)?
    - Do we need to addition or can we dismiss a specific release path?
    - What are the benefits brought by new materials (piping, core, etc.)?
    - Is there sufficient knowledge about material behaviour and failure mechanisms (fatigue, corrosion, breaks)?
- Multi-unit: is it planned to implement several modules in the same facility, and does it have any consequence in the safety demonstration related to the feature under consideration?
    - Do they share operational systems? Safety systems? Support systems? Human dependencies? Spatial dependencies?
    - Are there optimizations resulting from the expected sequential shutdown of the modules for maintenance and refuelling?
    - Are there internal or external events that may trigger PIE simultaneously in several modules?
    - Co-activity: some modules may be under construction while others are already in operation, does this create specific risks?
    - Are severe accident and subsequent releases considered simultaneously for several modules?
    - Are additional risk metrics needed to present the site risk profile resulting from simultaneous accidents at several units?
- Defence in depth: how is defence in depth implemented for the specific feature under consideration?
    - How are DBC, DEC-A and DEC-B (see section 2.3) determined and which fault sequences are considered?

- Is there a main line of defence (DBC) and diverse line of defence (DEC-A) or an argument for practical elimination?
- Is explicit independence between the lines of defence credited?

- Severe accidents:
  - Are severe accidents explicitly postulated?
  - How are core melt sequences defined?
  - Is the design compatible with the mitigation strategy (in-vessel retention, etc.)?
  - What is credited as the ultimate barrier? How are the applicable loads defined?
  - Is containment available in any plant mode?
- Practical elimination: are there specific issues regarding early or large accidental releases for the specific feature under consideration?
  - Are the severe accident conditions that are practically eliminated clearly identified?
  - Do the design choices bring specific risks regarding potential accidental releases?
  - Is the usual list of severe accident conditions that are demonstrated to be practically eliminated on large PWR still relevant? Is it possible to reduce it? Are the respective demonstration approaches applicable?
- Appropriate analysis tools: are there appropriate tools to perform the safety demonstration regarding the specific feature under consideration?
  - What are the plant specificities compared to usual large PWR? (integral reactor, passive system, extensive credit of natural circulation in complex geometry, etc.)?
  - Are the available calculation codes used in large PWR expected to be directly qualified for SMR safety demonstration?
- Appropriate methodologies: are the usual methodologies used in large PWR still applicable for the safety demonstration regarding the specific feature under consideration?
  - List of events to be considered (initiating events and derived accident sequences)
  - All plant normal operation modes defined and considered as possible initial state for accident sequences
  - Analysis rules for deterministic assessment, including single failure, passive failure, common cause failure, consideration of uncertainty and sensitivity, safe state, safety classification, requirements on systems credited in the demonstration
  - Analysis rules for probabilistic assessment, including multi-unit PSA, risk metrics and criteria, consideration of passive systems, human factors, common cause mutual influence of modules
  - Adequacy of decoupling criteria for safety analysis
  - Relevance of large NPP safety analysis acceptance criteria
- Support systems: is there any specific dependence on support systems for the safety demonstration regarding the specific feature under consideration?
  - Electrical system
  - Cooling systems
  - Ventilation systems
  - Other essential support systems

- Control room: how much the safety demonstration regarding the specific feature under consideration relies on operator actions?
    - Are there provisions to control and surveil the state of the plant during, operating and accidental conditions including severe accidents?
    - What kind of support systems are required to ensure habitability and operability of the control room and other control stations (electricity source, ventilation, etc.)?
    - Is the control room shared between several units?
- Spent fuel pool: are there specific challenges regarding the fuel pool for the safety demonstration regarding the specific feature under consideration?
    - Safety systems can provide essential functions irrespective of status of the reactor, include severe accident scenarios?
    - PIEs specific to the SFP addressed in the safety demonstration?
    - Deterministic safety assessment performed with the same rule as the reactor (DID, DBC/DEC, failures, etc.)?
    - Specific challenges created by fuel handling?
    - Are additional risk metrics/targets needed to evaluate acceptability of risk resulting from several SFPs, from simultaneous accidents at reactor and SFP?
    - Availability of containment?
    - Assumptions for practical elimination?
- Hazards: are some specific risks or benefits identified regarding internal and external hazards, compared to large PWR, for the safety demonstration regarding the specific feature under consideration?
    - Addition or deletion of specific hazards
    - Addition or deletion of specific hazard-induced common cause failure risks
    - Specific consideration regarding physical separation
    - Independence of the lines of defence regarding hazards

In the following subchapters, the related WP2 topics are analysed regarding their specific potential challenges given by the above listed SMR specific items.

4.2 Potential challenges regarding “Reactivity Control”

Fuel: Many SMR concepts are intended to be sited at remote locations. In order to minimise the maintenance work, the planned fuel cycles of up to 48 months (IRIS) or even 84 months (RITM-200) are larger than in current PWRs. To achieve these timelines, the power density of the core is lower than in large PWRs. For safety demonstration in some designs higher enriched fuels must be considered which are needed to get a higher excess reactivity (KLT-40S smaller than 20 %). The compensation of the resulting excess reactivity must be shown. Measures here are control and safety rods, burnable absorbers in the fuel itself as well as boron acid in the coolant. The latter is forgone by some designs to mainly as a cost reduction measure. However, due to a lower coolant density reactivity coefficient, omitting such a system might benefit the core safety design of the reactor. In any case, in cold shutdown mode, a sub-critical core must be demonstrated. Besides the control rods, also a diverse and redundant shutdown system must be provided by the design to achieve robust implementation of the **defence in depth concept**. If the additional burnable absorbers are integrated into the fuel matrix, the heat conductivity as well as the ductility of the fuel is changed, which impacts the fuel temperature as well its structural behaviour.

Passive systems: Regarding passive systems relying on small driving forces special attention has to be given towards their safety and reliability demonstration. This includes systems like passive shutdown of the core by control rods driven by gravity, compressed air or springs. The first one might be challenged, when the SMR is mounted on a barge and there is heavy sea. The swell is also important, when borated water should be injected into the core out of elevated tanks or core make up tanks, since such systems are also driven by gravity.

Furthermore, some reactors work under natural circulation even in normal operation. This can result in higher temperatures within the core impacting the Doppler coefficient of the used fuel and lead to a specific distribution and local maxima of power in the core.

Innovative equipment: Beside the use of burnable absorbers and consideration of boron free cores, the control rod drive mechanisms are often integrated in the RPV. Thus, the pressure difference between the core and the upper parts of the rods are considerably smaller than in current large PWRs and prevents the SMR against the rod ejection accident.

Practical elimination: This point is principally similar to large PWRs and must be explicitly demonstrated.

Appropriate analysis tools: The analysis tools for core calculations must be applicable and validated for specific core geometries (e.g. square or hexagonal lattice) as well as for different materials used in the core (e.g. burnable absorbers). Furthermore, the tools must provide capabilities to model inhomogeneous power distributions. Generally, that is the case already, however core designs with large discontinuities or steep local gradients in neutron flux and power pose some challenges.

Appropriate methodologies: Safety of the smaller cores with specific SMR related features must be demonstrated. Furthermore, decoupling criteria might differ from larger PWR and must be reviewed.

Supporting systems: For relevant supporting systems safety classification must be performed.

4.3 Potential challenges regarding “Heat Removal”

Low Power: The compact design in combination with the lower power density of SMR cores of the different SMRs might lead to a simplification of the safety cooling chains.

Passive systems: Passive systems used for heat removal are mainly based on natural circulation which is driven by low driving forces. Therefore, passive heat removal systems must be carefully examined and special attention must be given to safety and reliability demonstration. Some points for special attention might be:

- Non-condensable gases degrade condensation heat transfer of passive heat exchangers
- Impact of fouling in heat exchangers and piping on form losses and heat transfer
- Failure of natural circulation due to heavy sea (SMR on barge or ship)
- Safety demonstration of unintended activation of a passive system
- Single and common cause failures
- Consideration of uncertainties relating to system properties (e.g. specific form losses, heat resistances, etc.) and simulation models (e.g. two-phase pressure loss factor, film condensation rate but also nodalisation and other input effects)

When using a so-called integrated design, the maximum break size of a LOCA, which should be smaller than in current LWR must be determined so that inacceptable core uncovery and containment overpressure can be prevented.

Innovative equipment: Some concepts provide elevated large water pools as water inventory to inject water into the core during accidents or working as heat sink for residual heat exchangers. These high elevations can impact the seismic resilience of the reactor.

Multi-Unit: It must be demonstrated that accidents within one SMR module do not propagate to other modules and degrade safety functions there. E.g. in the NuScale design, all modules are located in one water pool. What happens, if the water pool is emptied by one module by its decay heat removal system, which transfers the heat into the pool? What is the safety impact to the other module, if an accident occurs? How is decay heat removed, if the pool is emptied?

Defence in depth: Regarding defence in depth, special attention might be needed to demonstrate sufficient independence of safety provisions assigned to different DiD levels for controlling an event or accident, particularly if the heat removal chain relies on passive systems.

Severe accidents: What safety provisions are there for heat removal from the containment in the long-term during postulated severe accidents not practically eliminated? Do any specific features of the distinct SMR concept lead to major differences in severe accidental phenomena?

Practical elimination: This is point is principally similar to large PWRs and must be explicitly demonstrated.

Appropriate analysis tools: Demonstration of appropriate heat removal can be done by different analysis tools. These tools need to be applicable and validated for the respective thermal-hydraulic phenomena related to the specific systems. Some points to look at are (list is not exhaustive):

- Specific heat exchanger geometries (e.g. vertical or horizontal tube heat exchanger, compact plate heat exchanger, etc.) including respective heat transfer regimes
- Heat transfer to large water pools by high containment walls (high Ra numbers)
- Consideration of 3D flows and thermal stratification inside the pools
- Flow instabilities relevant to a passively operated heat removal system
- Model uncertainties in simulation tools
- Lack of validation of analysis tools for the specific operating conditions of passive safety systems

Appropriate methodologies: Similar to the item “Passive systems”, the question is, how to deal with single failure criteria when using a passive safety system? In some regulations, the single failure of a passive system might be not assumed, if special requirements are fulfilled (see e.g. chapter 2.6 for the German regulations). Furthermore, passive systems can be subject of common cause failure even of different types if they rely on the same phenomena.

Support systems: Support systems (like the ultimate heat sink) should be sufficiently independent. Alternatively, remaining fault sequences have to be practically eliminated.

4.4 Potential challenges regarding “Containment”

Low power: The decay heat depends mainly on the core power and impacts therefore the pressure built up within the containment during a LOCA. Entering steam is condensed on the containment surfaces and possibly transferred to an outer pool or the ocean. For most SMR designs, in order to reduce costs containments are comparatively smaller in terms of volume to decay heat ratios than those in a larger-scale NPP. This can be at least partially compensated for by a larger effective heat flux to volume ratio. Containment overpressure control during accidental sequences is therefore an important issue especially for LW-SRM designs.

Fuel: The materials of the core of LW-SMR are basically the same than in current large PWR. Thus, no impact might be assumed on the kinds of fission product releases into the containment. However, depending on the metal mass inside the core, melting of the RPV could be faster due to higher thermal conductivity of the metallic melt in comparison with the ceramic melt.

Passive systems: In this part, the identified challenges of chapter 4.3 can be applied, since the systems rely on the same phenomena. Furthermore, demonstration of in-vessel retention of corium also with passive water injection into the RPV is an important feature of most SMR concepts. It is an area of active research for large NPP as well. Finally, to limit containment pressure during severe accidents, the correct installation and efficiency of hydrogen recombiners provided in most designs must be shown.

Innovative equipment: Also here large water pools and the impact on the seismic resilience of the reactor should be reviewed. Furthermore, the leak tightness of the pools could be challenged due to an earthquake. Another point is related to the very small containments, in relation to the large PWR: the pressure built up due to a leak into the containment is potentially larger (and faster) than in current reactors. The containment must withstand higher pressures and higher pressure peaks and leak tightness must be demonstrated. Leak tightness could also be challenged if no metallic but a concrete containment is used by the design. Finally, some designs provide a second containment. The impact of such a second containment on the safety case should be shown.

Multi-unit: Consideration of multi-unit aspects is of high importance regarding the containment. However, all points mentioned to that in chapter 4.3 can be applied respectively here, which is the same for **defence in depth**.

Severe accidents: in order to avoid early or large releases into the environment, containment integrity during severe accidents must be demonstrated. During severe accidents, releases of non-condensables, e.g. hydrogen, into the containment can pose increased challenges if the power to volume ratio of the SMR is smaller than for conventional NPP. Furthermore, smaller containments, especially in the sump or reactor pit area could lead to difficulties in demonstrating melt coolability after RPV failure, re-enforcing the need for in-vessel retention. Unmitigated corium attack on SMR containments, especially steel shell containments seen in several SMR designs, could lead to containment failure. Unacceptable releases into the environment have to be **practically eliminated**.

Appropriate analysis tools: Analogously to chapter 4.3, analysis tools must be validated for the specific phenomena inside or outside the containment. Generally, the challenges appear similar as to larger NPP.

Appropriate methodologies: It is needed to determine, whether additional failure scenarios might be considered for safety demonstration, e.g., due to innovative equipment or multi-unit approaches. Furthermore, refuelling might require opening the RPV as well as the containment vessel at the same time, so that two barriers are unavailable. Finally, for containment integrity safety case, rules are needed regarding single and common cause failures of passive safety systems and treatment of structures in the safety case.

Supporting systems: For relevant supporting systems safety classification must be performed.

Spent fuel pool: The spent fuel pool might be of importance for the reactor containment, if it is located inside the containment (only possible if the containment vessel is not too small). Otherwise, the confinement of potential releases from the spent fuel pools needs to be demonstrated and unacceptable releases need to be shown to be practically eliminated. Additionally, spent fuel might be stored inside a pool sharing water with the containment vessel (possibly more than one). If the water in the pool is evaporated by one module, what happens with the fuel?

Hazards: The list of potential hazards must be checked and modified to the special SMR design and location.

4.5 Potential challenges regarding “Spent Fuel Pool” and “Refuelling”

Low power: Due to the low power cores of the SMR, the power inside the pool might be lower than at larger PWR in the first glance. However, since most of the SMR concepts are multi-unit designs, more than one module is unloaded into a common pool so that the overall stored power and inventory might be of the same order of magnitude as at larger PWR.

Fuel: As mentioned before in chapter 4.2, many SMR concepts use high burnup cores to achieve high cycle lengths. Due to the high burnup the amount of fission products inside the spent fuel is potentially higher than in current reactor designs. A higher residual power density, specifically in freshly unloaded fuel, could reduce safety margins both against potentially exceeding the critical heat flux under certain degraded coolability conditions and against cladding failure in case of (partial) fuel uncovery. Furthermore, the residual reactivity of the fuel after unloading might be higher which impacts the acceptable distances between the fuel assemblies in the pool. The higher excess reactivity within the core leads also to fuel elements with higher reactivity. Consequently, an error in refuelling can be more severe.

Passive systems: As mentioned before, the driving forces of passive systems for decay heat removal are small. When using such systems for cooling of the spent fuel pool, the driving forces might be even smaller due to smaller temperature differences. The system effectiveness must be demonstrated. Furthermore, thermal stratification must be considered in the spent fuel pool, when using passive systems for cooling.

Innovative equipment: In the case of a boron free plant also the sub-criticality in the spent fuel pool must be demonstrated.

Regarding refuelling: If the containment is close to the RPV, both barriers, the primary circuit as well as the containment must be opened during refuelling (see also section 4.4).

Multi-unit: If every module would have its own spent fuel pool it must be shown, that the cooling systems for all pools are independent from each other (including instrumentation and control). Furthermore, a load drop into the shared pools or even on top of other modules while refuelling should be avoided or even excluded. The same shall be applied if the whole module is shifted inside the reactor building for reloading.

Defence in depth: It must be demonstrated that the cooling systems of the spent fuel pool(s) are independent and diverse, including support systems and intermediate cooling circuits.

Severe accidents: Postulated severe accident scenarios in the reactor should not lead to severe accident scenarios in the spent fuel pool, especially when located in the reactor containment. For postulated accidents in the spent fuel pool, a robust containment has to be provided in the design or relevant scenarios have to be shown to be practically eliminated.

Appropriate analysis tools: Regarding the spent fuel pool behaviour few high-quality validation data is available. Due to that, computer simulations might only give a rough qualitative analysis of the behaviour, but high-fidelity results might be hard to obtain. These challenges are similar to larger NPP designs.

Appropriate methodologies: The list of initiating events must be established on a systematic basis.

Supporting systems: Supporting systems used for the spent fuel pools might be linked if several pools are used. Such a linkage can impact the safety approach of the reactor.

Hazards: The list of possible hazards must be reviewed. E.g. due to underground implementation of the spent fuel pools in some design, some hazards could be dismissed.

4.6 Potential challenges regarding "Multi-Units and shared Systems"

Passive systems: As passive systems lack active control, passive systems servicing multiple units can be hard to design and operate. Even more importantly, it should be shown that interactions between multiple units do not interfere adversely with the operation of a passive safety system.

Overall figures of merit like CDF should be very low for SMR designs, when they are intended to be brought into highly populated areas. In the case that more than one module is on the plant site, the CDF per module should be even lower. Is the reliability of passive or innovative systems high enough to reach such low values? And is it possible to demonstrate such reliability figures given the uncertainties inherent in analysis tools, modelling approaches and assessment methods. In any case, the system reliability must be carefully checked.

Innovative equipment: As innovative equipment might suffer from unforeseen (common) failure modes, its presence could lead to simultaneous failures in several units, challenging safety systems and operator capacity. Is it necessary to protect against such possibilities, e.g. by putting modules into operation on a sequential schedule? At multi-unit sites, it can be imagined that specific items used for one specific module need to be repaired. Are there safety aspects to consider? Or should it be excluded?

Defence in depth: Since accidents in one module should not impact other modules it must be determined, whether common cause failures in different modules are possible and how they could be prevented.

Severe accidents: Severe accidents occurring at a multi-unit site might be more challenging than at single-unit sites. In the best case, the severe accident in one module does not impact the other ones. Accident mitigation could be done by shared systems, if system capabilities are sufficient and physical separation to achieve robust defence in depth can be maintained. How many (simultaneously) affected modules need to be assumed?

Practical elimination: Performing a PSA, the probabilistic targets are impacted by the fact that more than one unit is on the site. A multi-unit PSA is needed (MUPSA). One aspect is the question about probabilistic target figures of merit, e.g. is a core damage frequency of 10^{-7} per year assumed for one module or the whole plant? Also, systems shared (as well human, other dependencies and mutual influences) by all or multiple units must be considered. Finally, the practical elimination demonstration for a multi-unit site needs to be complemented by multi-unit deterministic arguments and robust design provisions.

Appropriate Methodologies: For multi-unit sites, specific PSA methodologies are needed. This is also the case for consideration of human factors. Existing approaches, as e.g. in IAEA SRS 96 /IAEA 19b/ need to be tested and further improved. Finally, also deterministic analysis approaches for multi-unit sites are needed. Is it sufficient to assume a PIE and one bounding single failure or common mode failure or is it necessary to apply at least a single failure to each module?

Support systems: If support systems are shared between the modules, what is the impact on the safety approach? Is it needed to force independence between all modules for all safety systems and their support system? What kind of interdependency is acceptable and what are relevant acceptance criteria?

Control room: If one control room is shared for several modules a number of questions arise: How many people are needed to correctly control the units? What kind of information do they need from the different modules? How are ambiguity errors prevented? How does an occurrence of an accident in one module impact the control and supervision of the others? In any case, co-activities as well as crisis situation aspects need to be considered for safety demonstration.

Spent fuel pool: If the spent fuel pool is shared by all of the modules, errors in handling of the fuel elements might can impact the other modules (see also chapter 4.5).

Hazards: The list of potential hazards must be reviewed. It has to be considered that external hazards can impact all or several other units in the plant. Internal hazards may propagate to several units, which needs to be prevented. Some designs are intended to be constructed module by module in such a way that some modules may already be in operation, while the others are still under construction. This state of the whole plant needs to be investigated.

4.7 Potential challenges regarding "Severe Accidents"

Low power: The low power of one individual module can lead to less radioactive material to deal with in source term considerations. However, since most of the SMR designs provide multiple units, the inventory can be potentially as large as a current PWR or even larger, depending on number of units, excess reactivity and planned burnup.

Passive system: As stated in the chapters above, passive systems challenge safety demonstration in such a way that reliability demonstration can be challenging. This is also the case if passive systems are intended to be used during severe accidents (e.g. passive in-vessel retention by condensation of steam on the containment walls and condensate drainage into the reactor pit or sump, which is used to cool the RPV wall from its outer side).

Innovative equipment: The containment vessels are often small, leading to higher pressure built up during LOCA compared with large LWR. Smaller distances between the RPV and the containment wall as well as less volume in the containment might lead to higher heat loads during severe accident scenarios, either from heat radiation or releases of hot gases. Leak tightness needs to be demonstrated to prevent early or large releases into the environment (see chapter 4.4). If the RPV is destroyed, some concepts provide cooling of the corium inside the containment (in-vessel corium containment).

Practical elimination: The list of severe accident conditions to be practically eliminated has to be adapted depending on the severe accident strategy adopted (e.g. vessel failure during in-vessel retention). Similar to other nuclear facilities, adequate emergency preparedness provisions need to be put into place.

Appropriate analysis tools: Severe accidents can be analysed by thermal-hydraulic system codes with respective models. Nevertheless, validation of the models might be a critical point.

Appropriate methodologies: Methodologies for mitigation or prevention of severe accidents need to be further developed.

Support systems: It is needed to justify what kind of support systems need to be safety graded, since they might be needed for severe accident mitigation. For ensuring robust defence in depth, support systems for design basis safety provisions and accident mitigation systems should be independent.

Control room: When one control room is shared by all modules the aspects shown in chapter 4.6 might be useful also in the case of a severe accident.

Spent fuel pool: If there is no cooling anymore of the spent fuel pool, the fuel can heat up and degrade. Consequences need to be determined and evaluated and unacceptable consequences need to be practically eliminated.

Hazards: As in the other chapters, the list of hazards need to be justified.

4.8 Potential challenges regarding "Emergency Planning Zone"

Low power: As stated before, the low power of individual modules leads potentially to a lower source term to consider in the emergency planning zone. This needs to be demonstrated. However, since more than one module is located at one site in most of the concepts, the overall source term must be carefully determined and might not be beneficial for the emergency planning zone. This is important for **multi-unit** considerations. Otherwise the overall requirements on the definition of emergency planning zones are not substantially different to other nuclear facilities.

Severe accidents: Depending on the planned site location, a reduction of the emergency planning zone, e.g. just on the plant site, might be needed (e.g. highly populated areas). For compliance with the Vienna declaration, long-term off-site emergency measures shall not be needed for scenarios not practically eliminated. In order to do so, more stringent criteria should be applied on radiological releases and a careful demonstration by **appropriate methodologies** is needed.

4.9 Potential challenges regarding "Operation and Human Factors"

Passive systems: Special attention has to be given on implementation and installation of the passive systems based on natural circulation, since even small deviations from the planned geometries (e.g. pipe inclinations, non-conforming welds, fouling, etc.) can impact the natural circulation or even jeopardise it. It might be needed to check the specific performance of the passive system after installation over a range of working conditions, since it might differ from the planned performance. Importantly, surveillance of the operational state of passive systems during all conditions is essential for situational awareness of operators. Information presented to operators and human machine interfaces and operator training need to be adequate, which might pose specific challenges.

A second point is maintenance of passive systems. Some system cannot be inspected and tested during plant operation, since testing would influence the plant operation itself. However, performance testing may require working conditions which are not available during a shutdown reactor (e.g. high thermal power might be needed to test the performance of a isolation condenser).

Finally, since the passive systems do not rely on operator actions (depending on the grade of passiveness), the problem of degraded human reliability must be considered.

Multi-unit: Most of the point for "Control room" of chapter 4.6 can be seen as also valid here, when multiple units are controlled by one **control room** and a reduced staff. The questions here are how to safely operate a number of modules in a single control room and how to minimise the risk of human error due to mixing of units. The staff must also be aware of initiating events influencing several units and how to deal with it. Also **support systems** can be shared by several modules which must be managed by the staff. These issues must be determined within **appropriate** safety demonstration **methods**.

Control room: When sharing one control room, it must be designed in such a way, that confusions of the staff leading to mixing of units are minimised. Furthermore, the control room must be capable to monitor accident evolutions in a clear way, so that the staff can react properly to the accident. What human factors engineering and man-machine interface design requirements need to be considered? Are there robust deterministic and probabilistic assessment methods and what acceptance criteria should be used? All measures needed to control the plant must be available.

Hazards: The staff must be aware that hazards can affect all units.

5. Summary

The review of the safety directives given by the European Union, IAEA, WENRA, ENSREG as well as by selected countries (EU and non-EU) showed that the extant regulation can be basically applied also for LW-SMR.

To achieve a safety design compliant to regulations and expectations, SMR designers implement safety systems and features, which were listed in chapter 3. A lot of information about the safety systems for decay heat removal, reactivity control as well as severe accident mitigation were published in publicly available references. Information about refuelling, fuel management and decommissioning or human factors were hard to find. This might be due to the fact that most of the screened concepts are not scheduled to be built in the next few years.

We find that many of the different concept are designed as so-called integral reactors in which steam generators, pressuriser and pumps and sometimes also the control rod drive mechanisms are installed inside the RPV. For decay heat removal often passive safety system come into play. Finally, cores with high burnups and high excess reactivity are used to enlarge the cycle lengths to make the SMRs economically more interesting.

These and other features lead to a number of concerns regarding potential challenges for safety demonstration. In chapter 4 we have presented some ideas on which potential challenges should be looked at in more depth. In summary, we find that the main features which can challenge the current methodologies are

- the use of passive safety systems affecting reliabilty assessment, consideration of single and common cause failures and requiring a consideration of uncertainties,
- multi-unit sites, e.g. when only a single control room with reduced staff is provided,
- high burn-up cores leading to new materials due to burnable absorbers with no boron systems and
- severe accident mitigation affecting practical elimination and emergency planning zones.

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