ELSMOR

Towards European Licensing of Small Modular Reactors

Containment Assessment

T2.6: Safety Methodology for Containment Assessment

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Summary

In this report, the potential challenges for safety assessment identified in WP1 regarding the containment are mirrored against IAEA and WENRA guidelines and German, British, Finnish, US and French regulations. A clear focus is put on operational, design basis and design extension conditions, whereas the discussion of severe accident conditions is relegated to the dedicated assessment report of the ELSMOR project. It was found that basically all regulations have consistent requirements, but with differences in details. For instance, specific numbers of redundancies and diversities of safety systems in different levels of defence or the issue of accessibility of containment SSC's can differ from country to country but must be taken into account for licensing procedures.

Finally, additional recommendations according to the identification of initiating events and fault sequences, adequate implementation of defence in depth and probabilistic and deterministic safety assessment with respect to containment assessment are given.

Main findings are:

- The use of passive safety systems should not jeopardize the robust implementation of defence in depth. Therefore, if a passive safety feature is foreseen for use in different levels of defence in depth, further justifications are needed either to practically eliminate additional fault sequences or to demonstrate the adequacy of the solution.
- Impacts of events and hazards on external water heat sinks like water pools and their further influence on containment integrity must be carefully examined.
- Prepare and validate evidence tools for the assessment of specific passive safety features, particularly innovative ones, and the phenomena relevant for the operation of these safety features well in advance.

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1. Introduction

Small modular light water reactors like the current NUWARD design provide special designs of its components and special safety features (e.g. metallic containment vessel with a comparatively high design pressure, which is submerged in a large water pool). Since such features are not realized in already built and running larger light water reactors, they might not be reflected by established methodologies for safety demonstration. In this report, potential challenges for containment safety assessment identified in WP1 of the ELSMOR project are mirrored against the current regulations and guidelines of IAEA, WENRA, Germany, Finland, France, United Kingdom and the United States in order to identify gaps and regulatory issues in the regulations and methodologies which could potentially hamper a licensing procedure. A clear focus is put on operational, design basis and design extension conditions, whereas the discussion of severe accident conditions is relegated to the dedicated assessment report of the ELSMOR project.

In section 2 we briefly summarize the potential challenges for containment safety assessment relevant to the investigation, including some severe accident related issues. In section 3, we explain the results and insights from our review of extant regulation regarding containment design and containment safety assessment. However, we do not look into all severe accident aspects and limit ourselves to those aspects, which are also relevant for safety assessment for the design envelope. In section 4, recommendations according to items in the ELSMOR proposal with respect to containment safety demonstration and assessment are derived. Both recommendations for the systematic consideration of potential new challenges as well as recommendations on the development and improvement of safety assessment methodologies are given. In section 5, we briefly summarize the main findings from the report.

2. Potential challenges for containment safety assessment

2.1 Issues regarding NUWARD

2.1.1 Passive heat removal through containment wall

The main fundamental safety function the containment system fulfils is the confinement of radioactive material inside the containment wall. Therefore, containment integrity as well as leak tightness must be ensured, which can be challenged by overpressure, underpressure and thermal loads, amongst others. Leak tightness is demonstrated by setting a maximum allowed leakage rate and showing that the actual leakage rates in all considered sequences are below this value. Overpressure and thermal loads for design basis accidents are generally enveloped either by loss of coolant accidents (LOCA) or by main steam line breaks due to steam inflow into the containment atmosphere. For some integral designs like NUWARD, the main steam line break (MSLB) could be more limiting due to elimination of large and medium break LOCA by design. In any case, the limiting scenarios both for the design basis and design extension conditions need to be identified. For design extension and severe accident conditions, build-up of non-condensables and heat loads from core degradation phenomena can lead to increased challenges.

In the NUWARD design, the containment vessel is submerged in a large water pool. Inflowing steam is condensed on the containment inner wall and heat is efficiently transferred into the pool. On the inside, condensation heat transfer is dominating. Within the wall, heat is transferred by heat conduction, while on the outside of the containment, heat is transferred by free convection and later on by subcooled or saturated nucleate boiling. For all SMR designs it must be demonstrated that for all kinds of LOCA and MSLB (assumed to be enveloping for other DBC) the heat transfer into the large water pool is sufficient to keep containment integrity. This includes the effect of non-condensable gases on the condensation heat transfer. We understand that in NUWARD due to the low secondary volume the LOCA case is assumed to be enveloping DBC including MSLB. The more compact containment design comes with the potential for more severe overpressure transients, however the exclusion of a large break from the design reduces short term loads. Additionally, if one overall pool is used as in the NuScale concept (see /IAEA 20/) it must be checked, how all other modules are affected, if the pool is heated up by one module due to an accident. In NUWARD, all modules are within a specific pool, which is not shared.

A simulation of the accident sequences of different LOCA and MSLB scenarios is needed to determine the actual pressure and temperature loads on the containment wall. Due to lack of validated models for the heat transfer on high containment walls, especially for natural convection on the outer containment wall side at which high Ra numbers are achieved, demonstrating that simulations results are valid can be challenging. A well validated code is therefore needed.

The pressure build-up will be influenced by the water pool located in the containment sump as a suppression pool for automatic depressurization of the RPV from the pressurizer /CHE 20/. While its effectiveness during automatic depressurization should not pose new challenges, the pool interaction with the remainder of the containment during LOCA scenarios could have a relevant effect on peak pressure loads, pressure and temperature evolution and containment flow patterns. More information on this potential design feature would be needed for more substantial conclusions.

The pressure and thermal loads can subsequently be used for structural mechanical analyses of the containment vessel to demonstrate its integrity during the considered accidents.

2.1.2 Impact of earthquakes

Containment structures can also be challenged by earthquakes. In the NUWARD case, the large water pools in which the containment vessels are submerged are located underground. It is a well-known effect in seismic analyses that soil properties influence the transmission of seismic waves. Importantly, ground motion amplification is known to take place in unconsolidated or soft soils and exacerbated when those soils have a high groundwater content.

The same effects affect the NUWARD design, where seismic waves will induce oscillations of the water pool outside the containment, which would be amplified near eigenfrequencies of the pool. These will be transferred to the containment vessel. At the same time, seismic waves will be transferred to the containment vessel via its connection to the ground plate, which may or may not be in phase with pool oscillations. This leads to a complex vibromechanic problem where the interaction of the vibration models and thus frequency response spectra of the containment vessel – and its internals – and the modes of the water pool and its sloshing superimpose. Only a detailed analysis could expose occurrence of peak loads to containment structures, its weld and seals, potentially challenging the leak-tightness of the vessel. Therefore, it must be demonstrated that leak-tightness of the containment is ensured during and after earthquakes.

In addition, on the pool side, it needs to be demonstrated that there is no consequential unacceptable loss of pool inventory for containment cooling due to seismic impact so that it is available as heat sink for DBC and DEC as needed. Considering the design paradigm for the main reactor buildings, this challenge should be able to be met with established assessment methods.

2.1.3 Refuelling with open containment vessel and periodic testing

During refuelling operation, the containment vessel needs to be opened for facilitating the unloading of spent fuel and re-loading of new fuel into the reactor core. While the reactor is safely shut down, this inevitably means that the containment is no longer available as a barrier. Moreover, because the containment is an integral part of the heat removal concept, particularly the heat removal concept for possible LOCA type faults is also compromised. Specific information on alternative safety features for shutdown is not yet available. Consequently, it needs to be demonstrated that the plant is safe nonetheless, and that fault scenarios leading to large or early accidental level releases are practically eliminated.

In addition, the refuelling operation itself and also periodic testing of the reactor pressure vessel (RPV) and connecting pipes requires personnel to be above the reactor or enter the containment while it and the RPV are open. For testing or maintenance purposes, co-workers need to go inside the containment vessel to inspect the RPV wall and pipes, service active components needed during plant operation (e.g. recirculation pumps or control rod drives). Against this backdrop, several safety related issues need to be questioned. Firstly, maintenance and testing can potentially trigger initiating events due to human error. Such faults could include LOCA type scenarios. Secondly, work place safety needs to be demonstrated. This does include protection and effective mitigation of fault scenarios leading to exposition to radiation fields, including human-error induced events. For that, workers in the containment need to be able to evacuate quickly. This does include workers performing inspections and testing of the RPV and other structures. Furthermore, if the fault was a break of the RPV boundary or of the containment barrier to the outer water pool, the containment vessel would be filled with water. In such scenarios, evacuation would be necessary to escape rising water levels. An open access tunnel from the side would be a valuable evacuation route, but might compromise the safety case, if it not possible to close it quickly, as that would allowing water to flow out of the containment into the rest of the building. Finally, it must be noted, that the previous points are connected with each other in such way, that the containment cannot be closed, if people are still inside the vessel. With use of remote surveillance and testing can alleviate this problem, it is expected that at least some work will have to be done by humans in the containment or at the RPV during an outage. In NUWARD design, there is a personnel airlock to access the containment. Moreover, there is also a small breach on top of the containment for maintenance purpose beside the main breach used for unloading/loading the fuel.

2.1.4 DEC-B issues

Regarding accidents with core melt (DEC-B), the molten core is planned to be kept in the RPV by cooling the RPV wall by flooding the space around the reactor vessel (so-called in-vessel retention). Furthermore, a depressurisation system, a diverse in-vessel water make-up system and a hydrogen mitigation system are provided /IAEA 20/. Since DEC-B conditions and sequences are out of the scope of this report but in the scope of T2.9 of ELSMOR, this will be assessed within D2.9. It is, however, important to remind that containment integrity demonstration for DEC including postulated severe accidents will be necessary.

2.2 Further SMR issues

In NUWARD, the containment and its penetrations are protected against overpressure and excess heat loads during DBC and DEC-A by the passive residual heat removal system formed by the submerged containment. We understand that the overpressure protection is supported by passive residual heat removal system RRP transferring heat by natural circulation to the water wall in DBC case and redundant RRP with specific diversified DEC-A systems (depressurisation and water-make-up systems) in complex DEC-A scenarios. Nevertheless, the only measure to depressurise the containment atmosphere is condensation on the containment wall and transferring the heat to the water wall. Other SMR designs use other systems as shown in the Table 2.1.

In some designs a condenser, which is located inside the containment as a passive heat removal system, is foreseen, so the main challenge is to demonstrate that the performance of the condenser is sufficient for keeping the containment pressure and temperature below certain acceptance criteria for DBC and DEC-A, respectively. Importantly, the safety demonstration of passive safety systems inside the containment does not qualitatively differ from the approach for those provided for the primary system of the reactor, which is treated in T2.5 and documented in D2.5 of the ELSMOR project. However, the influence of non-condensable gases, which is not a significant phenomenon in the reactor cooling circuit during DBC conditions, needs to be investigated thoroughly as the containment is either filled by air or inerted by nitrogen. Furthermore, free convection processes including temperature and density driven stratification are much more relevant than in the coolant circuit, even during natural circulation conditions, because in the RPV the flow is restricted and guided by the piping and RPV structures.

Table 2.1 Systems related to depressurisation and pressure control of the containment /BUS 20/

System	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[1], ACPR50S, Flexblue, IMR, mPower, NuScale, SMR-160, Westinghouse SMR, CAP200, NUWARD

Non-condensable gases affect the heat transfer on heat exchangers by reducing the steam temperature to the saturation temperature according to the partial pressure of steam in the air-steam mixture.

If the containment consists of several separate compartments (e.g. wetwell and drywell), transport processes of non-condensables can lead to accumulation in specific compartments and depletion in others, affecting local temperatures and pressures. This can pose specific challenges for the operation of passive heat removal systems as well as heat and pressure loads to the containment. If spray systems are activated, even underpressure challenges to containment integrity can occur. It might be needed to check, whether other modules could be affected by an underpressure failure of a module.

3. Mirroring of requirements on the identified safety issues

To find potential gaps in safety demonstration for containment assessment, it is needed first to summarise the different safety requirements given by IAEA and regulators. Information given in this paragraph are mainly taken from the following sources:

- IAEA guidelines
  - IAEA SSR-2/1 (rev. 1), Safety of Nuclear Power Plants: Design /IAEA 16/
  - IAEA SSG-53, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants /IAEA 19a/
- German Regulations
  - Safety Requirements for nuclear power plants (German Regulations) /BFS 15/
  - Different Safety Standards of the Nuclear Safety Standards Commission (KTA)

[1] Depends on reference

- UK Regulations
    - ONR Safety Assessment Principles for Nuclear Reactors /ONR 20/
    - Different ONR Technical Assessment Guides
- Finnish STUK Guides
- American Regulatory guides and specifically NUREG-800
- French ASN Guides
- WENRA positions

3.1 IAEA guides

3.1.1 General aspects

Fundamental requirements on safety of nuclear power plants and specifically on safety demonstration for containment assessment can be found in SSR 2/1 (rev. 1) from IAEA /IAEA 16/. It defines the fundamental safety functions under requirement 4, which are:

- Control of reactivity
- Removal of heat from the reactor and from the fuel store
- Confinement of radioactive material, shielding against radiation and control of planned radioactive releases as well as limitation of accidental radioactive releases

By means of a defence in depth concept, the containment serves as a barrier in order to fulfil the third fundamental safety function. It "[…] shall be provided to ensure, or to contribute to, the fulfilment of the following safety functions at the nuclear power plant: (i) confinement of radioactive substances in operational states and in accident conditions; (ii) protection of the reactor against natural external events and human induced events; and (iii) radiation shielding in operational states and in accident conditions." (/IAEA 16/ p. 43, Requirement 54).

3.1.2 Ensuring the containment integrity

As the tightness of the containment is also affected by its structural integrity, the design limits of the containment in terms of temperature, pressure and combustible gases need to be ensured as stated in Requirement 58. This implies the need for permanent and non-permanent provisions to control pressure, temperature and the amount of combustible gases within the containment during accidents. Additional safety features should be integrated to control the most likely "representative core melting conditions" /IAEA 19a/ p. 6.

In order to maintain containment integrity, temperature and pressure within the containment need to be kept below the design values for DBC. To achieve that during accident conditions in which mass and energy are released into the containment, different systems need to be provided. One of the safety features is the large free space of the containment in which steam can be stored as well as the structures within the containment on which steam can be condensed. Further systems mentioned in /IAEA 19a/ are spray systems, pressure suppression pool systems and containment heat removal systems (active and passive).

A permanent provision to control pressure and thermal loads is given in the NUWARD design by the heat removal through the containment wall into the water pool. The demonstration, whether this provision is sufficient must be achieved firstly by determination of the pressure built up and thermal loads during the respective accidents by engineering judgement /IAEA 16/, /IAEA 19b/ (based on assessment criteria that are scientifically or technically reproducible) or by simulation of the accident and the behaviour of the containment atmosphere. Here, the applicability of the computer code for simulation is the crucial point. The heat transfer models must be adequate to simulate all relevant phenomena (e.g. in this case the heat transfer on the outside of the containment wall at high Ra numbers, but also the steam and non-condensable distribution inside the containment and its impact on the condensation heat transfer on the containment inner wall). Good practice on code validation in line with /IAEA 19a/ can be taken from IAEA SSG-2 /IAEA 19b/. In a second step, the results of the analysis must be given to a strength analysis of the containment structure including welds and penetrations to assess the leak-tightness.

/IAEA 19a/ paragraph 2.9 states, that “multiple measures to remove heat from the containment shall be implemented. Specifically, systems provided for design extension conditions shall be as far as practicable independent from other safety systems”. Taking the strategy of NUWARD into account (see paragraph 2.2), in which the residual heat is transferred during LOCA from the containment through its wall into the surrounding pool, there is not yet enough information about diverse alternative means of heat removal from the containment for DEC conditions. This situation is related to on-going discussions if the provision of passive safety features justifies a reduction of diverse provisions between levels of defence-in-depth. This issue should be thoroughly investigated in the ELSMOR project as it is central to licensing of SMR designs.

IAEA SSG-53 /IAEA 19a/ does give specific guidance on loads and combinations of loads to be assumed for containment integrity demonstrations. Guidance is given for normal operation, DBC and DEC conditions, including combinations of loads from different kinds of events. If specific load characteristics are known, there are adequate structural mechanics methods available to demonstrate containment performance. The crux in safety demonstration will be the selection of bounding postulated initiating events in line with SSG-2 /IAEA 19b/. While the recommendations given in SSG-53 are already comprehensive, they should be checked for each specific containment design for completeness.

One particular challenge will be identifying limiting combinations of internal and external hazard loads with consequential internal faults for DBC and DEC. As explained above, hazard assessment, particularly seismic hazard assessment, for submerged containments will come with specific practical challenges.

Furthermore, /IAEA 19a/ provides specific guidance on safety features to control containment pressure for DBC and DEC loads, including spray systems and venting systems, and guidance on the confinement of radioactivity for operating conditions up to DEC.

Finally, SSG-53 also specifically requires adequate leak tightness for the containment with strict leak tightness criteria for the design envelope and possible limited increase for design extension conditions /IAEA 19b/:

3.1.3 Open containment during refuelling and periodic inspections

Notably, /IAEA 19a/ does point to maintenance, inspection and work place safety requirements for containment design. Visual inspections are seen as a measure to detect material ageing and cracks and to monitor both. Such inspections should therefore be possible if feasible in the design. As explained above, these requirements also do apply to integral PWR designs with compact and submerged containments like NUWARD. While analysis and demonstration methods are available in principle, performing such a demonstration for a specific SMR design and for all modes of operation (i.e. including refuelling) will still be a challenge.

3.2 German Regulations

The German regulations consist of the Safety Requirements for Nuclear Power Plants /BFS 15/ and several Safety Standards of the Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA).

3.2.1 Requirements of a containment system

All reactors shall have a containment system consisting of a containment hull, a surrounding building as well as auxiliary systems for retention and filtering of radioactive materials. All sources of radioactivity have to be placed an effective containment /BFS 15/, with obvious implications for spent fuel pools. The system should be designed in such a way, that the release of radioactive material is as low as possible and below the specified limits. Importantly, /BFS 15/ does require that the barrier function of the containments shall be restored in time to prevent accident level releases as part of the safety concept. This applies to all operating states up to design basis conditions. The containment shall be placed in a (reactor) building structure that protects against hazard impacts and enables ventilation conditions preventing releases.

The Safety Requirements further include quite specific criteria on containment integrity, including lists of relevant postulated initiating events, consideration of containment loads, and details of safety demonstration assumptions and safety margins.

3.2.2 Ensuring the containment integrity

Leak tightness of the containment shall be demonstrated using a maximum leak rate. This includes also the test of penetrations through the containment. Within the safety standards of the KTA 3401.1 – 3401.4 it is described, how a steel containment vessel shall be designed. Information about materials including testing (3401.1, /KTA 88/), design and analysis (3401.2 /KTA 16/), manufacturing (3401.3, /KTA 86/) and in-service inspections (3401.4, /KTA 17/) is provided. To design the containment, the expected load must be determined, which is described in KTA 3413 (Determination of Loads for the Design of a Full Pressure Containment against Plant-internal Incidents, /KTA 16a/). The containment needs to be designed against general overpressure, slight vacuum or external overpressure, pressure loads (non-uniformly acting load), temperature distributions, jet forces and reaction forces and missiles and whipping pipes. Pressure waves does not need to be considered. Leak sizes to be used for design are also described (e.g. for the reactor coolant pipe: double of the pipe cross section area (CSA), high energy pipes with diameter lower than DN50: circumferential rupture, etc.) as well as to be considered plant states. In the case of an integrated design (like the NUWARD) this is not needed, since large reactor coolant pipes are not provided anymore. For the latter, the most penalizing state is needed to be used except for the reactor coolant pipes in a PWR: here, the nominal operating conditions shall be used. The load treatment includes the use of conservative assumptions, e.g. an increased reactor coolant and secondary circuit volume (by 2 %) as well as a decreased containment volume (by 2 %), decreased inner and outer containment surfaces (by 2 %) and decreased surface of internals used as heat sinks (by 10 %). In Appendix A of KTA 3413 the heat transfer correlations to be used for determination of the thermal and pressure loads on the containment is given. They are valid for steam and air heat transfer to steel and concrete. The case of convective heat transfer on high containment walls into a large water pool is not considered.

KTA 2201.1 to 6 describe the “Design of Nuclear Power Plants against Seismic Events”. Components are classified regarding their specific importance for safety into different classes /KTA 11/ (I, IIa, IIb). First class refers to components needed to fulfil the fundamental safety functions and limiting radiation exposure. Class IIa components can possibly affect class I components during an earthquake and class IIb components refer to all other components. For seismic assessment it must be shown that class I equipment can still fulfil their designated tasks and that class IIa components does not negatively affect the class I components.

When modelling, building structures, soil and relevant components must be considered in the model. Although a large water pool surrounding the containment vessel is not explicitly mentioned, such a pool needs to be taken into account since it influences the analysis.

Also, in the German DiD concept, measures of one level of defence shall be independent from the measures of the level before, where DiD level 3 corresponds to DBC and level 4 corresponds to DEC. This is also true for residual heat removal during a LOCA from the containment atmosphere (2.1 (6) of /BFS 15/). The question is (when considering NUWARD): What system or measure is provided in the safety concept for a design extension condition sequence (e.g. a LOCA) for containment pressure control, if the sequence DBC was already claimed to provide heat removal through the containment wall? While suitable measures provided by the levels 1-3 can also be used in level 4 (2.1 (10) /BFS 15/), it is not admissible to use any accident management measures of 4b and 4c in the other levels as part of the safety demonstration (2.1 (11) /BFS 15/) and certainly not the same system for level 3 and level 4. Again, this boils down to the question if heat removal via the containment wall is treated as a safety provision of a safety system.

3.2.3 Open containment during refuelling and periodic inspections

Personnel conducting periodic inspections and maintenance inside the containment need to get into it by personnel airlocks. The locations of these airlocks have to be chosen in such a way that a quick evacuation of the containment is ensured and that the dose the co-workers, which might be exposed, is minimised. In a small SMR containment, which is intended to be opened only on the top, no airlocks might be provided by the design. Therefore, inspection is possible only if the containment is opened and might be difficult to re-establish. As mentioned before, in NUWARD there is a personnel airlock on the side to access the containment. Moreover, there is also a small breach on top of the containment for maintenance purpose beside the main breach used for unloading/loading the fuel.

In general, work in an opened containment is possible, when “effective and reliable retention functions are available and an inadmissible release of radioactive materials from the containment is prevented or stopped in due time.” (/BFS 15/ paragraph 3.6 (1)). Refuelling is not excluded. One measure could be to close the containment as fast as possible but before that an evacuation of the inspection personnel is needed. For this case, a suitable number of personnel locks are needed due to conventional fire safety regulations, which cannot be found in /BFS 15/. A solution could be to exclude the inspections for which a co-worker needs to be on the spot. If systems cannot be tested due to accessibility issues, ”[...] it shall be ensured that [...] provisions are taken against failure resulting from potential damage mechanisms, such as fatigue, corrosion and other ageing mechanisms, that a manufacturing documentation is available and that no irregularities or deviations from requirements to be fulfilled can be derived from it.” (/BFS 15/ paragraph 3.1 (12a)). However, this requirement is hard to fulfil if extended to the whole of the containment.

3.3 ONR Regulations

The ONR safety assessment principles /ONR 20/ include some high-level requirements relevant to containment design mainly in principles ECV.1 to ECV.10. As this ONR guide is applicable to all facilities and technologies, it is quite generic. Moreover, the principles repeatedly call for minimising releases and hazards, in line with the UK’s so-far-as-is-reasonably-practicable approach to risk reduction. Importantly, the confinement function shall be fulfilled primarily by passive systems and intrinsic safety features of the containment instead of active ones.

As is customary, containment load cases need to be defined for normal operation, design basis and design extension/severe accident conditions. Design basis loads needs to be derived from fault analysis. The design basis envelope needs to be determined from initiating events and their fault sequences, where the cut-off for infrequent faults (and thus DBA) according to FA.5 is about 10^{-5} pa. Moreover, it is relevant good practice in the UK for fault analysis – and thus design basis loads – only to credit safety systems claimed in the fault schedule for the respective fault. This approach also extends to internal and external hazards, where design basis threshold values are 10^{-5} pa and 10^{-4} pa, respectively. This includes, in principle, combinations of internal event and hazard fault sequences.

Similar to other regulation, ONR also requires safety demonstration for design extension conditions and also for postulated severe accidents in order to identify further reasonably practicable measures for risk reduction, identify and mitigate potential cliff-edge effects near the design envelope and to underpin accident management and emergency preparedness arrangements.

3.3.1 Ensuring the containment integrity

EMC.1 – EMC.3 deal with integrity of metal components and structures in general, which cover also the containment vessel structure. Here, guidelines and requirements are stated, on how the integrity of high reliability structures as well as other structures can be ensured. No details about the issues given by the NUWARD containment as given in paragraph 2.1 of this document are provided. The requirements are open and can be used also for the SMR related issues.

ONR has issued TAG-020 /ONR 20a/ with detailed and partly design-specific guidance for containment integrity safety cases from a civil engineering perspective. Some of that guidance is immediately applicable to SMR designs, e.g. regarding loads imposed by the presence of a liner on concrete structures and vice versa.

Also notable is the advice that safety cases should be based in a consistent set of codes and standards. This is relevant to SMR safety demonstration, insofar as this can impact design choices which can be justified under existing standards. If designers would need to go beyond existing standards, a new challenge arises: Either to get a standard setting body to adapt its standards so that they also cover innovative solutions, or to prepare an extended justification of their design – with associated regulatory risk.

3.3.2 Open containment during refuelling and periodic inspections

For refuelling, in the SAPs it is stated only that for all normal operation cases (including refuelling) the fundamental safety functions need to be fulfilled (540 /ONR 20/, ERC.1) and requirements need to be specified (542), which is again very open for the licensee.

Regarding ECV.3 in paragraph 525 (e) of the SAPs it is mentioned that (if appropriate) the size and number of service penetrations should be reduced to strengthen the confinement function. Also, the personnel access should be minimised (ECV.5). These requirements could be seen as beneficial, if the SMR design provides no service penetrations.

3.4 STUK Guides

3.4.1 Ensuring the containment integrity

Similar to the German regulations, STUK guide YVL B6 requires that a leak tight containment shall be provided by the reactor design. Keeping below the maximum allowed leakage rate ensures leak tightness /STUK 19/. Notably, concrete containments shall be made leak tight with a steel liner. Moreover, design loads derive from the most challenging design basis (and severe) accident with a 10% margin on pressure, while also considering a capability to withstand loads from complete oxidation of susceptible reactor materials (e.g. zircaloy, but also steel) and demonstrating leak tightness for predicted severe accidents pressures increased by 50%.

A secondary containment is specifically mentioned in the YVL B6, requiring ventilation systems to be filtered.

During class 1 and 2 accidents[2] overpressure protection of the containment by decay heat removal systems shall be provided by systems meeting the (n+2) failure criterion[3] and 72 h self-sufficiency criterion (448a of /STUK 19a/). Using passive components, these components only need to fulfil the (n+1) failure criterion. Furthermore, a diverse system shall be provided to cope with initiating events of any anticipated operational occurrences (AOO) or class 1 accidents so that acceptance limits for DEC A are fulfilled[4] (449 of /STUK 19a/). The diverse system shall fulfil the (n+1) criterion and the 72 h criterion and can be used to fulfil the (n+2) criterion for class 1 and class 2 accidents as stated before. DEC B and DEC C events are described similarly but without any restrictions on single failure criteria (450 and 451 of /STUK 19a/). Similar to these criteria, /STUK 19/ requirements 336 and 337 state that under postulated accidents, containment heat removal systems should be applied meeting the 72 h criterion and considering single failure and maintenance case. For severe accidents only single failure shall be taken into account (besides of 72 h requirement).

Since the large water pool surrounding the NUWARD containment acts as the ultimate heat sink for LOCA due to the heat transfer through the containment wall and for the passive safety systems transferring residual heat by natural circulation from the core. Both the pool and the containment are available only once and it is not possible to meet any (n+x) criteria by both of them (even due to the fact that it is a passively driven heat removal). For this case, the requirements could be changed in such a way that the redundancy criteria would only be needed to meet if it is feasible, since it is rather a structure than a system. Nevertheless, the safety function is subject to the redundancy and diversity criteria so that this issue must be discussed with the regulator.

Functional and physical independency shall be ensured for systems used to control severe accidents in level 4 of the DiD with respect to SSC's used in other levels of DiD (431 in /STUK 19a/). Up to now, it is not clear, how NUWARD would fulfil this requirement in the case of a severe accident with LOCA for heat removal from the containment atmosphere, since information on a diverse system besides the heat transfer through the containment wall into the large water pool is missing.

Design basis seismic loads to the containment are derived from the so-called design earthquake, which with a threshold frequency of 10^{-5} per year /STUK 19b/. Stronger earthquakes (until possibilities of 10^{-7} per year) are considered in the design extension conditions. SSC's are categorised into three categories (S1, S2A, S2B) /STUK 19c/.

S1 SSC's belong to safety related structures, systems and components needed to fulfil safety functions. S2A comprises SSC's not needed for the safety functions directly, but which could negatively affect S1 SSC's. All other SSC's can be categorised to S2B. S1 SSC's shall withstand a design basis earthquake in such a way, that they "[…] maintain their integrity, leaktightness,

[2] With respect to /STUK 19a/, class 1 and 2 accidents differ in their possibility to occur: while class 1 accidents occur less than once in 100 operating years but at least once in 1.000 operating years, class 2 accidents occur less than once in 1.000 operating years.
[3] (n+2): Taking single failure of one system and repair or maintenance of another system into account, (n+1): taking only single failure into account
[4] DEC A refers to common cause failure conditions, DEC B refers to combination of failures, DEC C refers to rare external events

functionality and proper position in a loading situation […]" (/STUK 19c/ p. 10). Due to their safety function, containment and water pool in the NUWARD case should be classified to S1. Since the mass of the water pool influence also the seismic behaviour of the containment vessel, this must be taken into account for determination of the design loads as well in safety assessment of the whole reactor, although the large water pool or similar is not mentioned in the reference.

3.4.2 Open containment during refuelling and periodic inspections

YVL B6 specifically requires that the containment components can be inspected and periodically tested. /STUK 19/ also deals with personal access locks. Two of them need to be in place as a minimum located in such a way, "[...] that at least one of them can be used as an emergency exit [...]" (/STUK 19/ requirement 320). They shall be implemented as air locks (requirement 321). If in specific SMR designs, however, no personnel would enter into the containment, it should be clarified how inspection and maintenance could be carried out on a level that is compatible with high integrity components and safety-classified systems.

For large equipment, an equipment hatch is needed to be provided by the design of the containment (requirement 322). If it is opened, it is needed to be ensured that it can be closed quickly. Furthermore, requirement 353 states that the containment, or alternatively the secondary containment, shall be leak tight during e.g. refuelling operation. Furthermore, containment integrity has to be restored before any unacceptable release could occur. Specifically, this last requirement can pose a challenge to the NUWARD design concept, as the refuelling operation from the top of the containment could preclude quick restoration of containment tightness.

3.5 NRC Regulatory Guides and Standard Review Plans (NUREG-800)

The General Design Criteria (GDC) all nuclear facilities in the US must fulfil can be found in Appendix A of Part 50 of /NRC 20/. Via 10 CFR Part 52 /NRC 20b/, which provides the framework for new reactor licensing in the U.S., these generic requirements are fully applied to LWR SMR, which are the subject of ELSMOR. The GDC give high-level similar to other regimes, including

- Leak tightness for the design envelope, considering dynamic loads, with additional margins to withstand accident conditions
- Capability to isolate containment penetrations
- Effective containment heat removal to an ultimate heat sink
- Control of containment atmosphere, including flammable gases and radionuclides, and monitoring of leakages and discharges
- Capability for inspection and testing

Further detailed recommendations can be found in the standard review plan (SRP) NUREG-0800, /NRC 20a/, which addresses the containment in multiple sections.

3.5.1 Ensuring the containment integrity

In /NRC 10/ it is described that the design of a steel containment can be accepted by the NRC reviewers if it can be demonstrated that its design pressure and temperature as well as the design leak rate is not exceeded during (design basis) accident conditions. The containment design shall include a sufficient margin to cover for uncertainties and withstand capability for severe accident conditions. Furthermore, the containment shall withstand loads generated by external hazards including the most severe earthquakes considered for the site as well as combinations of accidents and external hazards, internal hazards (e.g. missiles, pipe whipping, discharging fluids) and pressure loads following cladding water reactions, hydrogen burning or post-accident inerting. Quality standards need to be taken into account. Loads and load combinations to consider are also described. The load resulting from the water mass introduced to the containment by the water wall in the NUWARD case (but also in other SMR designs in which the containment is immersed inside a large water pool) can be grouped under “live loads”.

Margin for the design pressure of the containment shall be 10% above the peak pressure followed by any kind of LOCA (/NRC 07/). Heat removal systems used in the containment shall be capable to reduce the pressure inside the containment to less than 50% of the peak pressure after a LOCA within 24 h. For subatmospheric designs the pressure after a LOCA shall be decreased within 1 h to subatmospheric conditions again and maintained there for at least 30 d. The LOCA used for the determination of the highest peak pressures should be based on LOOP conditions as well as a most severe single failure in emergency power system, containment heat removal system or core cooling system /NRC 07/.

In addition, there is specific guidance related to the design of containments with a suppression pool (like e.g. BWR reactors) and ice condenser containments. These include somewhat different specific guidance. A common noteworthy theme is that the containment pressure envelope needs to consider the contribution from non-condensables gases generated in the severe accident in the core.

Notably, there is also guidance for the secondary containment, of which credit can be taken for achieving the confinement safety function – if present.

There is also further specific guidance on containment-related safety systems, e.g. for spray systems, containment heat removal, containment isolation and combustible gas control. This guidance is partially quite specific to certain designs and so may not well fit to an SMR design like the NUWARD. Given that the NRC might issue design-specific guidance in such a case, this does not appear to be a major concern. Currently, guidance like in /NRC 07a/, calls for a sump in the containment from which an ECCS can take suction is applied to NUWARD design as written, it will not be met due to the different safety concept. In the NUWARD design, we understand that recirculation of make up water is provided by low pressure recirculation pumps used after 3 days providing make-up of reactor coolant water inventory in case of LOCA accident, to maintain the heat removal function provided by the passive safety systems RRP. This might not be fully aligned with the guidance since there is no ECCS taking the suction from a sump and could therefore result in some residual regulatory risk regarding the position of the NRC.

3.5.2 Open containment during refuelling and periodic inspections

Regarding testing and inspections of the heat removal systems of the containment in /NRC 07a/ it is just stated that this shall be done where and when needed. Also, in the screened documents no requirement concerning number of personnel locks as in the STUK guides is given, although any locks in the containment shall be monitored.

3.6 French ASN Guides

In the scope of this report, the ASN guide No. 22 regarding the design of PWR /ASN 17/ as well as the “technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors” /ASN 00/ have been considered, where the latter was intended for the EPR design. ASN guide No 22 is available in French only, without an official translation into English available from ASN. A specific guide on containment design requirements is not available on the ASN website. As in other regulatory regimes, relevant good practice from operating plants, previous decisions and international developments will be taken into account. For example, /ASN 17/ requires for normal operation that “the best available technology in the sense of annex 1 to the ordinance of 26th April 2011 is taken into consideration” during the design process.

3.6.1 Ensuring the containment integrity

/ASN 17/ is basically consistent to other high-level guidance like IAEA SSR 2/1 /IAEA 16/ regarding requirements on the containment. In line with the EU safety directives, guide No 22 also provides a set of requirements for the third confinement barrier design and in general for the implementation of the confinement function.

Static provisions (e.g. a containment) must be provided as the 3rd barrier to confine radioactive material. If necessary, active provisions are added /ASN 17/. It shall be designed to withstand any DBC, DEC-A and DEC-B and to meet associated safety objectives. Containment isolation provisions shall be foreseen and demonstrated to be effective in all conditions. Requirements related to the containment integrity and leak tightness have to be established for the design basis domain and the extended design domain (these domains include the consideration of hazards).

Containment integrity requires containment heat removal. Guide 22 requires not only containment heat removal as a safety function for the design envelope up to DEC-A, but also an independent heat removal system, so far as practically feasible, for DEC-B conditions. Heat removal after a core melt accident shall be possible for a sufficient grace period, which has implications for the over containment design.

/ASN 17/ provides also requirements about:

- The consideration of hydrogen risk
- The risk of containment by-pass, in particular:
    - Direct leak of radioactive fluid outside the containment shall be avoided as well as conditions likely to lead to containment bypass.
    - The crossing of the containment by piping, cables or access opening shall be minimized and shall get into peripheral buildings with adequate confinement capabilities.
    - Potential leaks of circuits circulating radioactive fluids out of the containment shall be studied.
    - The adequate closure of the by-pass paths shall be implemented.
    - The adequate retention of the corium in the containment shall be implemented.
- The ventilation system provisions in support to the confinement function,
- The test of the containment leak tightness.

The seismic resilience is considered either by using site specific spectra and acceleration values or standardised spectra. Three different spectra considering different soils are given in /ASN 00/, which are seen as appropriate for Europe.

Specific to the control of postulated severe accidents with core melt, ASN Guide 22 includes further requirements, e.g. regarding the withstand capability against deflagrations of flammable gases and against corium attack after an RPV failure, if relevant.

3.6.2 Open containment during refuelling and periodic inspections

As for some other guidelines given in this report, during normal shutdown, open equipment hatch and fuel elements inside the RPV, provisions are needed to be capable to quickly close the hatch in the events of incidents and accidents which could lead to a release of radioactive material into the containment /ASN 17/. The designer has to show, during which state of the reactor can have an open containment This implicates that the chosen design option to refuel the RPV with open containment in the NUWARD case is possible in the frame of the French regulations, if the designer can show, that the containment can be closed quickly enough to avoid radioactive releases.

Periodic inspections of the containment provisions shall be possible /ASN 17/. Generally, the design must either allow for inspections and testing of the structures or the design must demonstrate their effectiveness over the lifetime by numerical analysis (in advance). Active systems contributing to the confinement function shall be testable. Personnel airlocks must provide independent interlocks /ASN 17/. How many personnel locks a design must provide is not stated, but it could depend on e.g. non-nuclear fire safety regulations, etc.

3.7 WENRA RHWG position on new reactors

In 2013, WENRA published an RHWG report on the safety of new NPP designs /RHWG 13/. The main requirements related to the containment are the following. The containment shall be the main barrier for the confinement of radioactivity in core melt accidents. For those, the containment shall ensure that no long-term emergency measures have to be implemented outside of the plant boundary and that no early releases into the environment occur. Safety functions for reducing containment pressure and fission product inventory shall be foreseen and there shall be secondary structures in place for the collection of leakages from containment penetrations.

Finally, /RHWG 13/ also requires for the plant – and therefore its containment – to withstand an airplane crash with the confinement of radiological materials effectively intact.

3.8 Summary

A review of extant regulation regarding containments for nuclear reactors show generally a good convergence. On a high level, requirements are consistent. Depending on the degree of detail and specificity of regulation, there are some notable requirements, which are particularly important for SMR designs. These include:

- Requirements to (re-) established the confinement function during fault sequences starting in operating conditions with an open (primary) containment,
- Specific requirements on number of redundancies as well independence of and diversity for systems assigned to different levels of defence in depth, even if passive safety features are claimed. This particularly relates to systems for design basis conditions vs. design extension or severe accident conditions.
- Containment structures, safety functions and their support systems need to be inspectable and testable.
- Worker exposure and workplace safety during design basis up to severe accident sequences needs to be considered in the safety case. This can include evacuation routes.
- A secondary containment (or confinement structure) can be required to capture leakages and from the primary containment (both for design leakages and as an additional barrier for design extension conditions)
- If a large water pool is used as the ultimate heat sink for accident conditions, additional requirements should be established e.g. if it is used as a common pool or robustness against earthquakes.
- Containment integrity and adequate leak tightness need to be demonstrated for DBC, DEC and postulated severe accidents as an element of the demonstration for the practical elimination of large or early (accidental level) releases.

4. Additional Recommendations

According to the proposal and needed work in WP2, recommendations for

- Identification of initiating events and sequences,
- Verification of the adequate implementation of the defence-in-depth,
- Probabilistic safety analysis
- Deterministic safety analysis (in terms of rules for analysis),
- as well as Identification of related R&D needs

are given in the following paragraphs. Importantly, aspects of containment integrity and leak tightness specific to severe accidents are not subject to recommendations. For the purpose of the ELSMOR project, these are subject to report D2.9.

4.1 Identification of initiating events and fault sequences

A list of postulated initiating events is given in /IAEA 19a/ or /BFS 15/ (annex 2). These lists include breaks of the primary system, the steam and feedwater system as well as equipment failures of systems dealing with radioactive material inside the containment and fuel handling accidents. Whether large break LOCAS are needed to be considered depends on the specific design of the primary circuit. For integral PWR designs, in which major components are placed within the reactor pressure vessel, a large break LOCA might be eliminated. Events during non-power operation are also relevant, including events where the containment is open.

Lists of internal and external hazards to be considered can also be found in /IAEA 19a/, but must be adapted for the specific case. Although units or module in a multi-module plant shall have their own safety systems and their own safety features for DEC (requirement 33 of SSR 2/1 (rev. 1) /IAEA 16/) in some SMR concepts systems like large water pools are shared for all modules. If such systems are shared, design basis accident sequences or design extension conditions have to be considered where either more than one unit requires the safety functions provided by the shared system (e.g. the pool) and also situations where fault conditions would progress through a shared system (e.g. a structural failure of the reactor hall during a severe earthquake affecting multiple units). In the first case, depending on the design envelope of the plant, the shared system has to be able to provide its safety function to multiple modules. In the latter case, potential consequential failures from design basis or design extension conditions fault sequences in another unit need to be systematically identified and assessed.

One important challenge to designs with compact and/or submerged containments will be demonstrating the confinement function during refuelling operation. Some regulatory regimes require restoration of containment integrity within timescales before accident conditions would occur. This might be a significant challenge, especially if consequential failures and fault sequence boundary conditions are taken into account. This includes steam generation and radionuclide release from an open reactor vessel during refuelling after loss of (active) heat removal and the subsequent deterioration of environmental conditions. But this also applies to potential leaks into the containment from outside pools or the RPV when access hatches, e.g. to the reactor pit, are opened for inspection, testing and maintenance.

Furthermore, if the SMR is located near to or used for co-generation with a e.g. chemical or hydrogen plant, external hazards affecting the containment integrity introduced by the other plant (hydrogen combustion, fire, explosions, etc.) should also be considered.

In summary, the process for identification of initiating events and hazards and the derivation of fault sequences does not differ substantially from existing procedures. With regard to containment issues, the following generic recommendations can be given:

- Systematically consider event and hazard impacts on external heat sinks (water pools) and the containment structures
- Investigate possible interactions between plant units via shared containment systems, including common water pools or a secondary containment and its ventilation system.
- For operating states consider sequences where the containment should be re-sealed, and look for consequential effects, including internal hazard phenomena, that might challenge (re-) establishing the confinement function.

4.2 Verification of the adequate implementation of defence in depth

All above mentioned guidance and regulations require a defence-in-depth concept, in which all levels are to the extent reasonable independent from each other (for one accident sequence). In /WEN 13/ it is stated, that the SSC classified to their respective level in defence in depth should be designed considering diversity, physical separation (structural or by distance) and functional isolation as far as practical.

Regarding NUWARD, the containment itself is used to remove residual heat from the containment towards the outer water pool in the case of a design basis LOCA. Depending on the regulations, using a passive system to fulfil the related safety function might decrease the number of redundant systems. Nevertheless, (n+1) redundancy may still be needed e.g. according to the Finnish regulations. Since also no secondary containment is claimed so far as a redundancy, it must be checked, whether a redundant system for residual heat removal in the LOCA case is proportionate for the NUWARD design.

For design extension conditions, respective safety provisions for controlling such an accident should be independent to the extent reasonably practicable to SSCs and measures on other DiD SSC's. Considering NUWARD, is might be difficult to demonstrate how DEC could be controlled, e.g. removing residual heat from the containment during a DEC LOCA sequence, without relying on the passive safety features like the external water pool, which are already claimed as the first line of defence for design basis conditions. A second line of defence, which is independent and diverse, might be needed to fulfil defence-in-depth requirements. In this regard, demonstrating adequate and robust implementation of defence in depth might necessitate strong justification and discussion of proportionality. Notably, the AP 1000 design was accepted by several regulators and does rely on similar safety concepts for certain faults.

This can be condensed into the following recommendations

- During design, make clear and justifiable claims on the assignment of SSCs to levels of defence in depth. Identify independent and possible diverse safety provisions also for protecting the confinement function and classify and categorize them accordingly. Usually, phenomena cannot be claimed as SSCs.
- In case a passive safety provision for the containment needs to be claimed on multiple levels of defence-in-depth for postulated fault sequences, prepare a robust justification that fault sequences are either practically eliminated or that alternative solutions would not be reasonably practicable. Early engagement with regulators on such issues would be advisable.
- A safety case with claims on a secondary containment (or confinement structure), that is available for DBC and DEC and for all relevant operating modes for capturing containment

leakages might be helpful for discussion with regulators and might be required in some countries.

4.3 Probabilistic safety analysis

Consideration of containment performance in the PSA is often an issue exclusively for PSA level 2 for current LWR designs, as containment failures during non-severe-accident sequences are often probabilistically irrelevant. Some exceptions can be interfacing LOCA sequences, where containment isolation fails. Similarly, if worker doses are determined, loss of containment integrity will need to be considered.

For SMR designs, and particularly for those with a submerged containment, no fundamental new challenges for PSA assessments are expected. It will still be necessary to determine the ultimate failure envelope of the containment and quantify failure probabilities for event sequences. Similarly, containment isolation failure probabilities and consideration of open containment operational states will still be necessary. As the containment is an integral part of the decay heat removal chain – similar to the AP 1000 design, its effectiveness will need to be assessed probabilistically. Here, the issue is how to justify the high reliability claimed for these passive safety features, also considering potential common cause failure mechanisms. As with other passive safety features with small driving forces, this will have to be based largely on deterministic simulations in combination with uncertainty and sensitivity analyses. The quality and validity of simulation tools and their validation status will therefore be essential for valid PSA results. For common cause failure mechanisms and their relevance, a systematic analysis of phenomena affecting the performance of the system, ageing and also human factors impact and their uncertainties is needed to substantiate PSA claims.

Many of the current SMR designs are dealing with multiple modules inside the reactor building. As already stated in 4.1, for identification of initiating events, also events introduced by the other modules must be taken into account. Therefore, a single unit probabilistic safety analysis might not be suitable anymore, when considering a multi module SMR plant. Therefore, a so-called multi-unit probabilistic safety assessment should (MUPSA) /IAEA 19c/ be performed for safety demonstration taking into account the impact of all modules on each other. This should include shared systems (e.g. final heat sink in NuScale case), internal events in one module affecting other modules, external events affecting all modules. Furthermore, if the SMR is coupled to a neighbouring chemical or hydrogen plant, the impact on the containment integrity due to hazards introduced by the other plant should be taken into account for PSA.

The major recommendation on this section is to prepare a robust justification of reliability (and thus failure) probabilities and failure modes for passive containment safety features. This pertains to the validity of simulation codes, but also to the identification and assessment of common cause failure mechanisms.

4.4 Deterministic safety analysis and potential development needs for thermal-hydraulic system codes

The design envelope for the reactor results in decoupling criteria, which the design of the containment needs to comply with. In terms of the containment, these decoupling criteria are derived from the maximum stresses, the containment structure is exposed to during design basis conditions including accidents and external hazards like earthquakes, so that the stresses challenging the containment integrity during accidents and external hazards are mainly the result of pressure and temperature loads, as well as oscillations, on the structures. Determination of loads as well as stresses is based on engineering judgement usually supported by computer codes. Thermal-hydraulic system codes are used in this context to determine enveloping pressure and temperature loads presuming an adequate applicability to simulate the related phenomena, which means in the scope of this report, within the containment. This includes reliable models implemented in the code, a broad validation basis and experienced code users. Moreover, uncertainties in model parameters and phenomena have to be taken into account. Some recommendations regarding potential further development needs of thermal-hydraulic codes with respect to containment safety systems and phenomena can be found in the following section.

4.4.1 Heat removal and pressure suppression

Heat removal and pressure suppression of the containment in the design case is performed in NUWARD by the use of the containment wall as a heat exchanger by condensing steam on the inner wall and transferring the heat to the outer water pool. The thermal-hydraulic system code used for simulation needs to be qualified for the respective phenomena:

- Natural circulation in the containment and of the water inventory
- Condensation on containment walls
- Effect of non-condensable gases on the condensation heat transfer
- Convective heat transfer at high Rayleigh numbers on the outer wall side
- Subcooled and saturated nucleate boiling on the outer wall side

Other designs provide passive heat exchangers of various geometries on higher containment positions to condensate steam and reducing the pressure inside the containment. The list of phenomena a qualified code needs to simulate depend highly on the specific design but generally the following phenomena should be dealt with:

- Natural circulation in the containment and the inner side of the heat exchangers
- Condensation heat transfer on the heat exchanger outer walls (including the effect of non-condensable gases)
- Convective and boiling heat transfer at the inner side of the heat exchanger
- Specific phenomena depending on the geometry of the heat exchanger (e.g. flashing, geysering, etc.)

If the containment is segmented into a drywell and a water filled wetwell the code needs to simulate the overflow of a steam/non-condensable mixture from the dry- to the wetwell. Bulk condensation in the wetwell takes place, while the non-condenables are concentrated at the wetwell upper parts. Steam either needs to diffuse into this layer, which significantly increases the heat resistance of the process, or needs to be transported to(wards) the heat exchanger surface by means of (turbulent) mass transport. It is therefore necessary to understand and adequately predict local flow conditions in the vicinity of the heat exchanger. Especially, if the containment compartments are large, the code should be applicable also to resolve 3D flows inside the dry- and wetwell including thermal stratification.

Finally, systems to spray water directly into the containment atmosphere are also provided by some SMR designs. To simulate this, the tools must be capable to model droplet induced condensation as well the droplet movement downward the containment.

Primary depressurisation is foreseen in some designs by relief of steam from the pressuriser either directly into the containment or into a large water pool, which is located inside the containment. While the first possibility should not differ from the modelling of a leak of the primary system into the containment, the second possibility might lead to challenges equal the wetwell modelling mentioned above (3D flows, etc.).

4.4.2 Emergency core cooling

In case of a LOCA, some SMR designs provide a recirculation system for long term core cooling. The steam is condensed at the containment inner wall or structures and drained into the sump or reactor pit. From here, it can be fed back into the reactor pressure vessel either by pump (active) or by the hydrostatic head of the water inventory of the sump or pit (passive). System codes must be able to simulate both possibilities. This might be especially challenging, if codes need to be coupled with each other (like ATHLET and COCOSYS in which ATHLET simulates the reactor coolant system and COCOSYS the containment) and the flow through the interfaces (e.g. recirculation valves within the reactor pressure vessel wall) is affected by low driving heads or even flow reversal.

4.4.3 Uncertainties

The assessment of passive safety features for the containment and the determination of failure envelopes will necessitate an analysis of uncertainties. This pertains firstly to uncertainties of simulation models and other evidence tools. As engineering judgment treatment of uncertainties (conservative approach) might be difficult, all the necessary information for robust uncertainty analyses should be available for the respective simulation tools. As this needs to be based on uncertainty determination for (complex) phenomena and the related constitutive models, the necessary data sets, including experimental results, need to be available. Uncertainty ranges of (constitutive) model parameters should be determined specific for simulations tools, preferably by the authors of these tools.

4.4.4 Recommendations

With regard to deterministic safety analysis and simulation codes for the containment function, an important recommendation is to prepare and validate those evidence tools needed to support a safety case well in advance. For that and in order to independently substantiate the safety case, it is recommended to utilize experimental facilities. It should be considered to operated full-scale test facilities for passive safety features which do not scale well regarding dimensions and/or geometric details. This is particularly important for innovative concepts, for which no previous experience and specific validation is available.

Considering that a safety case for containment safety features will need to rely on uncertainty and sensitivity analyses, it should be ensured that adequate uncertainty quantification input data for codes and other evidence tools are available.

Finally, it is recommended to have evidence tools available that can realistically simulate the behaviour also during design extension and severe accident conditions.

5. Summary

Potential challenges for safety assessment for light water SMR related to the containment have been mirrored against regulations and guidelines of IAEA, RHWG and several countries. All considered references refer to a containment as a third confinement barrier against the release of radioactive material. The containment integrity has to be ensured with regard external events and so contributes to the protection of the confinement function. To fulfil these purposes, it must be demonstrated that the containment integrity and leak tightness is ensured in all design basis and extension conditions. In this aspect, all regulations are consistent. Going into low level requirements they differ in some details as given in chapter 3. Some of the requirements are important for SMR:

- Requirements to (re-) established the confinement function during fault sequences starting in operating conditions with an open (primary) containment,

- Requirements for confinement capabilities of the reactor building and peripheral buildings with regard to potential leaks from the containment and radioactive fluid circuits located outside the containment,
- Specific requirements on the number of redundancies as well independence of and diversity for systems/provisions assigned to different levels of defence in depth, even if passive safety features are claimed. This particularly relates to systems for design basis conditions vs. design extension or severe accident conditions.
- A safety case with claims on a secondary containment (or confinement structure), that is available for DBC and DEC and for all relevant operating modes for capturing containment leakages might be helpful for discussion with regulators and might be required in some countries.
- Containment structures, safety functions and their support systems need to be inspectable and testable.
- Worker exposure and workplace safety during design basis up to severe accident sequences needs to be considered in the safety case. This can include evacuation routes.
- Containment integrity and adequate leak tightness need to be demonstrated for DBC, DEC and postulated severe accidents as an element of the demonstration for the practical elimination of large or early (accidental level) releases.

Finally, also recommendations for specific aspects of a safety demonstration of the containment have been collected. The main findings were the following:

- The use of passive safety systems should not jeopardize the robust implementation of defence in depth. Therefore, if a passive safety feature is foreseen for use in different levels of defence in depth, further justifications should be prepared to show the robustness and proportionality of the solution.
- Prepare a robust justification for the reliability and failure modes of passive safety features.
- Impacts of events and hazards on external water heat sinks like water pools and their further influence on containment integrity must be carefully examined.
- Prepare and validate evidence tools for the assessment of specific passive safety features, particularly innovative ones, and the phenomena relevant for the operation of these safety features well in advance.
- Prepare adequate uncertainty characterisation data for passive safety features, possibly derived from dedicated experiments.
- Consider the need to couple evidence tools for best-estimate calculations to substantiate a safety case.

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