Working Group on Design and Safety Analysis

Phase 3 Report

Containment Systems

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EXECUTIVE SUMMARY

There continues to be sustained global interest in small modular reactors (SMRs), which have the potential to play an important role in globally sustainable energy development as part of an optimal energy mix. In particular, SMRs may enhance energy availability and security of supply in countries expanding their nuclear energy programs and those embarking on a nuclear energy program for the first time.

As the interest in SMRs continues to grow, so does the importance of international collaboration. Given that its main purpose is to bring together experienced regulators to identify and address key SMR-related challenges, the SMR Regulators' Forum has an increasingly important role to play in making such collaboration possible.

The SMR Regulators' Forum was formed in 2014 as a regulator-to-regulator entity to consider key issues that could emerge in future SMR regulatory discussions and propose common positions regarding the way in which these could be addressed. The Forum's work is expected to help enhance safety as well as efficiency in SMR regulation, including licensing, and to enable regulators to inform changes, if necessary, to their requirements and regulatory practices. Since then, the Forum has had three phases of work. For more details about the Forum, please visit: https://www.iaea.org/topics/small-modular-reactors/smr-regulators-forum.

This report has been produced by the Design and Safety Analysis (DSA) Working Group (WG) of the SMR Regulators' Forum during its Phase 3 (2021 to 2023). It examines the concept of containment as currently defined by the IAEA Safety Glossary [1] and as used within the various IAEA Safety Standards and then seeks to understand how this might change with the introduction of SMR, especially those that break away from standard water-reactor technology. Underpinning the discussions within the WG were the responses to the questionnaire in Appendix that had been previously circulated. The text presents "common positions", i.e. agreements reached within the Working Group, on issues such as barriers for defence in depth, protection against hazards etc.

The questionnaire and the responses to it are presented in appendices.

Common Positions for this report

Common Position 1

> The design should include multiple independent and diverse means (Defence-in-Depth) to ensure that the function of containment is met for all operating states in accordance with SSR-2/1 [2].

Common Position 2

> Due to an SMR's compact size, the independence of the barriers could be more challenging to achieve for SMRs than for large reactors. In accordance with the Defence-in-Depth (DiD) approach, the design should ensure that measures are included at each level. The measures included at any particular level should remain independent as far as practicable of those at all other levels.

Common Position 3

Designs may be based on a graded approach in assessments of the novel containment systems to achieve safety, security and safeguards objectives. Nonetheless, a safety case must be presented to the regulatory authority to demonstrate that the proposed containment system design can and will comply with the overarching licensing requirements.

Common Position 4

Regardless of how the containment systems are designed, provisions are required to prevent accidents associated with internal and external (natural and manmade, accidental or intentional) hazards.

Common Position 5

Where containment systems are shared among the units/modules, the design should take account of the potential hazards such arrangement may introduce.

Common Position 6

Depending on the siting considerations (for example, for the underground/submerged containments or for floating SMR installations), the design of the containment systems needs to consider such potential specific hazards this arrangement may introduce.

Common Position 7

The choice of the design extension conditions (DECs) should be explained and justified. For this purpose, probabilistic assessment should be used in a complementary manner and not be used as a sole justification to screen out low frequency events.

Common Position 8

Regardless of how the containment function is met, all designs should demonstrate that there will be no early or large release under any accident conditions. Requirement 56 of SSR-2/1 [2] specifically applies including Paragraphs 6.22, 6.23, and 6.24, and Requirement 58, Paragraphs 6.27, 6.28 B and 6.30 specifically apply. For more information, see TECDOC-1936 [3].

Common Position 9

Functional containment should be designed to minimize ingress of substances that may have a negative impact on structures, systems and components (SSCs) important to safety. For example, for High Temperature Gas Cooled SMRs, functional containment should be designed to minimize air and water ingress, which can lead to oxidation of the core and possibly other SSCs in case of a depressurization accident. Requirement 57, Paragraph 6.26 of SSR-2/1 [2] specifically applies. For more information, see TECDOC-1936 [3].

Common Position 10

The design should provide for the control of fission products, hydrogen, oxygen, and other substances from the containment system atmosphere as stated in SSR-2/1 [2], Paragraph 6.29.

Common Position 11

For designs that may experience high energy releases in accidental conditions, the capability to remove heat from the containment should be ensured, in line with SSR-2/1 [2], specifically Paragraph 6.28.

Common Position 12

For maintenance, testing, inspection and repair (MTIR) on or off site, the design should provide for suitable access arrangements for the 3S (safety, security and safeguards) SSCs. Requirement 57, Paragraph 6.25 of SSR-2/1 [2] applies. For more information, see TECDOC-1936 [3].

Common Position 13

Design should accommodate the IAEA safeguards activities and provide physical access when required.

Common Position 14

Considering novel SMR designs that may produce different and/or more rapid containment system degradation, the design should provide for the aging management for the required lifetime of an SSC. This also includes monitoring of the degradation of the SSC associated with containment function. Aspects related to siting (for example, submerged containments and underground construction) may require novel inspection techniques.

Common Position 15

The leakage rates assumed in the design of the containment system should be justified and demonstrated to the regulator. The design should provide for the verification that the designed leakage rates are not exceeded for the required lifetime. If for a design there is no claim on the need for a leakage rate on the containment structure, detailed justification and demonstration of the adequacy of such claim must be provided to the regulator.

Common Position 16

The design should provide for the control and cleanup of fission products, hydrogen, oxygen and other substances that may have a negative impact on SSCs important to safety, from the containment system atmosphere in line with SSR-2/1 [2], Paragraph 6.29.

Common Position 17

Regulators should strive and continue to develop or review regulatory requirements and guidance pertaining to SMR technologies where appropriate. This is especially true in case of non-Light Water SMRs, where containment system designs are substantially different from typical large light water-cooled reactors (LWR) and may change the emphasis of which particular SSCs are important.

Common Position 18

The IAEA should continue to assess the extent to which the current safety standards address the safety of SMRs and develop guidance to address the identified gaps. We commend the ongoing work by the IAEA in this area and recommend that the SMR Regulators´ Forum considers it when the results are in hand.

1. INTRODUCTION

In traditional LWR designs, the function of containment is achieved by the SSCs that protect the reactor and other systems from the external and internal hazards, accidental or intentional. The principal barrier to radionuclide release credited during accidents is the containment building/structure. The limiting licensing basis event for the LWR is the large loss-of-coolant accident resulting from a breach of the reactor coolant system. This postulated accident sequence is a rapid transient characterized by high energy release of high temperature, pressurized-water reactor coolant into the containment structure. Since the initiating event is a breach in the reactor coolant circuit, it is assumed that the fuel cladding and reactor coolant pressure boundary are compromised. Thus, the containment building/structure is required to withstand the released mass and energy from the coolant system and to contain radionuclides released from the fuel, all of which are reliant on the integrity and reliability of its design basis functions of pressure-retention and low-leak rates. In this way the plant is designed so that a major accident would not cause the regulatory limit on radiation dose at the site boundary to be exceeded.

Some advanced non-LWRs claim that a leak-tight and pressure retaining containment structure is not relied upon to restrict the consequences of accidents - operational, external or human induced events. This claim is based on inherent and passive safety features which are intended to reduce the reliance on the structure and its associated systems to provide the containment function. Such designs may propose different provisions that limit radionuclide releases to the environment, for example retention of radionuclides at their source in the fuel rather than allowing significant fuel failures and subsequent reliance upon other barriers (the reactor coolant system pressure boundary and containment structure) to ensure that dose at the site boundary meets regulatory limits as a consequence of postulated accidents. For these types of design, without a pressure retaining containment structure, the USNRC uses the term “functional containment” for High Temperature Gas Cooled Reactor (HTGR) technologies as defined in Section 2.1 “Terminology”. It is recognized that internationally there is no consistent definition for “functional containment”, but the DSA WG decided to adopt the interpretation outlined in USNRC Regulatory Guide 1.232 [4]. This report aims to provide common positions in line with this definition.

Regardless the approach to achieve containment function, all reactor designs need to ensure that the containment Safety, Security and Safeguards (known as the 3S) functions are adequately implemented.

2. SMR CONTAINMENTS

The principal technical requirements for the design of the containment and its associated systems are provided in SSR-2/1 [2]. According to Requirement 54 of SSR-2/1:

"A containment system shall be provided to ensure, or to contribute to, the fulfilment of the following safety functions at the nuclear power plant:

- *(i)* *confinement of radioactive substances in operational states and in accident conditions;*
- *(ii)* *protection of the reactor against natural external events and human induced events; and*
- (iii) *radiation shielding in operational states and in accident conditions"*

A design will include a containment system which may be formed as a single containment structure or as multiple barriers. Regardless of the approach taken, all reactor designs need to ensure that the above-mentioned containment functions are adequately implemented.

A typical LWR containment structure sits on a thick, steel-reinforced concrete foundation with steel-reinforced concrete walls and an interior liner made from steel plate. Several SMR designs are based on the claim that severe fuel damage and high energy events related to failure in the reactor circuit may not occur as part of design; therefore, a robust containment structure such as in large LWRs may not be relied upon. These SMRs have approached plant design and the means of maintaining DiD somewhat differently from large LWR designs. In general, the focus has shifted from mitigation features to prevention features. For example, HTGR designs aim to achieve high reliability of safety functions by using simple and passive decay-heat-removal and reactor-shutdown methods, as well as tristructural isotropic (TRISO) fuel which provides a barrier to radionuclide release. These passive features in HTGRs are directed toward maintaining fuel integrity, even during very unlikely events. Mitigation is provided in the HTGR containment systems design through different provisions, specifically, physical phenomena (fission product retention, plateout, and holdup), and through use of the long-time response of the reactor in accident sequences. This has resulted in designs that propose to accomplish prevention, mitigation, and emergency planning in ways different from those used for LWRs. The claim for such SMRs is that a leak-tight and pressure retaining functional containment is not needed to restrict the consequences of accidents.

The multiple barriers comprising the functional containment for non-LWRs may be internal and/or external to the reactor and its cooling system and are provided to control the release of radioactivity to the environment and to ensure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. In addition, the scale of SMRs enables different type of design provisions in comparison to large LWRs, e.g., underground siting and containments submerged in water pools (security-by-design[1]). SMRs' physical size and siting may necessitate different type of assessment analyses and provisions for the external and internal hazards.

[1] Security-by-design (SeBD) is an approach to the design of a nuclear facility in which nuclear security principles and provisions are integrated into the design process as early as possible.

The approach to containment in some new designs is captured in the IAEA's safety report "Applicability of Safety Standards to Non-Water-Cooled Reactors and Small Modular Reactors" [5]:

"In WCRs (Water Cooled Reactors), the containment system comprises a containment structure and its support systems. For non-water-cooled EIDs (Evolutionary Innovative Designs), however, the functions of the containment system may be achieved quite differently. An engineering combination of design provisions (systems) and civil structures may be proposed to meet containment safety objectives. In HTGRs (High Temperature Gas-Cooled Reactors), for example, the confinement of radioactive substances under operational and accident conditions is mainly achieved by the TRISO fuel, which provides a robust barrier, and a containment structure, such as the one used by WCRs, may not be required to provide this function. Nevertheless, it is still needed for other functions, notably the protection against natural and human induced external events. The requirements and recommendations related to the containment system in the IAEA safety standards do not consider this major novelty, nor do they address situations potentially arising from the EIDs. Nevertheless, an entity, such as the NPP operator, need to still demonstrate that the proposed safety provisions practically eliminate large releases of radioactive material to the NPP's surrounding environment and that other releases are kept below acceptable limits and as low as reasonably achievable. This concept of a confinement function using barriers other than the traditional containment system of WCRs is sometimes referred to as functional containment. As is well known, WCRs have other barriers to prevent the release of radioactive material to the environment, in addition to this containment system."

2.1.TERMINOLOGY

The IAEA Safety and Security Glossary 2022 [1] proposes the following definitions:

confinement

Prevention or control of releases of radioactive material to the environment in operation or in accidents. Confinement is closely related in meaning to containment, but confinement is typically used to refer to the safety function of preventing the 'escape' of radioactive material, whereas containment refers to the means for achieving that function.

containment system

A structurally closed physical barrier (especially in a nuclear installation) designed to prevent or control the release and the dispersion of radioactive substances, and its associated systems.[2]

containment

Methods or physical structures designed to prevent or control the release and the dispersion of radioactive substances.[2]

barrier

A physical obstruction that prevents or inhibits the movement of people, radionuclides, or some other phenomenon (e.g., fire), or provides shielding against radiation.

[2] This is a partial definition relevant to this report. For the full definition, please see [1].

The issue of inconsistent terminology

In SSR-2/1 [2], the term ‘containment’ is used interchangeably to describe the confinement function and to refer to the structure which provides the functions captured in Requirement 54.

Each of the WG countries has its own definitions of the terms, which are provided in the annexes A to G, their responses to the Questionnaire in the Appendix.

It is important to note that the components of the containment systems of High Temperature Gas Cooled Small Modular Reactors (HTG-SMR) can be variously named. Such structures can, for example, be called ‘containment’, ‘reactor building’, ‘citadel building’, ‘vented low pressure containment (VLCP)’, ‘confinement structure’, or ‘functional containment’; in addition, the fuel is claimed to contribute significantly to the confinement function during accidents. It is important to note that the functional intent of the original design requirements is not lost in the choice of terminology.

In 2020, the IAEA published TECDOC-1936 [3] which focused on the applicability of the IAEA design safety requirements established in SSR-2/1 [2] to the SMR reactor technologies intended for land-based stationary deployment in the near-term, i.e., Light Water Cooled SMRs (LW-SMRs) and HTG-SMRs.

IAEA’s TECDOC-1936 [3] suggested the interpretation of Requirement 54 in SSR-2/1 [2] as follows: *“The term ‘containment system’ is to be interpreted here as a ‘reactor functional containment’ consisting of multiple barriers, internal and external to the reactor, including the reactor building*”. The justification for the Suggested Interpretation is that “*the expected contribution of the different barriers of the containment system of HTG-SMRs to the fulfilment of the safety functions of the NPP is different than in the case of the traditional LWRs. In HTG-SMRs, the fuel acts as the dominant contributor to the confinement function, and less importance is placed on the containment structure (reactor building). Multiple barriers are provided to control the release of radioactivity to the environment and to ensure that the ‘reactor functional containment’ design conditions important to safety are not exceeded in any of the plant states.”*

To avoid issues related to inconsistent terminology, and to provide clarity, please see Figure 1 that further explains the concept of functional containment.

Functional containment

The IAEA Safety and Security Glossary 2022 [1] does not provide a definition for a functional containment applicable to advanced non-LWRs without a pressure retaining containment structure/building. A functional containment is defined in USNRC Reg. Guide 1.232 [4] as:

“a barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, AOOs, and accident conditions.”

This definition most closely aligns with discussions held within the Working Group, noting the three functions from Requirement 54 of SSR-2/1 [2].

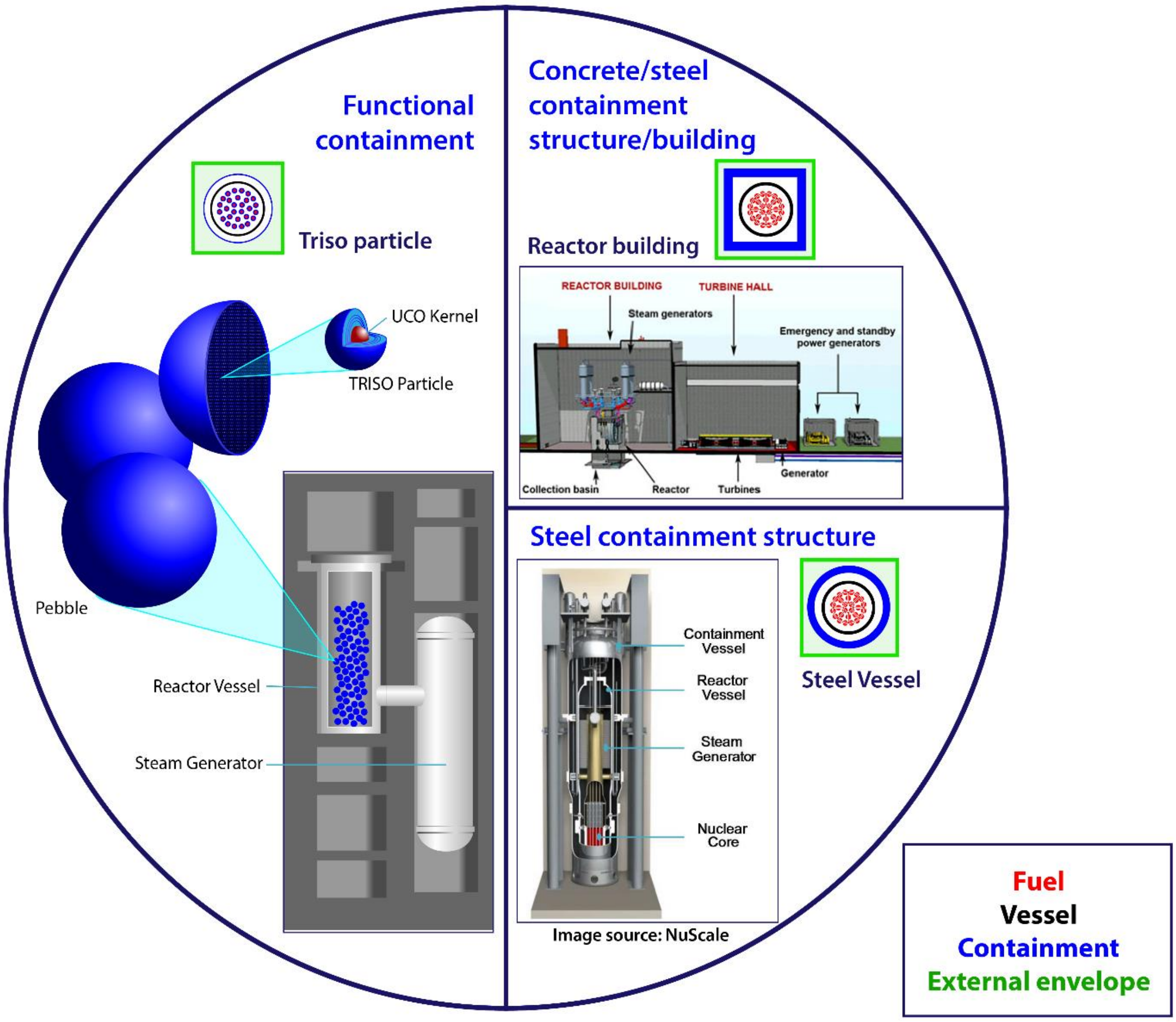

Figure 1: Different types of Containment

3. COMMON POSITIONS

The position of the Working Group is that irrespective of technology, the adequacy of the design of containment systems should be judged considering its features (e.g., the design shall have barriers, robustness, prevention of consequential failures) and overall effectiveness, consistent with a risk-informed and performance-based regulatory approach. Common Positions presented in this Report are the product of:

1. discussions at WG's meetings, and
2. Member States' answers to a Questionnaire in the Appendix of this report (for comprehensiveness, the raw answers are provided in the annexes A to G).

3.1.BARRIERS FOR DID

The "containment system" is part of DiD both conceptually and as means to prevent radioactive releases. Traditionally, containment systems are especially credited in Levels 3 and 4 DiD, to prevent or minimize the consequences of radioactive releases. Currently, non-LWR SMR designs are seeking to balance enhancements in safety features at Levels 1-3 DiD with less onerous engineering measures at Level 4 DiD.

Safety features at Level 4 DiD are necessary to assure fundamental safety functions, particularly to prevent release of radionuclides in severe accident scenarios so far as is reasonably practicable.

The methods for achieving Level 4 DiD for LWRs are traditionally by having pressure retaining, leak tight containment systems. Some LW-SMRs may not have traditional containment structures (steel lined, steel reinforced concrete containments). Instead, they may have steel containment vessels, which still provide pressure retention and leak tightness. Some proposed SMR designs include reactors operating at low pressure (e.g., molten salt reactors, sodium cooled reactors) or reactors using TRISO fuels (e.g., HTG-SMRs). For such reactors, there may be different ways of meeting expectations for DiD.

Common Position 1

> The design should include multiple independent and diverse means (DiD) to ensure that the function of containment is met for all operating states in accordance with SSR-2/1 [2].

Common Position 2

> Due to an SMR's compact size, the independence of the barriers could be more challenging to achieve for SMRs than for large reactors. In accordance with the DiD approach, the design should ensure that measures are included at each level. The measures included at any particular level should remain independent as far as practicable of those at all other levels.

3.2.GRADED APPROACH

TECDOC-1936 [3] recognizes that some wording in SSR-2/1 [2] allows the use of graded approach. For example, Requirement 58: Control of containment conditions, Paragraph 6.28 states:

"6.28. The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled."

The use of the term 'acceptably low levels' allows a graded approach depending on the requirements about the SSCs credited in the safety case.

For some SMR designs, the claims are made that some specific regulatory requirements pertaining to containment systems do not apply or are not technically relevant. Three such examples for LW- SMRs using containment vessels are for:

i. venting/purging – certain SMRs do not make a claim to incorporate a purge/venting system within their containments to be designed to regulatory requirements,

ii. penetrations – certain SMR designs do not rely on containment penetrations to be designed to regulatory requirements to provide access,

iii. requirement to perform a periodical integrated leak rate test (for SMR designs with metallic containment vessel).

Common Position 3

> Designs may be based on a graded approach in assessments of the novel containment systems to achieve safety, security and safeguards objectives. Nonetheless, a safety case must be presented to the regulatory authority to demonstrate that the proposed containment system design can and will comply with the overarching licensing requirements.

3.3.PROTECTION AGAINST HAZARDS

External events have the potential to penetrate multiple layers of DiD and cause multi-unit or multi-module accidents (where applicable) if they are not adequately addressed in the design.

Common Position 4

> Regardless of how the containment systems are designed, provisions are required to prevent accidents associated with internal and external (natural and manmade, accidental or intentional) hazards.

Common Position 5

> Where containment systems are shared among the units/modules, the design should take account of the potential hazards such arrangement may introduce.

Common Position 6

> Depending on the siting considerations (for example, for the underground/submerged containments or for floating SMR installations), the design of the containment systems needs to consider such potential specific hazards this arrangement may introduce.

3.4.ACCIDENT CONDITIONS

The physical configuration and layout of SMRs, especially the ones based on novel and advanced technologies, may be very different from typical large LWRs. It is necessary to identify all areas within the SMR containing radioactive material to determine where the actual release barriers providing confinement should be located. The identification of severe accident scenarios should consider a full range of initiating events for which accident progression should be assessed based on justified assumptions concerning the credible degree of barrier degradation. For this purpose, probabilistic assessment can be used in a complementary manner, but it should not be used solely to screen out low frequency events, since measures at Level 4 are intended to address such events. This is in accordance with SSR-2/1 [2] which reinforces that practical elimination should not be claimed solely based on compliance with a probabilistic cut-off value, but should primarily be justified by design provisions, and in some cases also strengthened by operational provisions. Moreover, a justification for practical elimination should be based on a deterministic analysis taking account of uncertainties due to the limited knowledge of certain physical phenomena.

Common Position 7

> The choice of the design extension conditions (DECs) should be explained and justified. For this purpose, probabilistic assessment should be used in a complementary manner and not be used as a sole justification to screen out low frequency events.

Common Position 8

> Regardless of how the containment function is met, all designs should demonstrate that there will be no early or large release under any accident conditions. Requirement 56 of SSR-2/1 [2] specifically applies including Paragraphs 6.22, 6.23, and 6.24, and Requirement 58, Paragraphs 6.27, 6.28 B and 6.30 specifically apply. For more information, see TECDOC-1936 [3].

Common Position 9

> Functional containment should be designed to minimize ingress of substances that may have a negative impact on SSCs important to safety. For example, for HTG-SMRs, functional containment should be designed to minimize air and water ingress, which can lead to oxidation of the core and possibly other SSCs in case of a depressurization accident. Requirement 57, Paragraph 6.26 of SSR-2/1 [2] specifically applies. For more information, see TECDOC-1936 [3].

Common Position 10

> The design should provide for the control of fission products, hydrogen, oxygen, and other substances from the containment system atmosphere as stated in SSR-2/1 [2], Paragraph 6.29.

Common Position 11

> For designs that may experience high energy releases in accidental conditions, the capability to remove heat from the containment should be ensured, in line with SSR-2/1 [2], specifically Paragraph 6.28.

3.5.PERSONNEL ACCESS

Due to an SMR's compact size and design, access by personnel for various activities (for example, MTIR, security inspections and safeguards inspections) can offer different challenges. The access arrangements should take into consideration design-specific hazards without compromising the containment system design intent.

Common Position 12

> For MTIR on or off site, the design should provide for suitable access arrangements for the 3S (safety, security and safeguards) structures, systems and components (SSCs). Requirement 57, Paragraph 6.25 of SSR-2/1 [2] applies. For more information, see TECDOC-1936 [3].

Common Position 13

> Design should accommodate the IAEA safeguards activities and provide physical access when required.

3.6.AGING AND DEGRADATION

Novel SMR technologies and aspects related to their siting (for example, submerged containments and underground construction) may introduce unique degradation mechanisms of the containment systems. The degradation rate may also differ from that traditionally experienced in the nuclear industry.

Common Position 14

> Considering novel SMR designs that may produce different and/or more rapid containment system degradation, the design should provide for the aging management for the required lifetime of an SSC. This also includes monitoring of the degradation of the SSC associated with containment function. Aspects related to siting (for example, submerged containments and underground construction) may require novel inspection techniques.

3.7.LEAKAGE

Leakage rates for the containment system design are an important assumption in the safety analysis to demonstrate that the regulatory dose limits are met. As specified in the introductory text, some leakage rate assumptions are more significant in certain designs (i.e., designs requiring strict limits on leak tightness).

Common Position 15

> The leakage rates assumed in the design of the containment system should be justified and demonstrated to the regulator. The design should provide for the verification that the designed leakage rates are not exceeded for the required lifetime. If for a design there is no claim on the need for a leakage rate on the containment structure, detailed justification and demonstration of the adequacy of such claim must be provided to the regulator.

3.8. CONTROL AND CLEANUP OF CONTAINMENT SYSTEM ATMOSPHERE

Control and cleanup of containment system atmosphere is important to ensure that the functionality or integrity of SSCs important to safety inside the containment, are not compromised by environmental conditions (e.g., leading to degradation due to high radiation fields and corrosion over time).

Common Position 16

> The design should provide for the control and cleanup of fission products, hydrogen, oxygen and other substances that may have a negative impact on SSCs important to safety, from the containment system atmosphere in line with SSR-2/1 [2], Paragraph 6.29.

3.9. REGULATORY REQUIREMENTS/EXPECTATIONS

During the course of Working Group discussions, it was clear that Member States recognize the importance of performance of containment systems. The regulatory expectations provided by Member States are included in the annexes A to G of this report. The focus of past regulatory effort was predominantly for large LWRs. This brought to light a potential gap for Member States to review their guidance available for SMRs.

Common Position 17

> Regulators should strive and continue to develop or review regulatory requirements and guidance pertaining to SMR technologies where appropriate. This is especially true in case of non-LW-SMRs, where containment system designs are substantially different from typical large LWRs and may change the emphasis of which particular SSCs are important.

Common Position 18

> The IAEA should continue to assess the extent to which the current safety standards address the safety of SMRs and develop guidance to address the identified gaps. We commend the ongoing work by the IAEA in this area and recommend that the SMR Regulators´ Forum considers it when the results are in hand.

APPENDIX: CONTAINMENT QUESTIONNAIRE

Each state please provide a short summary of the regulatory approach and requirements for the following:

Terminology

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

a) "Means" or "Provisions" versus "System(s)"
b) Containment
c) Confinement
d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

Specific requirements

1) How requirements and guidance are articulated:
    a. Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?
    b. What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?
2) Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?
3) Safety Classification:
    a. Under what conditions are containment provisions classified as safety systems?
    b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?
4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?
5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?
6) Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)
7) Do you have specific requirements/limitations for other penetrations (e.g. pipe-runs, electrical/I&C cabling)?
8) How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?

9) How are your requirements expressed to address protection from internal and external hazards?
10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?
11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?
12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?
13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/commissioning/in service)?
14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?
15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?
16) What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?
17) How do you articulate requirements for managing containment ageing and degradation?
18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?
19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?
    a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?
    b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?
20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

REFERENCES

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LIST OF ACRONYMS AND ABBREVIATIONS

ALARA	as low as reasonably achievable
AOO	anticipated operational occurrence
DBA	design basis accident
DEC	design extension condition
DiD	Defence-in-Depth
DSA	Design and Safety Analysis
EID	Evolutionary Innovative Design
HTGR	High Temperature Gas Cooled Reactor
I&C	Instrumentation and Control
LOCA	loss of coolant accident
LWR	light water-cooled reactor
MTIR	maintenance, testing, inspection and repair
NPP	nuclear power plant
SeBD	Security-by-design
SMR	Small Modular Reactor
SSC	structures, systems and components
TRISO	tristructural isotropic
USNRC	U.S. Nuclear Regulatory Commission
WCR	Water Cooled Reactors
WG	working group
VLCP	vented low pressure containment

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ANNEX A: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT - CNSC (CANADA)

Terminology

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

a) “Means” or “Provisions” versus “System(s)”

CNSC’s REGDOC-3.6 “Glossary of CNSC Terminology” [1] defines “systems” as part of the structures, systems and components (SSCs) definition, and as part of the systems important to safety (SIS) definition as:

“*structures, systems and components (SSCs) (structures, systèmes et composants [SSC])*

A general term encompassing all of the elements of a facility or activity that contribute to protection and safety. Structures are the passive elements: buildings, vessels, shielding, etc. A system comprises several components, assembled in such a way as to perform a specific (active) function. A component is a discrete element of a system. Some examples are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves.”

“systems important to safety (SIS) (systèmes importants pour la sûreté [SIS])

Systems of a reactor facility associated with the initiation, prevention, detection or mitigation of any failure sequence and that have an impact in reducing the possibility of damage to fuel, associated release of radionuclides or both.

OR

With respect to reliability programs for a reactor facility, those structures, systems and components of the facility that are associated with the initiation, prevention, detection or mitigation of any failure sequence and that have the most significant impact in reducing the possibility of damage to fuel, associated release of radionuclides or both.”

A similar definition cannot be found in the glossary (REGDOC-3.6) for “means” and “provisions”, but it can be inferred from other regulatory documents (i.e. REGDOC-2.5.2). In CNSC regulatory documents, means/provisions imply more than systems. For example, doses to nuclear energy workers are maintained below the regulatory dose limit through administrative <u>means</u> and ALARA.

b) Containment

CNSC’s REGDOC-3.6 “Glossary of CNSC Terminology” [1] defines containment as:

“containment

A method or physical structure designed to prevent or control the release of nuclear or hazardous substances. Some examples are:

- *for waste management: a barrier system that controls releases to the environment through different chemical and physical applications*
- *for reactor facilities: see containment structure*

…

containment envelope

Structures and components that provide a pressure-retaining barrier to prevent or limit the escape of any radioactive matter that could be released from a nuclear reactor.

containment structure

For reactor facilities, a physical structure designed to prevent uncontrolled release and dispersion of nuclear substances.

containment system

Has the same meaning as in the IAEA Regulations. (Source: Packaging and Transport of Nuclear Substances Regulations, 2015)

OR

See containment structure."

REGDOC-2.5.2 [2] specifies the CNSC design requirements for Nuclear Power Plants – including design requirements for containment structures. These requirements were developed in the context of traditional water-cooled reactors and do not necessarily apply to all reactor designs. Under section 8.6.1 of REGDOC-2.5.2, it is required that:

"*Each nuclear power reactor shall be installed within a containment structure, so as to minimize the release of radioactive materials to the environment during operational states and DBAs.*

Containment shall also assist in mitigating the consequences of DECs. In particular, the containment and its safety features shall be able to perform their credited functions during DBAs and DECs, including melting of the reactor core. To the extent practicable, these functions shall be available for events more severe than DECs".

In REGDOC-2.5.2 many sections deal with containment requirements, including but not limited to:

- 8.6 Containment
    - 8.6.1 General
    - 8.6.2 Strength of the containment structure
    - 8.6.3 Capability for pressure tests
    - 8.6.4 Leakage
    - 8.6.5 Containment penetrations

- 8.6.6 Containment isolation
- 8.6.7 Containment airlocks
- 8.6.8 Internal structures of the containment
- 8.6.9 Containment pressure and energy management
- 8.6.10 Control and cleanup of the containment atmosphere
- 8.6.11 Coverings, coatings and materials
- 8.6.12 Design extension conditions

- 7.15 Civil structure
  - 7.15.1 Design
  - 7.15.2 Surveillance
  - 7.15.3 Lifting and handling of large loads
- 7.22 Robustness against malevolent acts
  - 7.22.1 Design principles
  - 7.22.2 Design methods
  - 7.22.3 Acceptance criteria

c) Confinement

CNSC's REGDOC-3.6 "Glossary of CNSC Terminology" [1] defines confinement as:

"confinement boundary

A continuous boundary without openings or penetrations, which prevents the release of radioactive materials out of the enclosed space.
Note 1: For small or research reactors, confinement, or confinement boundary, is the equivalent of a power reactor containment boundary but does not have significant pressure-retaining capability.
Note 2: For packaging and transport of nuclear substances, confinement means preserving criticality safety, and containment means containing radioactive material."

REGDOC-2.5.2 recognizes that specific technologies may use alternatives to containment structures. REGDOC 2.5.2 states that if a design other than a water-cooled reactor is to be considered for licensing in Canada, the design would be subject to the safety objectives, high-level safety concepts and safety management requirements associated with REGDOC-2.5.2. The burden of proof falls on the proponent to either demonstrate that REGDOC-2.5.2 requirements can be met, or an alternative approach is needed where the alternative approach, such as confinement, would result in an equivalent or superior level of safety.

Because REGDOC-2.5.2 requirements were developed in the context of traditional water-cooled reactors and do not necessarily apply to all reactor designs, CNSC developed RD-367 [3] for small reactors (defined as thermal output of 200 MW or less). The intent of RD-367 was to be technologically neutral and contains requirements pertaining to confinement. Definition of confinement is given in Section 8.6 of RD-367 which states that:

"*Confinement is a fundamental safety function and a means to achieve this safety function shall be provided for any reactor facility."*

More detailed requirements on the features of confinement are given in Section 8.6.1 of RD-367, for example:

"The confinement shall be designed to ensure that a release of radioactive material following an accident involving disruption of the core is within acceptable limits. The confinement shall include physical barriers designed to prevent or mitigate an unplanned release of radioactive material to the environment during normal operation, AOOs, DBAs and, to the extent practicable, BDBAs."

d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

Over the last 2 years, CNSC staff has been involved in the interpretation of containment requirements for non-water cooled and advanced reactors. The concept of functional containment is under deliberation, however, there are no conclusive results to share at this point in time. For ongoing SMR reviews, CNSC would assess a functional containment by applying requirements in Section 11 "Alternative Approaches" of REGDOC-2.5.2 in conjunction with all other relevant requirements.

Specific requirements

1) **How requirements and guidance are articulated:**
    a. **Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?**

In Canada, CNSC requirements and guidance which address structures that support containment functions are written to identify key safety objectives to be met. For containment structure, the requirements in REGDOC-2.5.2 are specified in Sections 8.6 "Containment", 7.15 "Civil structure" and 7.22 "Robustness against malevolent acts". For example, REGDOC-2.5.2, Section 8.6 "Containment" states the following:

"*8.6.1 General*

Each nuclear power reactor shall be installed within a containment structure, so as to minimize the release of radioactive materials to the environment during operational states and DBAs. Containment shall also assist in mitigating the consequences of DECs. In particular, the containment and its safety features shall be able to perform their credited functions during DBAs and DECs, including melting of the reactor core. To the extent practicable, these functions shall be available for events more severe than DECs.

The containment shall be a safety system and may include complementary design features. Both the containment system and the complementary design features shall be subject to the respective design requirements provided in this regulatory document.

The design shall include a clearly defined continuous leak-tight containment envelope, the boundaries of which are defined for all conditions that could exist in the operation or maintenance of the reactor, or following an accident.

…

8.6.2 Strength of the containment structure

The strength of the containment structure shall provide sufficient margins of safety based on potential internal overpressures, underpressures, temperatures, dynamic effects such as missile generation, and reaction-forces anticipated to result in the event of DBAs. Strength margins shall be applied to access openings, penetrations, and isolation valves, and to the containment heat removal system.

…

The containment structure shall protect systems and equipment important to safety in order to preserve the safety functions of the plant.

The design shall support the maintenance of full functionality following a DBE for all the parts of the containment system credited in the safety analysis.

The seismic design of the concrete containment structure shall have an elastic response when subjected to seismic ground motions. The special detailing of reinforcement shall allow the structure to possess ductility and energy-absorbing capacity, which permits inelastic deformation without failure."

It is important to note that in addition to REGDOC-2.5.2, CNSC uses more prescriptive requirements specified in Canadian Standard Association Group's documents such as the CSA N287 series that refer to concrete containments, N289 series for seismic design and qualification and N290 for general requirements for safety systems of nuclear power plants:

1. CSA N287 series
   - CSA N287.1: General Requirements
   - CSA N287.2: Material Requirements
   - CSA N287.3: Design Requirements
   - CSA N287.4: Construction, Fabrication, and Installation Requirements
   - CSA N287.5: Examination and Testing Requirements
   - CSA N287.6: Pre-operational Proof Testing and Leakage Testing Requirements
   - CSA N287.7: In-service Examination and Testing Requirements
2. CSA N289.1-18: General requirements for seismic design and qualification of nuclear power plants
3. CSA N290 series
   - CSA N290.3-11: Requirements for the Containment System of Nuclear Power Plants
   - CSA N290.3-16: Requirements for the Containment System of Nuclear Power Plants

- CSA N290.0-17: General requirements for safety systems of nuclear power plants

CSA standards use performance criteria for the reliability and capability of the SSCs fulfilling functions important to safety. In case when the containments are pressure vessels, the CSA N287 series are not appropriate and N290 series are not sufficient. In such cases, the vendors may use the following standards:

- CSA N285.0-17/N285.6 Series-17: General requirements for pressure retaining systems and components in CANDU nuclear power plants/Material Standards for reactor components for CANDU nuclear power plants, and
- CSA N291-15: Requirements for Safety-Related Structures for Nuclear Power Plants.

The vendor may propose the use of alternate codes and standards; however, the vendor must provide information that outlines the basis of how the alternate standards are broadly equivalent to Canadian codes and standards. This gap analysis is integral to the vendor demonstrating their understanding of Canadian requirements.

b. What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?

The baseline expectation is that the containment system is a safety system, however Section 7.1 "Safety classification of structures, systems and components" of REGDOC-2.5.2 allows for a risk-informed approach towards classification of alternatives intended to provide containment functions.

Section 11 "Alternative Approaches" of REGDOC 2.5.2 allows for flexibility while considering reactor designs which use alternatives to satisfy nuclear safety requirements. This applies to the containment functions which are intended to provide effective radionuclide barriers with robust defence in depth. Section 11 specifies that any such alternative approach shall as a minimum demonstrate equivalence to the outcomes associated with the use of the requirements in REGDOC 2.5.2.

2) Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?

Specific SSCs required to support reliable performance of the containment function are prescribed over the respective ranges for operational states, DBAs and DECs. Section 7.9.1 "General" of REGDOC-2.5.2 states:

"*The design shall include provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states, DBAs and DECs, in order to ensure that adequate information can be obtained on plant status.*

This shall include instrumentation for measuring variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, and containment, as well as instrumentation for obtaining any plant information that is necessary for its reliable and safe operation.

The design shall be such that the safety systems and any necessary support systems can be reliably and independently operated, either automatically or manually, when necessary."

Reliability requirements for the containment system are specified in the answer to question 12 below.

3) Safety Classification:

a. Under what conditions are containment provisions classified as safety systems?

The baseline expectation is that the containment system is a safety system, as explained in in the answer to question 1b.

REGDOC-2.5.2 has specific requirements to address safety classification in Section 7.1 "Safety classification of structures, systems and components":

"The design authority shall classify SSCs using a consistent and clearly defined classification method. The SSCs shall then be designed, constructed, and maintained such that their quality and reliability is commensurate with this classification.

In addition, all SSCs shall be identified as either important or not important to safety. The criterion for determining safety importance is based on:

- *safety function(s) to be performed*
- *consequence(s) of failure*
- *probability that the SSC will be called upon to perform the safety function*
- *the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation*

SSCs important to safety shall include:

- *safety systems*
- *complementary design features*
- *safety support systems*
- *other SSCs whose failure may lead to safety concerns (e.g., process and control systems)*

Appropriately designed interfaces shall be provided between SSCs of different classes in order to minimize the risk of having SSCs less important to safety adversely affecting the function or reliability of SSCs of greater importance.

Guidance

The method for classifying the safety significance of SSCs important to safety should be based primarily on deterministic methodologies, complemented (where appropriate) by probabilistic methods and engineering judgment. The safety classification of SSCs should be an iterative process that continues throughout the design process.

The SSC classification process should include the following activities:
- *review and definition of PIEs*
- *grouping and identification of bounding PIEs*
- *identification of plant-specific safety functions to prevent or mitigate the PIEs*
- *safety categorization of the safety functions, in accordance with their safety significance and role in achieving fundamental safety functions*
- *identification of SSCs that provide the safety functions*
- *assignment of SSCs to a safety class corresponding to the safety category*
- *verification of SSC classification*
- *identification of engineering design rules for classified SSCs*

This approach should be used for all SSCs including pressure retaining components, electrical, instrumentation and control (I&C) and civil structures.
…

The appropriate design rules and limits as indicated in section 7.5 are specified in accordance with the safety class of SSCs."

b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?

As implied from the REGDOC-2.5.2 Section 7.1 provided in the answer to question 3a above, if a failure of a system that supports containment and confinement functions can be accepted because such failure will result in acceptable consequences, such system may be classified at a lower safety level/class:

"The potential severity of the consequences of a function failure should be evaluated. The severity should be based on the consequences that could arise if the function was not performed. The consequences of a function failure should be made assuming that the safety functions belonging to the subsequent level of defence in depth remain functional."

4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

Containment design basis for internal and external events are addressed in the answer to question 9 below.

Section 6.1 "Application of defence in depth" of REGDOC-2.5.2 gives the principles behind the safety concept of DiD:

"6.1 Application of defence in depth

The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable.

Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence.
…
Level Four
Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable.

Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures.
…

Guidance
IAEAINSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth.

Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants.

The application of defence in depth in the design should ensure the following:

- *The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety.*
- *The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs).*
- *The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices.*
- *The design (e.g., in safety design guides, management system programs) should provide:*
    - *levels of defence in depth that are addressed by individual SSCs*
    - *supporting analysis and calculation*
    - *evaluation of operating procedures*

- *The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity.*
- *The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design.*
- *Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in the completeness of the supporting safety analyses should be documented and flagged as issues to be queried.*

To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences of the accident.

The independence between all levels of defence should be achieved, in particular, through diverse provisions. The strengthening of each of these levels separately would provide, as far as reasonably achievable, an overall reinforcement of defence in depth. For example, the use of dedicated systems to deal with DECs ensures the independence of the fourth defence level."

Requirements on containment systems therefore exist for Levels 1-4 DiD and are considered of particular importance for Level Four DiD.

5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?

REGDOC-2.5.2 has specific requirements to address leak tightness. Section 8.6.4 "Leakage" specifies:

"*Leakage rate limits*

The safety leakage rate limit shall assure that:

1. *Normal operation release limits are met*
2. *AOOs and DBAs will not result in exceeding dose acceptance criteria*

The design leakage rate limit shall be:

1. *Below the safety leakage rate limit*
2. *As low as is practicably attainable*
3. *Consistent with state-of-the-art design practices*

Test acceptance leakage rate limits

A test acceptance leakage rate shall provide the maximum rate acceptable under actual measurement tests.

Test acceptance leakage rate limits shall be established for the entire containment system, and for individual components that can contribute significantly to leakage.

The containment structure and the equipment and components affecting the leak tightness of the containment system shall be designed to allow leak rate testing:

1. *For commissioning, at the containment design pressure*
2. *Over the service lifetime of the reactor, in accordance with applicable codes and standards*

The design shall provide ready and reliable detection of any significant breach of the containment envelope."

6) Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)

REGDOC-2.5.2 has requirements for Containment airlocks in Section 8.6.7:

"*Personnel access to the containment shall take place through airlocks that are equipped with doors that are interlocked to ensure that at least one of the doors is closed during operational states, DBAs and DECs.*

Where provision is made for entry of personnel for surveillance or maintenance purposes during normal operation, the design shall specify provisions for personnel safety, including emergency egress. This requirement shall also apply to equipment air locks.

Guidance
Containment openings for the movement of equipment or material through the containment should be designed to be closed quickly and reliably, in the event that isolation of the containment is required.

The need for access by personnel to the containment should be minimized. Following an accident, access to the containment for the purpose of ensuring the safety of the facility (for either short or long term) should not be necessary."

In addition, REGDOC-2.5.2 has specific requirements to address other penetrations, as specified in Section 8.6.5 "Containment penetrations":

"The number of penetrations through the containment shall be kept to a minimum.

All containment penetrations shall be subject to the same design requirements as the containment structure itself, and shall be protected from reaction forces stemming from pipe movement or accidental loads, such as those due to missiles generated by external or internal events, jet impact, and pipe whip.

All penetrations shall be designed to allow for periodic inspection and testing.

If resilient seals such as elastomeric seals, electrical cable penetrations, or expansion bellows are used with penetrations, they shall have the capacity for leak testing at the containment design pressure. To demonstrate continued integrity over the lifetime of the plant, this capacity shall support testing that is independent of determining the leak rate of the containment as a whole.

Guidance
Keeping the number of penetrations through the containment to a minimum should consider the need for separation and redundancy, and be consistent with modern designs."

Other REGDOC-2.5.2 requirements pertaining to containment penetrations can be found in:

- Section 7.15.1 "Design" requires that:
  "*Ultimate internal pressure capacity should be provided for the containment building structures including containment penetrations.*"

- Section 8.6.2 "Strength of the containment structure" specifies that strength margins shall be applied to access openings and penetrations.

- Section 8.6.6 "Containment isolation":
  "Piping systems that penetrate the containment system shall have isolation devices with redundancy, reliability, and performance capabilities that reflect the importance of isolating the various types of piping systems."

- For ECC piping, in Section 8.5 "Emergency core cooling system":
  "As a piping extension to containment, it meets the requirements for metal penetrations of containment."

7) Do you have specific requirements/limitations for other penetrations (e.g. pipe-runs, electrical/I&C cabling)?

REGDOC-2.5.2 has specific requirements to address penetrations including pipe-runs, electrical/I&C cabling, which are specified in the answer to question 6, above.

8) How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?

REGDOC-2.5.2 states in Section 8.6.1 "General" under Section 8.6 "Containment" that:

"The containment shall include at least the following subsystems:

1. *the containment structure and related components*
2. *equipment required to isolate the containment envelope and maintain its completeness and continuity following an accident*

3. *equipment required to reduce the pressure and temperature of the containment and reduce the concentration of free radioactive material within the containment envelope*
4. *equipment required for limiting the release of radioactive material from the containment envelope following an accident"*

The objective of containment isolation is to prevent radioactive releases to the environment that exceed prescribed limits, by maintaining containment's leak tightness following an accident. More detailed requirements are given in Section 8.6.6 as follows:

"8.6.6 Containment isolation

Each line of the reactor coolant pressure boundary that penetrates the containment, or that is connected directly to the containment atmosphere, shall be automatically and reliably sealed. This requirement is essential to maintaining the leak tightness of the containment in the event of an accident and preventing radioactive releases to the environment that exceed prescribed limits.

Automatic isolation valves shall be positioned to provide the greatest safety upon loss of actuating power.

Piping systems that penetrate the containment system shall have isolation devices with redundancy, reliability, and performance capabilities that reflect the importance of isolating the various types of piping systems. Alternative types of isolation may be used where justification is provided.

Where manual isolation valves are used, they shall be readily accessible and have locking or continuous monitoring capability.

Reactor coolant system auxiliaries that penetrate containment

Each auxiliary line that is connected to the reactor coolant pressure boundary, and that penetrates the containment structure, shall include two isolation valves in series. The valves shall be normally arranged with one inside and one outside the containment structure.

Where the valves provide isolation of the heat transport system during normal operation, both valves shall be normally in the closed position.
…

Systems connected to containment atmosphere

Each line that connects directly to the containment atmosphere, that penetrates the containment structure and is not part of a closed system, shall be provided with two isolation barriers that meet the following requirements:

1. *two automatic isolation valves in series for lines that may be open to the containment atmosphere*

2. *two closed isolation valves in series for lines that are normally closed to the containment atmosphere*
3. *the line up to and including the second valve is part of the containment envelope*

Closed systems

All closed piping service systems shall have at least one single isolation valve on each line penetrating the containment, with the valve being located outside of, but as close as practicable to, the containment structure.
…"

9) How are your requirements expressed to address protection from internal and external hazards?

REGDOC-2.5.2 recognizes in Section 7.4 "Postulated initiating events" that:

"Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs, as well as operator errors, common-cause internal hazards, and external hazards.

For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site."

Requirements for protection from internal and external hazards are specified in REDOC-2.5.2 Section 7.4.1 "Internal hazards", Section 7.4.2 "External hazards" and Section 7.4.3 "Combination of events":

"7.4.1 Internal hazards

SSCs important to safety shall be designed and located in a manner that minimizes the probability and effects of hazards (e.g., fires and explosions) caused by external or internal events.

The plant design shall take into account the potential for internal hazards, such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures shall be provided to ensure that nuclear safety is not compromised.

Internal events which the plant is designed to withstand shall be identified, and AOOs, DBAs and DECs shall be determined from these events.

The possible interaction of external and internal events shall be considered, such as external events initiating internal fires or floods, or that may lead to the generation of missiles.

Guidance

The design should take into account specific loads and environmental conditions (temperature, pressure, humidity, radiation) imposed on structures or components by internal hazards.
…

7.4.2 External hazards

All natural and human-induced external hazards that may be linked with significant radiological risk shall be identified. External hazards which the plant is designed to withstand shall be selected, and classified as DBAs or DECs.

Various interactions between the plant and the environment, such as population in the surrounding area, meteorology, hydrology, geology and seismology shall be identified during the site evaluation and environmental assessment processes. These interactions shall be taken into account in determining the design basis for the NPP.

Applicable natural external hazards shall include such hazards as earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions. Human induced external hazards shall include those that are identified in the site evaluation, such as potential aircraft crashes, ship collisions, and terrorist activities.
…

7.4.3 Combination of events

Combinations of randomly occurring individual events that could credibly lead to AOOs, DBAs, or DECs shall be considered in the design. Such combinations shall be identified early in the design phase, and shall be confirmed using a systematic approach.

Events that may result from other events, such as a flood following an earthquake, shall be considered to be part of the original PIE.

…"

10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

For civil design, the design specifications shall define all loads and load combinations consistent with Section 7.15:

"*7.15 Civil structure*

7.15.1 Design

…

Civil structures important to safety shall be designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, AOO, DBA and DEC conditions, including external hazards. The serviceability considerations shall include, without being limited to, deflection, vibration, permanent deformation, cracking, and settlement.

The design specifications shall also define all loads and load combinations, with due consideration given to the probability of concurrence and loading time history.

Environmental effects shall be considered in the design of civil structures and the selection of construction materials. The choice of construction material shall be commensurate with the designed service life and potential life extension of the plant.

The plant safety assessment shall include structural analyses for all civil structures important to safety.

Guidance

The design authority should provide the design principles, design basis requirements and criteria, and applicable codes and standards, design and analysis procedures, the assumed boundary conditions and the computer codes used in the analysis and design.

All internal and external hazard loads are specified in section 7.4. Earthquake design input loads and impacts of malevolent acts, including large aircraft crash can be found in sections 7.13 and 7.22, respectively.

Load categories corresponding to the plant states are defined in this section so as to demonstrate structural performances as follows:

- *normal condition loads which are expected during the assumed design life of the NPP*
- *AOO loads (or severe environmental loads)*
- *DBA loads (or abnormal or extreme environmental loads)*
- *DEC loads (or beyond-design loads)*

The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria.

…

Containment structure

The design should specify the safety requirements for the containment building or system, including, for example, its structural strength, leak tightness, and resistance to steady-state and transient loads (such as those arising from pressure, temperature, radiation, and mechanical impact) that could be caused by postulated internal and external hazards. In addition, the design should specify the safety requirements and design features for the containment internal structures, (such as the reactor vault structure, the shielding doors, the airlocks, and the access control and facilities).

The design of the containment structure should include:

- *base slab and sub-base*
- *containment wall and dome design*
- *containment wall openings and penetrations*
- *pre-stressing system*
- *containment liner and its attachment method*

The design pressure of the containment building should be determined by increasing by at least 10% the peak pressure that would be generated by the DBA (refer to clause 4.49 of IAEA NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants).

Ultimate internal pressure capacity should be provided for the containment building structures including containment penetrations.

If the containment building foundation is a common mat slab which is not separated from the other buildings foundation, the impact should be evaluated.

…"

11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

Requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials are specifies in REGDOC-2.5.2, Section 6.1.1 "Physical barriers" which states:

"To ensure the overall safety concept of defence in depth is maintained, the design shall provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment. Such barriers shall include the fuel matrix, the fuel cladding, the reactor coolant pressure boundary, and the containment. In addition, the design shall provide for an exclusion zone.

To the extent practicable, the design shall prevent:
- *challenges to the integrity of physical barriers*
- *failure of a barrier when challenged*
- *failure of a barrier as a consequence of failure of another barrier*
- *the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences*

The design shall also allow for the fact that the existence of multiple levels of defence does not normally represent a sufficient basis for continued power operation in the absence of one defence level."

Section 7.22.1 "Design principles" specifies the requirements regarding barriers for protection against malevolent acts:

"Consistent with the concept of defence in depth, the design shall provide multiple barriers for protection against malevolent acts, including physical protection systems, engineered safety provisions, and measures for post-event management, as

appropriate. The failure of a preceding barrier shall not compromise the integrity and effectiveness of subsequent barriers."

For extreme events, which are a subset of BDBTs, Section 7.22.3 "Acceptance criteria" allows limited degradation of containment barrier:

"For extreme events, there shall be at least one means of reactor shutdown and core cooling. Degradation of the containment barrier may allow the release of radioactive material; however, the degradation shall be limited. In these cases, the response shall include onsite and offsite emergency measures."

Regarding the definition of tasks/functions of containment/confinement barrier(s), the requirements are provided in Section 8.6 "Containment".

12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

CNSC does have quantitative reliability requirements for containment systems, as specified in Section 7.6 "Design for reliability" of REGDOC-2.5.2:

"7.6 Design for reliability

All SSCs important to safety shall be designed with sufficient quality and reliability to meet the design limits. A reliability analysis shall be performed for each of these SSCs.

Where possible, the design shall provide for testing to demonstrate that the reliability requirements will be met during operation.

The safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 10-3.

The reliability model for each system may use realistic failure criteria and best-estimate failure rates, considering the anticipated demand on the system from PIEs.

Design for reliability shall take account of mission times for SSCs important to safety.

The design shall take into account the availability of offsite services upon which the safety of the plant and protection of the public may depend, such as the electricity supply and external emergency response services.

Guidance
The design for reliability is based on meeting applicable regulatory requirements and industry standards. The design should provide assurance that the requirements of CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants, will be met during operation. Not all SSCs important to safety identified in the design phase will necessarily be included in the reliability program.

The following principles are applied for SSCs important to safety:

- *the plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk importance of these SSCs*
- *these SSCs do not degrade to an unacceptable level during plant operations*
- *the frequency of transients posing challenges to SSCs is minimized*
- *these SSCs function reliably when challenged*

The reliability of SSCs assumed in the design stage needs to be realistic and achievable.

Deterministic analysis or other methods may be used if the PSA lacks effective models or data to evaluate the reliability of SSCs."

13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/commissioning/in service)?

REGDOC-2.5.2 has a section specifying the requirements for testing, examinations, inspections, and maintenance of SSCs important to safety, which typically include containment:

"7.14 In-service testing, maintenance, repair, inspection and monitoring

In order to maintain the NPP within the boundaries of the design, the design shall be such that the SSCs important to safety can be calibrated, tested, maintained and repaired (or replaced), inspected, and monitored over the lifetime of the plant.

These activities shall be performed to standards commensurate with the importance of the respective safety functions of the SSCs, with no significant reduction in system availability or undue exposure of the site personnel to radiation.
…

The design shall identify the needs for related testing when specifying the commissioning requirements for the plant.

The design shall provide the means to gather baseline data, in order to support maintenance-related testing, inspection and monitoring.

Guidance

While in-service testing, maintenance, repair, inspection and monitoring take place primarily during the operating phase of the plant's lifecycle, the NPP is designed to permit the effective implementation of these activities during operation. In particular, the reactor core should be designed to permit the implementation of a material surveillance program to monitor the effects of service conditions on material properties throughout the operating life of the reactor.

The design should establish a technical basis of SSCs that require in-service testing, maintenance, repair, inspection and monitoring.

The development of strategies and programs to address in-service testing, maintenance, repair, inspection and monitoring is a necessary aspect of the plant design phase. The strategies and programs to be implemented for these in-service activities should be developed so as to ensure that plant SSCs remain capable and available to perform their safety functions. The design should incorporate provisions recognizing the need for in-service testing, maintenance, repair, inspection and monitoring, as well as to permit the repair, replacement and modification of those SSCs likely to require such actions, due to anticipated operating conditions. In addition, activities which need to be carried out during the construction and commissioning phases should be identified, in order to provide a meaningful baseline data of the plant, at the outset of its operating life.

The strategies should include well-planned and effective programs for evaluating and trending SSCs performance, coupled with an optimized preventive maintenance program.

…

If risk informed in-service inspection methodologies are used when defining the scope of an inspection program, the methodology should be clearly documented.

SSCs important to safety should be designed and located to make surveillance and maintenance simple, to permit timely access, and in case of failure, to allow diagnosis and repair, and minimize risks to maintenance personnel.

Means provided for the maintenance of SSCs important to safety should be designed such that the effects on the plant safety are acceptable."

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

Requirements for protection from internal and external hazards are addressed under the response to question 9 (REGDOC-2.5.2, Section 7.4 "Postulated initiating events").

Section 7.15.1 "Design" requires that:

"*Civil structures important to safety shall be designed and located so as to minimize the probabilities and effects of internal hazards such as fire, explosion, smoke, flooding, missile generation, pipe whip, jet impact, or release of fluid due to pipe breaks*.

External hazards such as earthquakes, floods, high winds, tornadoes, tsunamis, and extreme meteorological conditions shall be considered in the design of civil structures."

As explained in the Guidance to the same section:

"All internal and external hazard loads are specified in section 7.4. Earthquake design input loads and impacts of malevolent acts, including large aircraft crash can be found in sections 7.13 and 7.22, respectively.

Load categories corresponding to the plant states are defined in this section so as to demonstrate structural performances as follows:

- *normal condition loads which are expected during the assumed design life of the NPP*
- *AOO loads (or severe environmental loads)*
- *DBA loads (or abnormal or extreme environmental loads)*
- *DEC loads (or beyond-design loads)*

The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria."

15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

REGDOC-2.5.2 considers DECs to be the subset of BDBAs that are considered in the design. REGDOC-2.5.2 contains Section 7.3.4 "Design extension conditions" which specifies DECs requirements for containment, i.e. that for DECs with severe core damage, the containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage:

"7.3.4 Design extension conditions

The design authority shall identify the set of design-extension conditions (DECs) based on deterministic and probabilistic methods, operational experience, engineering judgment and the results of research and analysis. These DECs shall be used to further improve the safety of the NPP by enhancing the plant's capabilities to withstand, without significant radiological releases, accidents that are either more severe than DBAs or that involve additional failures.

The design shall be such that plant states that could lead to significant radioactive releases are practically eliminated. For plant states that are not practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures.

Complementary design features shall be provided to cope with DECs. Their design shall be based on a combination of phenomenological models, engineering judgments, and probabilistic methods.

…

The design shall identify a radiological and combustible gas accident source term, for use in the specification of the complementary design features for DECs. This source

term is referred to as the reference source term and shall be based on a set of representative core damage accidents established by the design authority.
…

7.3.4.1 Severe accidents within design extension conditions

The design shall be balanced such that no particular design feature or event makes a dominant contribution to the frequency of severe accidents, taking uncertainties into account.
…

For DECs with severe core damage, the containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period.

Particular attention shall be placed on the prevention of potential containment bypass in severe accidents.
…

Guidance
…

Containment leakage in a severe accident should remain below the design leakage rate limit (as defined in section 8.6.4) for sufficient time to allow implementation of emergency measures. Beyond this time, containment leakage that would lead to exceeding the small and large release safety goals should be precluded. This may be achieved by provision of adequate filtered containment venting along with other features.
…"

16) What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?

The limiting accident scenarios are addressed in Section 7 "General Design Requirements" of REGDOC-2.5.2:

"The identified PIEs should be grouped into limiting cases, which are referred to as bounding or enveloping PIEs. Once these bounding PIEs are known and understood, the required safety functions can be identified. The number of categories and classes may be chosen to allow for graded design rules."

For LWRs, a loss of coolant accident (LOCA) or main steam line break (MSLB) would typically establish the peak containment pressure and temperature profile.

17) How do you articulate requirements for managing containment ageing and degradation?

Section 5.2 "Design management" requires that the "*plant design facilitates maintenance and aging management throughout the life of the plant.*" Section 5.7 "Design documentation" requires that:

"Design documentation shall include information to demonstrate the adequacy of the design and shall be used for procurement, construction, commissioning and safe operation, including maintenance, aging management, modification and eventual decommissioning of the NPP."

Section 7.15.1 "Design" specifies that *"the structural design should consider the impact of aging on the structure and its material".* Also, REGDOC-2.5.2 has a Section dedicated to aging, Section 7.17 "Aging and wear":

"The design shall take due account of the effects of aging and wear on SSCs. For SSCs important to safety, this shall include:

1. *an assessment of design margins, taking into account all known aging and wear mechanisms and potential degradation in operational states, including the effects of testing and maintenance processes*
2. *provisions for monitoring, testing, sampling, and inspecting SSCs so as to assess aging mechanisms, verify predictions, and identify unanticipated behaviours or degradation that may occur during operation, as a result of aging and wear"*

CNSC also uses RD-334, "Aging Management for Nuclear Power Plants" for aging and degradation management.

18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

In general, it is expected the ageing management for all SSCs complies with relevant ageing management requirements, such as those listed in Section 7.17 "Aging and wear" of REGDOC-2.5.2. Some SMR designs may take into account implicitly the ageing and wear of SSCs that contribute to the functional containment (e.g. HTGRs) such as the change in mechanical and retention properties of the TRISO particles and structural graphite with temperature and irradiation, as well as the creep and irradiation behaviour of the primary helium pressure boundary. Deterministic safety analyses are expected to confirm the safety systems effectiveness, including the containment, over the lifetime of the plant.

19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

   a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?

Although the CNSC requirements do not specifically address plant elevation, the seismic and hydrostatic loading on the containment will change depending on the grade level and would have to be addressed in the design. For example, below-grade construction is addressed in REGDOC-1.1.1 "Site Evaluation and Site Preparation for New Reactor Facilities". Section 3.5.4 "Groundwater hazards" of REGDOC-1.1.1 states:

"The applicant shall use a program of hydrogeological investigations, based on groundwater probing, monitoring data, and numerical modelling, to assess the potential effects of the groundwater flow system (groundwater level and quality) on the reactor facility, such as:

- *effects on the stability of the reactor facility's foundations*
- *effects on the integrity of the reactor facility's below-grade structures, such as wet storage bays"*

Section 3.5.5 "Geotechnical hazards" of REGDOC-1.1.1 requires that the applicant examines geological maps and other appropriate reference sources for the region to determine the existence of natural features that could affect the surface and subsurface stability of the site. For underground excavations, "*the applicant shall analyze underground instability (rock falls and underground collapses) and groundwater inflow using site-specific geotechnical and hydrogeological data to assess the potential risks to worker safety."*

b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?

CNSC does not have requirements that state explicitly that the containment should be above-ground. All requirements which apply to the traditional, above-ground containments would likely also apply to an underground containment or confinement.

As specified in the answer to question 19a, an underground structure would be required to meet seismic requirements given in Section 7.13.1 "Seismic design and classification" of REGDOC-2.5.2:

"The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE), and ensure that they are categorized accordingly. This shall apply to:

1. SSCs whose failure could directly or indirectly cause an accident leading to core damage
2. SSCs restricting the release of radioactive material to the environment
3. SSCs that assure the subcriticality of stored nuclear material
4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits".

Also, the design of the containment SSCs should meet the DBE criteria to maintain all essential attributes, such as pressure boundary integrity, leak-tightness, operability, and proper position in the event of a DBE.

For underground containments, attention should be paid to surveillance and monitoring in the vicinity of the plant to ascertain containment leak tightness.

20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

Other relevant CNSC requirements pertaining to SSCs performing a containment or confinement function include, but are not limited to requirements for Management System and Quality Assurance (CSA N286-12, Management system requirements for nuclear facilities; CSA N286.7, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants), Environmental Qualification (CSA N290.13, Environmental qualification of equipment for CANDU nuclear power plants), Reliability (CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants), Maintenance (CNSC, RD/GD-210, Maintenance Programs for Nuclear Power Plants), Aging (CNSC, RD-334, Aging Management for Nuclear Power Plants, Ottawa), Inspection (CSA N287.7, In-service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants), and Accident Management (CSA N290.6, Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident).

References:

[1] CNSC, Regulatory Document, REGDOC-3.6 “Glossary of CNSC Terminology”, April 2021.

[2] CNSC, Regulatory Document, REGDOC-2.5.2 “Design of Reactor Facilities: Nuclear Power Plants”, May 2014.

[3] CNSC, Regulatory Document, REGDOC-367 “Design of Small Reactor Facilities”, June 2011.

ANNEX B: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT – STUK (FINLAND)

Terminology

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

a) “Means” or “Provisions” versus “System(s)”

“ System” refers to “*a combination of components and structures that performs a specific function*”. There is a no formal definition for “means” or “provisions” in the Finnish framework. They may be used in some requirements though in wider context than a system (i.e. including guidelines, operations, etc).

b) Containment

Containment is defined in the following way: “*Primary containment shall refer to a pressure-proof and leak-tight building surrounding the reactor and its coolant circuit, the function of which is to protect the reactor and the coolant circuit from external events and prevent the release of radioactive substances into the environment in accidents. When the word ‘containment’ is used in Guide YVL B.6, it refers to the primary containment. The primary containment may be surrounded by a secondary containment. The purpose of the secondary containment is to make possible the recovery and processing of any radioactive substances leaking form the primary containment. For this purpose, the interim space between the primary containment and secondary containment is kept at underpressure. The secondary containment may also provide protection against external event*s.”

(It is to be noted a project for renewal of legislation, regulations and guides has been started and issues of the questionnaire are under consideration.)

c) Confinement

No specific term defined for confinement currently; this may change in future. It would be understood to mean approximately “prevention of spreading/dispersion of radioactive material”.

d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

No formal term such as “functional containment” exists. “Containment system” shall refer to the containment (structure) and its systems that are designed to isolate the containment, remove heat from inside the containment, and control radioactive substances and combustible gases in accident scenarios.”

Specific requirements

1) **How requirements and guidance are articulated:**
   a. **Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and**

guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?

b. **What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?**

The Nuclear Energy Act requires following Defence-in-Depth principle in structural design of a NPP. The Nuclear Energy Decree sets forth the limits for releases and doses in different plant conditions and also sets some other criteria for the plant as well:
"*The release of radioactive material as a result of a severe accident at a nuclear power plant may not necessitate large-scale protective measures for the population or any long-term restrictions on the use of extensive areas of land and water*."

"*In order to limit the long-term effects, the limit for atmospheric releases of caesium-137 shall be 100 terabecquerels. The possibility of exceeding the limit shall be extremely small*."

"*The possibility of a release in the early stages of an accident requiring measures to protect the population shall be extremely small.*"

A containment is required in STUK regulation on the Safety of a Nuclear Power Plant 1/Y/2018; the STUK regulations are binding and deviations are not really possible. Requirements in STUK regulations or on high level and quite generic though. Hence no detailed requirements on the containment structure for example are set in the regulation. Target for the containment function is to maintain its integrity in normal and accident conditions.

The YVL Guides set more detailed requirements; e.g there shall be a steel liner for a concrete containment. They are somewhat binding i.e. shall be fulfilled but another solution may be suggested by the licensee/license applicant, if the same safety level is achieved. Also, deviations from the requirements may be possible if justifiable. In these cases it of course shall be assessed if the related higher level goal in STUK regulation or Nuclear Energy Act/Decree is fulfilled.

2) **Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?**

No specific systems are prescribed: it is required that containment integrity and leaktightness is maintained in normal accident conditions.

3) **Safety Classification:**

   a. **Under what conditions are containment provisions classified as safety systems?**

Containment structure itself is defined as a "safety system"/structure. Functions intended to isolate the containment or maintain its' leaktightness and integrity are defined as safety systems. The assigned safety class depends on whether the function is intended for Design basis accident (SC2) or for DEC/SA (SC3)

b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?

Does the question refer to non-classified? No safety function performed and no significant release in case of non-function/failure/loss of integrity.

4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

Containment and containment systems: DEC levels 1-3 are defined as design basis and stricter requirements apply to them; e.g. the systems shall be safety class 2, fulfil N+2 criterion. Design extension conditions, i.e. common cause failures, complex sequences and extreme external conditions are also required to be taken into account; best estimate methodology can be followed and system requirements are less stringent.

Severe accidents are to be taken into account in the design; dedicated systems for them shall be implemented. This is required in the currently binding level of requirements, i.e. STUK regulation 1/Y/2018.

5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?

There are no separate requirements concerning “small releases”. It is required that nuclear power plant shall be provided with a leak tight containment system, and that leak tightness is maintained during normal operation, anticipated operational occurrences and accidents, and it is e.g. specifically required in STUK 1/Y/2018 that “The leak tightness of the containment during a severe reactor accident shall be reliably ensured.”
There is no numerical target for leak tightness though; it shall be suggested and justified by the license applicant/licensee. For ensurance during lifetime of the plant, there are requirements related to commissioning and testing of leak tightness, and in general for inspections, maintenance and so forth.

6) Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)

There shall be at least two personnel accessways. The personnel / accessways shall be airlocks and both doors shall be closed except for during access/exit . The hatches shall be provided with double seals (that can be tested for leak tightness). (YVL B.6)

Also, there shall be designed to withstand same temperature and pressure as the containment. (YVL B.6)

7) **Do you have specific requirements/limitations for other penetrations (e.g. pipe-runs, electrical/I&C cabling)?**

YVL B.6/ 318:"Location, structure, protection and sealing materials of containment penetrations, access locks and hatches, and isolation valves shall ensure their operability and leak tightness during normal operation, anticipated operational occurrences and accidents."

YVL B.6/ 319 "*Containment penetrations shall withstand the loads exerted by piping movements and accidents as intended in requirement* 318."

Also, there shall be designed to withstand same temperature and pressure as the containment. (YVL B.6)

8) **How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?**

Higher level legislations and regulations require that containment integrity/leak tightness shall be maintained in normal accident conditions.

Specific requirements regarding containment isolation function and SSC performing it are presented in YVL Guides hence it means it is possible to provide another solution if the safety level is reached or deviate from the requirement if it can be justified. Specific requirements for valves and other mechanical components are presented in YVL B.6. Principles are that there 2 valves in lines connected to the primary coolant or containment atmosphere located one inside one outside, otherwise one valve is sufficient. Valves shall be passively or automatically closing or locked closed, and diversity is expected.
There are separate requirements that address the electrical supply system and I&C functions.

9) **How are your requirements expressed to address protection from internal and external hazards?**

These are presented for the NPP as a whole and hence cover the containment as well. It is also defined that the containment is purposed to "*protect the plant against natural and human induced external events*".
The binding level of regulations requires: "T*he design of a nuclear facility shall take account of external hazards that may endanger safety. Systems, structures, components and access shall be designed, located and protected so that the impacts of external hazards deemed possible on nuclear facility safety remain minor. The operability of systems, structures and components shall be demonstrated in their design basis external environmental conditions.*

a) *External hazards shall include exceptional weather conditions, seismic events, the effects of accidents that take place in the environment of the facility, and other factors resulting from the environment or human activity. The design shall also consider unlawful and other unauthorised activities compromising nuclear safety and a large commercial aircraft crash*.

There are similar requirements applying to protection from internal hazards.

The Guide YVL B.7 elaborates the requirements for protection from the hazards. The selection of the hazards and their magnitudes for design is to be justified by the licensee/license applicants. More extreme conditions are required to be considered as “design extension conditions DEC” which the plant shall survive without severe core damage.

10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

Regulation on the Safety of a Nuclear Power Plant (STUK Y/1/2018) states that “*in order to ensure containment building integrity:*

a) the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident conditions; pressure, radiation and temperature loads, radiation levels on plant premises, combustible;

b) *gases, impacts of missiles and short-term high energy phenomena resulting from an accident shall be considered in the design of the containment; and*

c) *the possibility of containment leak tightness becoming endangered as a result of reactor pressure vessel fracturing shall be extremely low*.”

Hence the binding level of legislation and regulations is not very specific in this regard, but a core melt accident is assumed hence there is some technology dependency. The specific requirement (point c) however would not pose a problem but other severe accident-related requirements in combination with the current definition of severe accident would pose a problem for interpretation at the least.

The YVL guides set forth some more detailed requirements concerning mitigation of certain phenomena that also assume certain technology and hence significance of certain phenomena is also assumed. However, with respect to YVL Guides other solutions may be suggested if the same level of safety is achieved and even deviations from requirements may be accepted if justifiable.

11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

The Nuclear Energy Act requires following Defence-in-Depth principle in structural design of a NPP; no exact number is set. However, STUK regulation on the Safety of a Nuclear Power Plant STUK Y/1/2018 (binding) evidently assumes a light water reactor and requires a “containment” although there is no formal definition for containment on this level. Certain requirements are presented though, which assume a light water NPP.

The definition of containment is provided in the YVL guide B.6. YVL Guides are currently binding although another solution that provides the same level of safety may be suggested by the licensee/licence applicant. Tasks are defined more on detail in YVL B.6/301:

"A nuclear power plant shall be provided with a leak tight containment system to:

b) *limit the release of radioactive substances during normal operation, anticipated operational occurrences and accidents;*
c) *protect the plant against natural and human induced external events; and*
d) *provide a protective biological shield during normal operation, anticipated operational occurrences and accidents*."

Also, there is a requirement for steel liner in YVL B.6: "A concrete containment shall be lined with leak tight steel cladding." As stated, renewal of legislation, regulations and guides has been started and the role and requirements for containment/confinement will be reconsidered.

12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

No system specific quantitative targets are set forth. Balance of design is assessed by PSA and quality and qualification requirements, failure criteria etc. are to be graded according to the safety class (safety function) and possibly risk significance.

13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/commissioning/in service)?

Commissioning and testing is expected to be performed the same way as for any safety systems and safety related structures (i.e. shall be planned, performed etc before operation). Specific requirements are presented in YVL B.6

YVL B.6/350: "*A containment pressure test shall be performed prior to the commissioning of the plant to demonstrate the structural integrity of the containment. The overpressure used in the pressure test shall be at least 1.15 times the containment design overpressure. The requirements for commissioning of the nuclear power plant are set out in Guide YVL A.5. The requirements for the containment pressure and leak test plans are set out in Guide YVL E.6*"

YVL B.6/351: "*Regular leak tests shall be performed on the containment as well as its penetrations, access locks and hatches to ensure that the leak tightness of the containment remains at an acceptable level throughout the service life of the plant. The leak test shall be performed at a pressure equivalent to the maximum pressure in the postulated accident exerting the highest load on the containment. The leak test shall be performed at intervals that enable reliable monitoring of containment leak tightness*"

Inspections are YVL Guide E.6 *Buildings and structures of a nuclear facility*.

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

See questions 9, 10, 15

15) **How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?**

DEC without core melt are required to cover common cause failures of safety systems, complex sequences and extreme external conditions. The regulations and guides do not in general require specific: they are required to be defined based on the information of the DBA analyses combined to additional information from e.g. fault and failure analyses etc, PSA input and so on. (There is one exception for one, specific BWR containment related scenario.) DECs without core melt are required to be mitigated with either diverse systems (CCF of safety systems) or other means for which the requirements may be less stringent (complex scenarios, extreme external conditions/hazards).

Severe accidents are required to be taken into account in the design so that there are dedicated, independent systems for managing and mitigating them (STUK 1/Y/2018). The systems shall be safety classified (SC3) and single failure tolerant. Best estimate assumptions may be applied but "the more essential the function, the better assurance for its successful accomplishment shall be provided". (YVL B.3) All of the relevant phenomena are required to be covered by the analyses and design.

Early or large releases are required to be practically eliminated.

16) **What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?**

The limiting scenario is not defined in the legislation, regulation or guides. It is expected the licensee/license applicant presents the justification for the scenarios and necessary uncertainty/sensitivity analyses. It is required that severe accidents are considered but the same applies; scenarios are expected to be comprehensively analysed and selection is to be justified.

17) **How do you articulate requirements for managing containment ageing and degradation?**

The binding STUK regulation 1/Y/2018 requires aging management in general for the whole NPP:

The design, construction, operation, condition monitoring and maintenance of a nuclear facility shall provide for the ageing of systems, structures and components important to safety in order to ensure that they meet the design-basis requirements with necessary safety margins throughout the service life and decommissioning of the facility.

Systematic procedures shall be in place for preventing such ageing of systems, structures and components which may deteriorate their availability, and for the early *detection of the need for their repair, modification and replacement. Safety requirements and applicability of new technology shall be periodically assessed in order to ensure that the technology applied is up to date, and the availability of the spare parts and the system support shall be monitored*.

The guide YVL A.8 Ageing management of a nuclear facility elaborates these above requirements but there is no separate, specific set of requirements for containment only.

18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

No; there are no SMRs related licensing processes/ detailed discussions with vendors yet.

19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?

The underground location is not directly prohibited. Current requirements however assume a plant “conventionally” located hence there could be mismatches on the YVL Guide level. Underground location is currently under consideration in the renewal of legislation, regulations and guides.

b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?

The detailed requirements are presented in the YVL guide level hence “other solutions” and deviations could be possible. However, the assessment of the YVL guides has been started and e.g. below ground level location is one aspect to be contemplated.

20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

ANNEX C: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT – NRA (JAPAN)

Please note that the article descriptions given in the following responses are provisional English translations. Also, please refer to Attachment 1, which explains the structure of the relevant regulatory standards in Japan, when reviewing the responses provided below.

Terminology:

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

- **a) “Means” or “Provisions” versus “System(s)”**
- **b) Containment**
- **c) Confinement**
- **d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework)**

Nuclear Regulation Authority Japan (NRAJ) Response:

a) In the regulations, the corresponding Japanese terms for “means,” “provisions,” and “system(s)” are not defined but are used as general terms.

b) For commercial light water reactors (LWRs):
In Article 2, Paragraph 2, Item 36 of the ***NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities*** [1] (hereafter referred to as “***NRA Ordinance on Standards for Installation Permit for LWRs***”), the term “reactor containment vessel” is defined as “vessel provided to prevent leakage of radioactive materials released from machinery or equipment in the vessel of a power reactor facility in the primary cooling system.”

For test and research reactors (including non-LWR):
The definition of “containment” is not provided in the regulatory documents. However, Article 2, Paragraph 2, Item 41 of the ***NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Reactors Used for Testing and Research*** [2] (hereafter referred to as “***NRA Ordinance on Standards for Installation Permit for Test/Research Reactors***”) defines “reactor containment boundary” as “The part of the reactor facility for testing and research for gas-cooled reactors or sodium-cooled fast reactors that serves as a pressure barrier and a barrier to the release of radioactive materials in the event of an anticipated event in the reactor containment vessel.”

In addition, Article 52, Paragraph 1 of the ***Regulatory Guide of the NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Reactors Used***

for Testing and Research [3] (hereafter referred to as "***Guide for NRA Ordinance on Standards for Installation Permit for Test/Research Reactors***") states that "Reactor containment facility" means a facility to maintain negative pressure during normal operation and to prevent the release of radioactive materials outside the facility in the event of an accident. For gas-cooled reactors, the reactor containment facility consists of the reactor building (including the service area), the reactor containment vessel, and its appurtenances.

c) In IAEA Safety Glossary: 2018 Edition states that "confinement is typically used to refer to the safety function of preventing the 'escape' of radioactive material, whereas containment refers to the means for achieving that function." and that "Confinement in nuclear safety is the safety function that is performed by the containment." In light of these, we interpret "confinement" as a function of confinement of radioactive materials and respond to this question as follows.

For commercial LWRs:
In the ***Review Guideline for Classification of Importance of Safety Functions of Light Water Reactor Facilities for Power Generation*** [4], "Confinement function of radioactive materials" is listed as one of the functions of MS-1 systems, structures, and components (SCCs), which are the SSCs classified as safety importance class 1 with the function of mitigating the abnormal effects.

For test and research reactors (water-cooled reactors):
In the ***Review Guideline for Safety Design of Water-Cooled Reactor Facilities Used for Testing and Research*** [5], "confinement function of radioactive materials" is listed as one of the functions of SSCs with MS-2 and MS-3, which are the SSCs that have the function of mitigating the abnormal effects and are classified as safety importance class 2 and 3, respectively.

d) Interpretations or guidance on functional containment are not provided for commercial LWRs and test and research reactors.

Specific Requirements:

1) **How requirements and guidance are articulated:**
    a. **Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?**
    b. **What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?**

NRAJ Response:

a) Requirements for reactor containment structure are specified in the following Articles of the NRA's ordinances. They identify key safety objectives or performance criteria to be met in ensuring acceptable containment functions.

For commercial LWRs:
NRA Ordinance on Standards for Installation Permit for LWRs [1], Article 32 (reactor containment structure), Paragraphs 1 to 8
Examples are as follows.

Article 32, Paragraph 1 states:
"A reactor containment vessel shall be able to withstand expected maximum pressure, maximum temperature, and appropriate seismic load adequately and preserve leakage rate within specified limit through a combined measure with the properly operated isolation functions so that the public will not receive radiation effects by leakage of radioactive materials in the event of damages or failures of equipment that belongs to the primary cooling system."

Article 32, Paragraph 2 states:
"Equipment constituting reactor containment boundary shall have adequate fracture toughness to avoid instantaneous destruction during normal operation and in the event of an anticipated operational occurrence and a design basis accident."

NRA Ordinance Prescribing Technical Standards for Commercial Power Reactors [6] (hereafter referred to as "***NRA Ordinance on Technical Standards for LWRs***"), Article 44 (reactor containment structure) prescribes detailed design requirements.

For test and research reactors:
NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2], Article 27 (reactor containment structure for reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors), Article 52 (reactor containment structure of gas-cooled fast reactor) and Article 60 (reactor containment structure of sodium-cooled fast reactor)
Examples are as follows.

Paragraph 1 of Article 52 and Article 60 states:

"Reactor facilities for test and research, etc. shall be provided with reactor containment facilities according to the following:

Item 1: The inside of the unit shall be capable of maintaining a negative pressure during normal operation and shall not exceed the specified leakage rate.

Item 2: In the event of a design basis accident, radioactive materials emitted from the reactor containment facility shall be reduced to prevent radiation hazards to the public."

NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research [7] (hereafter referred to as "***NRA Ordinance on Technical Standards for Test/Research Reactors***")
The following Articles prescribe detailed design requirements.

(Reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors)
Article 37 (Reactor containment structure)

(Research and development stage reactor)
Article 50 (Reactor containment structure)

(Gas-cooled reactor)
Article 56 (Reactor containment structure)

(Sodium-cooled fast reactor)
Article 65 (Reactor containment structure)

b) There are no provisions for alternative means of confinement/containment functions for commercial LWRs and test and research reactors.

2) Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?

NRAJ Response:

All the requirements in the Articles shown in 1) a are provided with some condition (pressure, temperature, seismic force, design basis accident, etc.) or situation under which the SSCs should function. Examples are:

<u>For commercial LWRs:</u>

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 32 (reactor containment structure), Paragraph 6 states:

"Power reactor facility shall have a system (limited to those categorized as a safety system) to remove heat generated within the reactor containment vessel to prevent the integrity of the reactor containment vessel from being impaired due to increases in pressure and temperature in the reactor containment vessel resulting from damages or failures of equipment in the primary cooling system."

Article 32, Paragraph 7 states:

"Power reactor facility shall have a system to clean up the atmosphere within the reactor containment vessel (limited to those categorized as a safety system) to reduce the concentration of radioactive materials in the case that might cause radiological consequences to the public due to a leakage of gaseous radioactive materials from the reactor containment vessel, resulting from possible damages or failures of equipment in the primary cooling system."

Article 32, Paragraph 8 states:

"Power reactor facility shall have a combustible gas concentration control system (limited to such system categorized as a safety system) to control the concentration of hydrogen and oxygen in the case that may jeopardize the integrity of the reactor containment vessel due to hydrogen and oxygen generated resulting from possible damage or failure of equipment in the primary cooling system."

For test and research reactors:

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

Article 52 (reactor containment structure of gas-cooled fast reactor), Paragraph 4 states:

"Reactor facilities for test and research shall have a system to reduce the concentration of flammable gases and oxygen if there is a risk that the integrity of the reactor containment vessel may be affected by combustible gases and oxygen generated in the event of a design basis accident that results in damage to the primary cooling system piping or other pressure reductions in the primary cooling system."

Article 52, Paragraph 5 states:

"Reactor facilities for test and research shall have a system to reduce the concentration of radioactive materials in the reactor containment facility if there is a risk of radiation hazard to the public due to a design basis accident or other leakage of gaseous radioactive material from the reactor containment vessel."

3) Safety Classification:

- **a. Under what conditions are containment provisions classified as safety systems?**
- **b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?**

NRAJ Response:

a) For commercial LWRs, in the ***Review Guideline for Classification of Importance of Safety Functions of Light Water Reactor Facilities for Power Generation*** [4], "confinement function of radioactive materials" is listed as one of the functions of MS-1 SCCs, which are the SSCs classified as safety importance class 1 with the function of mitigating the abnormal effects. Therefore, since a reactor containment vessel has a "confinement function of radioactive materials," it is classified as MS-1 and a safety system.

For test and research reactors, the classification of a reactor containment vessel does not appear in regulatory documents.

b) There are no provisions for the systems that support containment and confinement to be classified at a lower safety level depending on the conditions, both for commercial LWRs and test and research reactors.

4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

NRAJ Response:

For commercial LWRs:
Articles of the NRA ordinances related to the development and demonstration of the containment design basis for internal and external events are as follows. There are requirements on containment systems at DiD levels 1-3 and at conditions exceeding design basis accidents. For Articles 12, 37, and 43 in the ***NRA Ordinance on Standards for Installation Permit for LWRs*** [1], the texts of the Articles are also shown below as examples.

NRA Ordinance on Standards for Installation Permit for LWRs [1]
Article 3 (Ground for installing the design basis response systems)
Article 4 (Prevention of damage due to earthquakes)
Article 5 (Prevention of damage due to tsunami)
Article 6 (Prevention of damage due to external hazards)
Article 8 (Prevention of damage due to fire)
Article 9 (Prevention of damage due to flooding)
Article 12 (Safety systems)
Article 32 (Reactor containment structure), Paragraph 6 (Heat removal), 7 (Cleaning up atmosphere), and 8 (Combustible gas concentration control system)
Article 37 (Prevention of severe accidents and its escalation)
Article 38 (Ground for installing SA response facilities)
Article 39 (Prevention of damage due to earthquakes)
Article 40 (Prevention of damage due to tsunamis)
Article 41 (Prevention of damage due to fire)
Article 42 (Specified SA response facilities)
Article 43 (SA response Equipment)
Article 44 (Reactor containment structure), Paragraph 1, Item 3 (Combustible gas concentration control system), 4 (Cleaning up atmosphere), 5(Heat removal)
Article 49 (Equipment to cool the inside of reactor containment vessel)
Article 50 (Equipment to prevent failures of containment vessel due to over-pressurization)
Article 51 (Equipment to cool molten cores at the bottom of reactor containment vessel)
Article 52 (Equipment to prevent failures of containment vessel due to a hydrogen explosion)

Article 12 (Safety systems):

Paragraph 3 states, "Safety systems shall be such as to achieve their functions under any environmental conditions postulated in design basis accidents and during the period when the situation escalates to a design basis accident."
Paragraph 5 states, "Safety systems shall be such that their safety functions is not impaired by missiles resulting from damage of any of the steam turbines, pumps, or other components or piping."

Article 37 (Prevention of severe accidents and its escalation):
Paragraph 2 states, "Power reactor facility shall be such as to be provided with measures necessary to prevent failures of the containment vessel and abnormal release of radioactive materials to the outside of the station in the event of a severe accident."

Article 43 (SA response equipment)
Paragraph 1, Item 1 states "SA response equipment shall be able to effectively achieve its functions necessary to cope with severe accidents under environmental conditions such as temperature, radiation, loads in the event of a postulated severe accident."

NRA Ordinance on Technical Standards for LWRs [6]
Articles 4 to 7, 11, 12, 14, 49 to 54, and 64 to 67 prescribe detailed design requirements.

<u>For test and research reactors:</u>
Articles of the NRA ordinances related to the development and demonstration of the containment design basis for internal and external events are as follows.

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

(Reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors)

Article 3 (Ground for test and research reactor facility)

Article 4 (Prevention of damage due to earthquakes)

Article 5 (Prevention of damage due to tsunami)

Article 6 (Prevention of damage due to external hazards)

Article 8 (Prevention of damage due to fire)

Article 9 (Prevention of damage due to flooding)

Article 12 (Safety systems)

Article 27 (Reactor containment structure)

(Water-cooled research reactors)

Article 40 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Gas-cooled reactors)

Article 52 (Reactor containment structure)

Article 53 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Sodium-cooled fast reactors)

Article 60 (Reactor containment structure)

Article 61 (Application, Mutatis Mutandis (Prevention of the escalation of accidents involving the release large amounts of radioactive materials))

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Articles 6 to 8, 19, 21, 37, 39, 50, 56, 58, and 65 prescribe detailed design requirements.

5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?

NRAJ Response:

The following Articles provide requirements for preventing or mitigating the release of radioactive materials.

For commercial LWRs:
NRA Ordinance on Standards for Installation Permit for LWRs [1]
Article 12 (Safety systems), Paragraph 4
"Safety facilities shall be such that they can be tested or inspected according to the importance of their safety functions during power operation or shut-down conditions to verify their integrity and capability."

Article 32 (Reactor containment structure), Paragraph 1
"A reactor containment vessel shall be able ... to preserve leakage rate within specified limit through a combined measure with the properly operated isolation functions..."

Article 50 (Equipment to prevent failures of containment vessel due to over-pressurization),
Paragraph 1
"In order to prevent damage to the reactor containment vessel due to overpressurization in the event of significant core damage, power reactor facilities shall be equipped with

the necessary facilities to reduce the pressure and temperature in the containment vessel while maintaining the containment vessel boundary."

Paragraph 2
"Power reactor facilities (limited to those that are likely to cause damage due to overpressurization of the reactor containment vessel within a short period in the event of significant damage to the reactor core due to its structure) shall be equipped with the necessary facilities to release the pressure in the reactor containment vessel to the atmosphere."

NRA Ordinance on Technical Standards for LWRs [6]
Article 21 (Pressure resistance test, etc.)
Paragraph 1
"... and reactor containment vessels shall be able to withstand pressure tests at the pressures specified below and shall not leak significantly. However, if the test is conducted by atmospheric pressure and it is confirmed that the tested SSCs can withstand the specified pressure, the pressure may be reduced to the maximum operating pressure (0.9 times the maximum operating pressure for a reactor containment vessel) to confirm that no significant leakage occurs. ..."
Paragraph 3
"The reactor containment shall be free of significant leakage when the airtight test is conducted at an atmospheric pressure equal to 0.9 times the maximum operating pressure."

Article 44 (Reactor containment structure), Paragraph 1, Item 1(c)
"The penetration points and the entrance/exit of the reactor containment vessel shall be capable of leak testing according to the expected leakage volume and other environmental conditions that affect leak testing."

Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Commercial Power Reactors [8] (hereafter referred to as "***Guide for NRA Ordinance on Technical Standards for LWRs***")
Article 21, Paragraph 3 and Article 44, Paragraph 2 refer to Japan Electric Association Code (JEAC) 4203 "Code for Leak Rate Tests of Nuclear Reactor Containment Vessels" with conditions for the implementation of Article 21, Paragraph 3 and Article 44, Paragraph 1, Item 1(c) of the ***NRA Ordinance on Technical Standards for LWRs*** [6].

<u>For test and research reactors:</u>
NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

Article 12 (Safety systems), Paragraph 4
"Safety facilities shall be such that they can be tested or inspected according to the importance of their safety functions during the operation or shutdown of the reactors for testing and research to verify their integrity and capability."

Paragraph 1 of Article 27 (Reactor containment structure for reactor facilities except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast

reactors), 52 (Reactor containment structure for gas-cooled reactor), and 60 (Reactor containment structure for sodium cooled fast reactor)
Item 1
"The inside of the reactor containment facility shall be able to be maintained in a negative pressure condition, and the leakage rate shall not exceed the specified leakage rate during normal operation."
Item 2
"In order to prevent radiation hazards to the public in the event of a design basis accident, radioactive materials emitted from the reactor containment facility shall be reduced."

Paragraph 2 of Article 52 (gas-cooled reactor), and 60 (sodium-cooled fast reactor)
"The equipment comprising the reactor containment boundary shall have sufficient fracture toughness to prevent instantaneous failure during normal operation, anticipated operational occurrence, and design basis accidents. It shall also not pose a leak exceeding the prescribed leakage rate through a combined measure with the properly operated isolation functions."

NRA Ordinance on Technical Standards for Test/Research Reactors [7]
Article 12, Paragraph 2
"Equipment belonging to a reactor facility for testing and research shall be able to withstand pressure resistance tests or leakage tests, as appropriate to the importance of its safety function, and shall be free from significant leakage."

Paragraph 2 of Article 56 (gas-cooled reactor) and 65 (sodium-cooled fast reactor)
"The reactor containment vessel belonging to the reactor facility for testing and research shall be capable of periodic leakage rate testing."

Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research [9]
For the implementation of Article 12, Paragraph 2 of the ***NRA Ordinance on Technical Standards for Test/Research Reactors*** [7], Article 12, paragraph 8 refers to the attachment for appropriate pressure resistance test, in which it is stated that the SSCs shall be able to withstand pressure test and leak-free at the test pressure indicated for the equipment category.

6) Do you have specific requirements/limitations for large penetrations (e.g., airlocks/hatches/ other accessways?)

NRAJ Response:

There are no requirements specific to large penetrations of the reactor containment vessels. However, the general requirements that also apply to penetrations are as follows.

<u>For commercial LWRs:</u>
NRA Ordinance on Standards for Installation Permit for LWRs [1]
Article 32 (Reactor containment structure), Paragraphs 1 and 2 (See 1) a)

NRA Ordinance on Technical Standards for LWRs [6]
Article 44 (Reactor containment structure), Paragraph 1, Item 1(b)
"If there is an opening in the reactor containment vessel, it shall be airtight."

For test and research reactors:
NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]
Item 1 and 2 of Paragraph 1 in Article 27 (Reactor containment structure for reactor facilities except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors), 52 (Reactor containment structure for gas-cooled reactor), and 60 (Reactor containment structure for sodium cooled fast reactor) (see 5))

Paragraph 2 of Article 52 (Reactor containment structure for gas-cooled reactor), and 60 (Reactor containment structure for sodium cooled fast reactor) (see 5))

NRA Ordinance on Technical Standards for Test/Research Reactors [7]
Item 2 of Paragraph 1 in Article 56 (Reactor containment structure for gas-cooled reactor) and 65 (Reactor containment structure for sodium cooled fast reactor)

"Airtight doors shall be provided at the openings of the reactor containment facility."

7) Do you have specific requirements/limitations for other penetrations (e.g., pipe-runs, electrical/I&C cabling)?

NRAJ Response:

The requirements related to piping penetrations other than the general requirements related to penetrations shown in 6) are as follows.

For commercial LWRs:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 32 (Reactor containment structure)

Paragraph 3

"Piping that penetrates the reactor containment vessel shall be provided with isolation valves (limited to those categorized to safety system; the same applies in Paragraphs 4 and 5) except the piping relating to instrumentation devices or control rod drive devices whose leakage is limited within an allowable level."

Paragraph 4

"Isolation valves to be provided in principal piping (excluding the piping in the systems necessary to cope with accidents) shall have functions to be closed automatically and completely in the case when isolation functions must be achieved in the event of a design basis accident."

Paragraph 5

"Power reactor facility shall be provided with isolation valves, as prescribed as follows: ..."

NRA Ordinance on Technical Standards for LWRs [6]

Article 44 (Reactor containment structure), Paragraph 1, Item 2 presents detailed requirement for design of isolation valves.

For test and research reactors:

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

Paragraph 3 in Article 52 (Reactor containment structure for gas-cooled reactor) and 60 (Reactor containment structure for sodium cooled fast reactor)

"Isolation valves shall be provided on piping that penetrates the reactor containment vessel...."

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Paragraphs 3 to 5 of Article 56 (Reactor containment structure for gas-cooled reactor) and 65 (Reactor containment structure for sodium cooled fast reactor) presents detailed requirement for design of isolation valves.

8) How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?

NRAJ Response:

The requirements related to the isolation function of the reactor containment vessels and the isolation valves of penetrating piping are as follows.

For commercial LWRs:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 32 (Reactor containment structure), Paragraph 1 states:

"A reactor containment vessel shall be able to withstand expected maximum pressure, maximum temperature, and appropriate seismic load adequately and preserve leakage rate within specified limit through a combined measure with the properly operated isolation functions so that the public will not receive radiation effects by leakage of radioactive materials in the event of damages or failures of equipment that belongs to the primary cooling system."

Article 32, Paragraphs 3 to 5 provide requirements for isolation valves. (See 7)).

NRA Ordinance on Technical Standards for LWRs [6]

Article 44 (Reactor containment structure), Paragraph 1, Item 2 (see 7))

<u>For test and research reactors:</u>

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

Paragraph 2 of Article 52 (Reactor containment structure for gas-cooled reactor), and 60 (Reactor containment structure for sodium cooled fast reactor) (see 5))

Paragraph 3 in Article 52 (Reactor containment structure for gas-cooled reactor) and 60 (Reactor containment structure for sodium cooled fast reactor) (requirements for isolation valves, see 7))

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Paragraphs 3 to 5 of Article 56 (Reactor containment structure for gas-cooled reactor) and 65 (Reactor containment structure for sodium cooled fast reactor) (see 7))

9) How are your requirements expressed to address protection from internal and external hazards?

NRAJ Response:

The requirements to address protection from internal and external hazards are prescribed in the Articles indicated in 4).

10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

NRAJ Response:

The requirements related to load management in a reactor containment vessel are as follows. These are part of the requirements indicated in 4) and those related to instrumentation equipment.

For commercial LWRs:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 8 (Prevention of damage due to fire)

Article 9 (Prevention of damage due to flooding)

Article 12 (Safety systems)

Article 23 (Instrumentation and control systems)

Article 32 (Reactor containment structure), Paragraph 6 (Heat removal), 7 (Cleaning up atmosphere), and 8 (Combustible gas concentration control system)

Article 37 (Prevention of severe accidents and its escalation)

Article 41 (Prevention of damage due to fire)

Article 43 (SA response Equipment)

Article 49 (Equipment to cool the inside of reactor containment vessel)

Article 50 (Equipment to prevent failures of containment vessel due to over-pressurization)

Article 51 (Equipment to cool molten cores at the bottom of reactor containment vessel)

Article 52 (Equipment to prevent failures of containment vessel due to a hydrogen explosion)

Article 58 (Instrumentation equipment)

NRA Ordinance on Technical Standards for LWRs [6]

Articles 11, 12, 14, 34. 44. 52. 54, 64 to 67, and 73 prescribe detailed design requirements.

For test and research reactors:

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

(Reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors)

Article 8 (Prevention of damage due to fire)

Article 9 (Prevention of damage due to flooding)

Article 12 (Safety systems)

Article 17 (Instrumentation and control systems)

(Water-cooled research reactors)

Article 36 (Instrumentation and control systems)

Article 40 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Gas-cooled reactors)

Article 48 (Instrumentation and control systems)

Article 52 (Reactor containment structure), Paragraphs 4 and 5

Article 53 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Sodium-cooled fast reactors)

Article 58 (Instrumentation and control systems)

Article 60 (Reactor containment structure), Paragraph 4

Article 61 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Articles 19, 21, 30, 37, 39, 48, 50, 55, 56, 58, 63, and 65 prescribe detailed design requirements.

11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

NRAJ Response:

The following Articles prescribe the requirements on fuel cladding, reactor coolant pressure boundary, or reactor containment facilities as barriers that confine radioactive materials.

<u>For commercial LWRs:</u>
NRA Ordinance on Standards for Installation Permit for LWRs [1]
Article 13 (Prevention of escalation of anticipated operational occurrence and design basis accidents)
Article 15 (Reactor core and its associated items)
Article 17 (Reactor coolant pressure boundary)
Article 32 (Reactor containment structure)

NRA Ordinance on Technical Standards for LWRs [6]
Article 13 (Prevention of escalation of anticipated operational occurrence and design basis accidents)
Article 15 (Reactor core and its associated items)
Article 17 (Reactor coolant pressure boundary)
Article 32 (Reactor containment structure)
(These Articles specify detailed requirement for equipment as barriers that confine radioactive materials.)

<u>For test and research reactors:</u>
NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

(Reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors)
Article 15 (Reactor core and its associated items), Paragraphs 3 and 5
Article 27 (Reactor containment structure)

(Water-cooled reactor)
Article 32 (Reactor core and its associated items), Paragraphs 2 and 4
Article 33 (Reactor coolant pressure boundary)

(Gas-cooled reactor)
Article 45 (Reactor coolant pressure boundary)
Article 52 (Reactor containment structure)

(Sodium cooled fast reactor)
Article 55 (Reactor coolant pressure boundary)
Article 60 (Reactor containment structure)

NRA Ordinance on Technical Standards for Test/Research Reactors [7]
(Test and research reactor)
Article 22 (Reactor core and its associated items)

(Reactors except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors)
Article 37 (Reactor containment structure)

(Research and development stage reactor)
Article 44 (Reactor coolant pressure boundary)
Article 50 (Reactor containment structure)

(Gas-cooled reactor)
Article 54 (Reactor coolant pressure boundary)
Article 56 (Reactor containment structure)

(Sodium-cooled fast reactor)
Article 61 (Reactor core and its associated items), Paragraphs 1, 2, and 4
Article 62 (Reactor coolant pressure boundary)
Article 65 (Reactor containment structure)
(These Articles specify detailed requirement for equipment as barriers that confine radioactive materials.)

12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

NRAJ Response:

The requirements on reliability of the containment system are as follows.

<u>For commercial LWRs:</u>

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 12, Paragraph 1 states, "Safety facilities shall ensure safety functions according to the importance of their safety functions."

Regulatory Guide of the NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities [10] (hereafter referred to as "***Guide for NRA Ordinance on Standards for Installation Permit for LWRs***")

Article 12, Paragraph 1 refers to the ***Review Guideline for Classification of Importance of Safety Functions of Light Water Reactor Facilities for Power Generation*** [4], which states that "the highest reasonably achievable level of reliability shall be ensured and maintained" for safety-importance class 1 SSCs, including reactor containment vessels.

Specific requirements related to reliability of reactor containment are provided in Articles shown in 1)a.

13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g., construction/commissioning/in service)?

NRAJ Response:

The requirements for testing and inspecting the containment function of a reactor containment vessel are as follows.

For commercial LWRs:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 12, Paragraph 4 states:

"Safety facilities shall be such that they can be tested or inspected according to the importance of their safety functions during power operation or shut-down conditions to verify their integrity and capability."

Guide for NRA Ordinance on Standards for Installation Permit for LWRs [10]

Article 12, Paragraph 9 states:

"A reactor containment vessel shall be designed to allow measurement of the leakage rate of the entire reactor containment vessel at regular intervals and at a predetermined pressure and leakage tests for important parts of penetrations, including wires, piping, and entrances and exits."

NRA Ordinance on Technical Standards for LWRs [6]

Article 21 (Pressure resistance test, etc.), Paragraph 1 states:

"... and reactor containment vessels shall be able to withstand pressure tests at the pressures specified below and shall not leak significantly. However, if the test is conducted by atmospheric pressure and it is confirmed that the tested SSCs can withstand the specified pressure, the pressure may be reduced to the maximum operating pressure (0.9 times the maximum operating pressure for a reactor containment vessel) to confirm that no significant leakage occurs. ..."

Article 21, Paragraph 3 states:

"The reactor containment shall be free of significant leakage when airtight test is conducted at an atmospheric pressure equal to 0.9 times the maximum operating pressure."

Guide for NRA Ordinance on Technical Standards for LWRs [8]

Article 21, Paragraph 1 refers to "Codes for Nuclear Power Generation Facilities – Rules on Design and Construction for Nuclear Power Plants -" of the Japan Society of Mechanical Engineers for the implementation of Article 21, Paragraph 1 of the ***NRA Ordinance on Technical Standards for LWRs*** [6].

Article 21, Paragraph 3 refers to Japan Electric Association Code (JEAC) 4203 "Code for Leak Rate Tests of Nuclear Reactor Containment Vessels" with conditions for the implementation of Article 21, Paragraph 3 of the ***NRA Ordinance on Technical Standards for LWRs*** [6].

For test and research reactors:

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

Article 12 (Safety systems), Paragraph 4 states:

"Safety facilities shall be such that they can be tested or inspected according to the importance of their safety functions during the operation or shutdown of the reactors for testing and research to verify their integrity and capability."

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Article 12, Paragraph 2 states:

"Equipment belonging to a reactor facility for testing and research shall be able to withstand pressure resistance tests or leakage tests, as appropriate to the importance of its safety function, and shall be free from significant leakage."

Paragraph 2 of Article 56 (Reactor containment structure for gas-cooled reactor) and 65 (Reactor containment structure for sodium cooled fast reactor) states:

"The reactor containment vessel belonging to the reactor facility for testing and research shall be capable of periodic leakage rate testing."

Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research [9]

For the implementation of Article 12, Paragraph 2 of the ***NRA Ordinance on Technical Standards for Test/Research Reactors*** [7], Article 12, Paragraph 8 refers to the attachment for appropriate pressure resistance test, in which it is stated that the SSCs shall be able to withstand pressure test and leak-free at the test pressure indicated for the equipment category.

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

NRAJ Response:

For commercial LWRs:

The requirements related to measures to prevent reactor containment failure in an accident exceeding the design basis accident are as follows.

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 37 (Prevention of severe accidents and its escalation)

Article 38 (Ground for installing SA response facilities)

Article 39 (Prevention of damage due to earthquakes)

Article 40 (Prevention of damage due to tsunamis)

Article 41 (Prevention of damage due to fire)

Article 42 (Specified SA response facilities)

Article 43 (SA response Equipment)

Article 44 (Reactor containment structure), Paragraph 1, Item 3 (Combustible gas concentration control system), 4 (Cleaning up atmosphere), 5 (Heat removal)

Article 49 (Equipment to cool the inside of reactor containment vessel)

Article 50 (Equipment to prevent failures of containment vessel due to over-pressurization)

Article 51 (Equipment to cool molten cores at the bottom of reactor containment vessel)

Article 52 (Equipment to prevent failures of containment vessel due to a hydrogen explosion)

NRA Ordinance on Technical Standards for LWRs [6]

Articles 53, 54, and 64 to 67 prescribe detailed design requirements.

Examination Criteria for the Technical Capability of Installers of Commercial Power Reactors to Implement the Necessary Measures to Prevent the Occurrence and Escalation of Severe Accidents for Commercial Power Reactors [11]

Chapter 2 requires the development of a plan for measures to mitigate damage to the reactor containment vessel in the event of large-scale damage to power generation reactor facilities (loss of large area) due to a large-scale natural disaster or intentional crash of a large aircraft or other forms of terrorism.

For test and research reactors:

The requirements related to measures in an accident exceeding the design basis accident are provided in the following Articles.

NRA Ordinance on Standards for Installation Permit for Test/Research Reactors [2]

(Water-cooled reactors)

Article 40 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Gas-cooled reactors)

Article 53 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

(Sodium-cooled fast reactors)

Article 61 (Prevention of the escalation of accidents involving the release large amounts of radioactive materials)

NRA Ordinance on Technical Standards for Test/Research Reactors [7]

Articles 39 and 58 prescribe detailed design requirements.

15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

NRAJ Response:

The resiliency of the design provision beyond DBA is prescribed in the requirements listed in 14).

16) What is the approach for defining the “limiting” accident scenarios used in the containment design (e.g., for large LWRs this may be main steamline break/LOCA)?

NRAJ Response:

<u>For commercial LWRs:</u>

The requirements for the selection of events in the design of engineering safety facilities, including reactor containment vessels, and in evaluations for the effectiveness of severe accident countermeasures are as follows.

For Design Base:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 13 specifies requirements for preventing an escalation of anticipated operational occurrence and design basis accidents for design basis response facilities.

Article 13, Paragraph 1, Item 2 includes the requirement for a containment vessel in the event of a design basis accident: “pressure and temperature of reactor containment boundary shall not exceed the maximum operating pressure and the maximum operating temperature, respectively.”

Guide for NRA Ordinance on Standards for Installation Permit for LWRs [10]

Article 13 refers to the ***Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities*** [12], which indicates accidents to be evaluated, including LOCA and MSLB.

For severe accident:

NRA Ordinance on Standards for Installation Permit for LWRs [1]

Article 37 (Prevention of severe accidents and its escalation), Paragraph 2 states “In the event of a severe accident, the reactor facility for power generation shall have the necessary measures in place to prevent failure of the reactor containment vessel and the release of abnormal levels of radioactive materials outside the plant.”

Guide for NRA Ordinance on Standards for Installation Permit for LWRs [10]

Article 37 provides accident sequence groups to be assumed concerning the prevention of significant core damage and containment failure modes to be assumed concerning the prevention of containment failure.

For test and research reactors:

There is no defined approach for defining accident scenarios to be used in containment design.

17) How do you articulate requirements for managing containment ageing and degradation?

NRAJ Response:

Requirements related to aging management of containment vessel are as follows.

For commercial LWRs:

NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors [13]

Articles 56, 82, 92, and 113 specify general requirements that are not limited to the containment vessel.

Article 56 (Implementation of a Licensee's Periodic Inspection)

Article 82 (Technical Evaluation on Aging of the Nuclear Power Reactor Facilities)

Article 92 (Operational Safety Program)

Article 113 (Application for Authorization of Extension of Operational Period of a Commercial Nuclear Reactor)

Guide for Application for Approval of Extension of Operational Period for Commercial Power Reactors [14]

Chapter 3 specifies the contents of special inspections of reactor containment vessels, including visual inspection of the inner containment steel plates and confirmation of strength by testing using concrete core samples taken from the containment vessel.

For test and research reactors:

NRA Ordinance Concerning the Installation and Operation of Reactors Used for Testing and Research [15]

Articles 3-9, 9-2, and 15 specify general requirements that are not limited to the containment vessel.

Article 3-9 (Implementation of a Licensee's Periodic Inspection)

Article 9-2 (Technical Evaluation on Aging of the Reactors Used for Testing and Research)

Article 15 (Operational Safety Program)

18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

NRAJ Response:

No.

19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

   a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g., material types, siting restrictions etc)?
   b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g., ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?

NRAJ Response:

No.

20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

NRAJ Response:

There are no other regulatory requirements for confinement of radioactive materials.

Attachment 1: Structures of regulatory requirements for facilities of nuclear reactors in Japan

(1) Commercial power reactors
Regulatory requirements for the basic design of facilities for commercial power reactors are set forth in the ***NRA Ordinance on Standards for Installation Permit for LWRs*** [1], which was established under Article 43-3-6, Paragraph 1, Item 4 of the ***Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors*** [16].
The ordinance provides requirements for facilities to achieve levels 1 to 3 of defence-in depth-in Articles 3 through 36 and requirements for facilities to prevent core damage and containment failure in the event of an accident that exceeds the design basis accident in Articles 37 through 62. The interpretations of the ordinance are provided in ***Guide for NRA Ordinance on Standards for Installation Permit for LWRs*** [10].

The requirements for the detailed design of equipment for commercial power reactors are specified by the ***NRA Ordinance on Technical Standards for LWRs*** [6], which was established under Article 43-3-14 of the *Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors* [16].
The ordinance provide requirements for facilities to achieve levels 1 to 3 of defence-in-depth in Articles 4 through 48 and requirements for facilities to prevent core damage and containment failure in the event of an accident that exceeds the design basis accident in Articles 49 through 78. The interpretations of the ordinance are provided in the ***Guide for NRA Ordinance on Technical Standards for LWRs*** [8].

Regulatory requirements for the maintenance and management of commercial power reactors, including aging management for long-term operation, are required by the ***NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors*** [13], which was established under Article 43-3-22, Paragraph 1 and Article 43-3-24 of the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors [16].

(2) Test and research reactors
Regulatory requirements for the basic design of facilities for test and research reactors are set forth in the ***NRA Ordinance on Standards for Installation Permit for Test/Research Reactors*** [2], which was established under Article, Paragraph 1, Item 3 of the ***Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors*** [16].
The ordinance lists requirements for test and research reactor facilities (except for water-cooled research reactors, gas-cooled reactors, and sodium-cooled fast reactors) in Articles 3 through 30, those for water-cooled reactors in Articles 31 through 41, those for gas-cooled reactors in Articles 42 through 54, and those for sodium-cooled fast reactors in Articles 55 through 61. The interpretations of the ordinance are provided in the ***Guide for NRA Ordinance on Standards for Installation Permit for Test/Research Reactors*** [3].

The requirements for the detailed design of equipment for test and research reactors are specified by the ***NRA Ordinance on Technical Standards for Test/Research Reactors*** [7], which was established under Article 28-2 of the ***Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors*** [16].
The ordinance lists requirements for facilities related to test and research reactors in Articles 5 through 42, for facilities related to research and development stage reactors in Articles 43

through 52, for facilities related to gas-cooled reactors in Articles 53 through 59, and for facilities related to sodium-cooled fast reactors in Articles 60 through 70. The interpretations of the ordinance are provided in the ***Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research*** [9].

Regulatory requirements for the maintenance and management of commercial power reactors, including aging management for long-term operation, are required by the ***NRA Ordinance Concerning the Installation and Operation of Reactors Used for Testing and Research*** [15], which was established under Article 35, Paragraph 1 and Article 37 of the ***Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors*** [16].

References

[1] Nuclear Regulation Authority, “NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities,” 2013. https://elaws.e-gov.go.jp/document?lawid=425M60080000005 (Accessed Sep. 13, 2022.)

[2] Nuclear Regulation Authority, “NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Reactors Used for Testing and Research,” 2013. https://elaws.e-gov.go.jp/document?lawid=425M60080000021 (Accessed Sep. 13, 2022.)

[3] Nuclear Regulation Authority, “Regulatory Guide of the NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Reactors Used for Testing and Research,” November 27, 2013. https://www.nra.go.jp/data/000172364.pdf (Accessed Sep. 13, 2022.)

[4] Nuclear Safety Commission, “Review Guideline for Classification of Importance of Safety Functions of Light Water Reactor Facilities for Power Generation,” August 30, 1990. https://warp.da.ndl.go.jp/info:ndljp/pid/9483636/www.nsr.go.jp/archive/nsc/shinsashishin/pdf/1/si003.pdf (Accessed Sep. 13, 2022.)

[5] Nuclear Safety Commission, “Review Guideline for Safety Design of Water-Cooled Reactor Facilities Used for Testing and Research,” July 18, 1991. https://warp.da.ndl.go.jp/info:ndljp/pid/9483636/www.nsr.go.jp/archive/nsc/shinsashishin/pdf/1/si018.pdf (Accessed Sep. 13, 2022.)

[6] Nuclear Regulation Authority, “NRA Ordinance Prescribing Technical Standards for Commercial Power Reactors,” 2013. https://elaws.e-gov.go.jp/document?lawid=425M60080000006 (Accessed Sep. 13, 2022.)

[7] Nuclear Regulation Authority, “NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research,” 2020. https://elaws.e-gov.go.jp/document?lawid=502M60080000007 (Accessed Sep. 13, 2022.)

[8] Nuclear Regulation Authority, “Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Commercial Power Reactors,” 2013. https://www.nra.go.jp/data/000382457.pdf (Accessed Sep. 13, 2022.)

[9] Nuclear Regulation Authority, “Regulatory Guide of the NRA Ordinance Prescribing Technical Standards for Reactors Used for Testing and Research,” 2020. https://www.nra.go.jp/data/000308612.pdf (Accessed Sep. 13, 2022.)

[10] Nuclear Regulation Authority, “Regulatory Guide of the NRA Ordinance Prescribing Standards for the Location, Structure, and Equipment of Commercial Power Reactors and their Auxiliary Facilities,” 2013. https://www.nra.go.jp/data/000382455.pdf (Accessed Sep. 13, 2022.)

[11] Nuclear Regulation Authority, “Examination Criteria for the Technical Capability of Installers of Commercial Power Reactors to Implement the Necessary Measures to Prevent the

Occurrence and Escalation of Severe Accidents for Commercial Power Reactors," June 19, 2013.
https://www.nra.go.jp/data/000187187.pdf (Accessed Sep. 13, 2022.)

[12] Nuclear Safety Commission, "Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities," August 30, 1990.
https://warp.da.ndl.go.jp/info:ndljp/pid/9483636/www.nsr.go.jp/archive/nsc/shinsashishin/pdf/1/si008.pdf (Accessed Sep. 13, 2022.)

[13] Nuclear Regulation Authority, "NRA Ordinance Concerning the Installation and Operation of Commercial Power Reactors," 2022.
https://elaws.e-gov.go.jp/document?lawid=353M50000400077 (Accessed Sep. 13, 2022.)

[14] Nuclear Regulation Authority, "Guide for Application for Approval of Extension of Operational Period for Commercial Power Reactors," June 19, 2013.
https://www.nra.go.jp/data/000069250.pdf (Accessed Sep. 13, 2022.)

[15] Nuclear Regulation Authority, "NRA Ordinance Concerning the Installation and Operation of Reactors Used for Testing and Research," 2022.
https://elaws.e-gov.go.jp/document?lawid=332M50000002083

[16] "Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material, and Reactors," Act No. 166 of June 10, 1957. (Provisional English translation)
https://www.nra.go.jp/data/000067232.pdf (Accessed Sep. 13, 2022.)

ANNEX D: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT – SEC NRS (RUSSIAN FEDERATION)

Disclaimer: Content of this file does not represent SEC NRS or Rostechnadzor official position and is intended for purposes of Forum SMR.

Terminology

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

a) "Means" or "Provisions" versus "System(s)"

Current federal rules and regulations use both "means" and "system(s)". Usually term "means" is used for more general formulation of the requirement, such as in para. 1.2.4 of NP-001-15: *"The system of technical and administrative means shall form five levels of defense in depth and include the following levels."*

However, confinement requirements specifically formulated with the use of term "system(s)", e.g. para. 3.6.1 of NP-001-15 *"Confinement safety systems (hereinafter CSS) for confinement of radioactive substances and ionizing radiation in case of an accident within the borders stipulated in the NPP design shall be provided."*

b) Containment

Containment – the set of the NPP power unit components (including structures) enclosing the space around the reactor plant or any other facility, which contains radioactive substances. This set of components is forming the boundary as provided by the NPP design in order to prevent releases of radioactive substances and ionizing radiation into the environment in the amounts, that exceed the established limits. (In accordance with Annex N 2 "Terms and definitions" of NP-001-15)

c) Confinement

Term "Confinement" isn't defined in terminology. In federal rules and regulations term "Confinement safety system" is defined and used (e. g. para. 3.6.3 of NP-001-15 *"Confinement safety systems shall be provided for each NPP power unit and perform the specified functions in case of design basis accidents ..."* or para. 6 of NP-010-16 *"CSS should be provided in NPP unit for ensuring performance of the following functions:*

prevention or limitation of emitted radioactive substances propagation out of the confinement area under normal operation, DBA and BDBA conditions;

limitation of ionizing radiation discharge out of the confinement area under normal operation, DBA and BDBA conditions;

limitation of ambient pressure inside the containment in case of accidents;

reduction of radioactive substances concentration discharged in the confinement area during accidents;

monitoring of explosive gases concentration in hydrogen-containing mixtures in case of generation thereof in the confinement area under normal operation, DBA and BDBA conditions;

hydrogen explosion protection.")

d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

Federal rules and regulations usually define all terminology, but sometimes terms and definitions are introduced in safety guides. There is no guidance on specialized terminologies in Russian Federation at this moment. According to para. 24 of NP-010-16 *"Controlled release of radioactive substances out of the reactor facility containment is permitted in case of severe accidents only for prevention of containment destruction. In such case, the measures shall be taken to ensure radiation safety of population (by filtration of radioactive substances release, sheltering, evacuation or other measures)."*

<u>Specific requirements</u>

1) How requirements and guidance are articulated:

a. Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?

Safety requirements for confinement systems are set in federal rules and regulations in the field of atomic energy use *"Rules of design and operation of confinement safety systems"* (NP-010-16). NP-010-16 sets general requirements for confinement safety systems, that establish key safety objectives and general requirements for confinement safety system design (mostly for containment), as well as specific requirements and performance criteria.

b. What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?

NP-010-16 requirements could be divided into 2 groups. Requirements of the first group are applicable to all NPP designs and shall be met regardless of plant design, e.g. *"non-exceedance of leakage design limits of containment shall be guaranteed in all NPP states"*. One of such requirements is para. 21 of NP-010-16 "*NPP unit shall provide for the reactor facility containment. The need of containment for other systems (components) containing radioactive substances shall be justified in the NPP design. …*". Requirements of the second group are applied only if containment design utilizes specific engineering solutions such as:

– heat removal systems with active components (or passive components with moving parts);

– double-walled containment;

– doors (hatches) for separate rooms inside confinement area;

– confinement areas where medium discharge is stipulated from one room to other or out of confinement area borders (in addition to discharge through passive steam condensers) in accordance with the NPP design for the purpose of containment destruction prevention;

– active sprinkler system;

– passive steam condensation system;

– walls of passive steam condenser form a part of containment;

– emergency gas-aerosol treatment plant;

– containment or its autonomous parts for which NPP design provides for keeping of underpressure under normal operation and in case of abnormal operation, including accidents.

Requirements of NP-010-16 allow use of new materials if it is justified. Requirements for justification procedure are established in NP-010-16.

2) Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?

Specific SSCs for reliable performance of the confinement function are prescribed in all plant states, including

– normal operation (para. 3.8.4 of NP-001-15 *"… Systems for purification of gaseous media prior to release into atmosphere and water purification prior to discharge into water bodies shall be provided in the NPP design."*);

– design basis accidents (para. 3.6.1 of NP-001-15 *"Confinement safety systems (hereinafter CSS) for confinement of radioactive substances and ionizing radiation in case of an accident within the borders stipulated in the NPP design shall be provided."*);

– BDBA (para. 1.2.11 of NP-001-15: *"NPP design shall provide special technical features and administrative means, aimed at accident prevention and accident consequences mitigations and insurance of:*

…

BDBA consequences mitigation through the use of special technical features for BDBA management, application of any other technical features, applicable regardless of their initial purpose, and realisation of administrative means, including means for BDBA management and plans for protection of the personnel and the public from BDBA consequences.").

Requirements of NP-010-16 prescribe specific SSCs in following paragraphs:

25. … In cases when in order to prevent pressure rise inside the containment there are heat removal systems having active components (or passive components with moving parts), mentioned systems shall include several independent channels.

31. For containment, where overpressure could arise, the NPP design shall provide for the means of control and record of the stressed-strained states and temperature of containment building structures.

80. Confinement areas where atmosphere discharge is stipulated by NPP design from one room to other or out of confinement area borders (in addition to discharge through passive steam condensers) for the purpose of containment destruction prevention, shall be fitted with safety and (or) bypass devices (e.g., discharge valves, burst discs) with filtration of medium discharged from the confinement area.

97. The NPP design shall provide for measures to exclude non-homogeneity of solution in sprinkler system (if it is stipulated in NPP design) water drainage sumps, as well as means for treatment and keeping solution chemical composition, and measures to limit solution aggressiveness to materials inside the containment.

3) Safety Classification:

a. Under what conditions are containment provisions classified as safety systems?

Containment provisions are always classified as safety systems. In Annex N 2 *"Terms and definitions"* of NP-001-15 definitions are as follows:

Confinement safety systems (components) - safety systems (components) intended for prevention or limitation of radioactive substance and ionizing radiation release outside the borders established in the NPP design as well as their discharge to the environment.

Safety systems (components) – systems (components) intended to perform safety functions in case of design basis accidents.

b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?

Systems that support confinement functions may be classified at lower safety level if they are used only in normal operation state of the NPP, e.g. para. 3.8.4 of NP-001-15 "*... Systems for purification of gaseous media prior to release into atmosphere and water purification prior to discharge into water bodies shall be provided in the NPP design.*".

Moreover, para. 3.1.13 of NP-001-15 allows containment systems to combine safety functions with normal operation functions if multi-purpose use of these systems or their components is justified. In such cases systems and components should be classified as higher safety level.

4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

NP-010-16 establish requirements as follows:

"Confinement safety systems and their elements shall perform their designed functions with due account to external and internal events, including shockwaves, jet streams, projectiles and force loads from connected pipelines. Durability and operational capacity of the confinement safety systems and their components in operational states and design basis accidents shall be justified." (Para. 8 of NP-010-16);

"Confinement safety systems elements operational capacity under the influence of negative ambient temperatures able of resulting in water crystallization on their surfaces shall be confirmed." (Para. 9 of NP-010-16);

"Protection of confinement safety systems elements against harmful impact of microorganisms and other biological objects shall be ensured." (Para. 10 of NP-010-16).

In addition, requirements that inform the development and demonstration of the design basis of NPP for internal and external events are established federal rules and regulations NP-064-17 "*Account for external natural and human-induced impacts on nuclear facilities*".

Requirements on containment systems at DiD levels 1-4 are established in par 1.2.4 of NP-001-15 as follows:

Level 3. ...

mitigation of consequences of accidents that could not have been prevented by confinement of the released radioactive substances.

Level 4. ...

prevention of progression of beyond design basis accidents and mitigation of their consequences, including through the use of special engineering features to manage beyond design basis accidents as well as any systems (components) including normal operation systems (components) and safety systems (components) capable of performing the required functions under the given conditions;

protection of the RP containment from destruction during beyond design basis accidents and maintaining its operability.

5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?

According to para. 3.6.6 of NP-001-15 *"The acceptable leakage value for the containment shall be substantiated in the NPP design. Compliance of actual leak-tightness with the design one shall be confirmed prior to the first fueling of the reactor and checked in the course of operation with the frequency specified in the NPP design.*

Testing of the containment in the course of the NPP power unit commissioning shall be performed under design pressure and the subsequent tests shall be performed under the pressure substantiated in the NPP design. Equipment located inside the containment shall withstand testing without any loss of operability. The method and technical features, meant for test the containment for compliance with the design parameters, shall be provided in the NPP design."

Requirements for leak-tightness testing of the containment are established in para. 158 – 174 of NP-010-16 and include requirements for frequency, methodology, equipment inside the containment and addressing of detected flaws. In addition, requirements for integral testing of the containment by "absolute" method are established in Annex N 6 *"General requirements for measurements in the course of integral testing of the containment by the "absolute" method"*, which includes requirements for atmosphere, measurements, measuring transducers and control and result analysis during testing.

6) Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)

Para. 42 – 58 of NP-010-16 establish requirements for hatches, doors and airlocks, that regulate design, acceptable leakage, radiation protection.

7) Do you have specific requirements/limitations for other penetrations (e.g. pipe-runs, electrical/I&C cabling)?

Para. 59 – 67 of NP-010-16 establish requirements for pipe-runs and cabling, that regulate design, installation and means of leakage control.

8) How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?

Isolation requirements are established in Chapter V of NP-010-16 and include the following:

118. Seals of CSS components forming the border of confinement area shall provide leak-tightness established in NPP design under normal operation and DBA conditions.

119. Replacement of seals of CSS components (hatches, doors, airlocks, valves and other components) which can result in depressurization of the reactor facility containment, shall only be performed with reactor shut down (in case of water coolant used in the primary circuit, reactor facility shall be cooled down).

120. It is allowed to make sealing by welding with use of adapter components of individual doors, hatches, and repair ventilation systems' communication components. Whereas, welded joints quality shall be controlled (requirements for welded joints control are established in NP-084-15 *"Rules for control of base metal, welded joints and surface welding* *during operation of equipment, pipelines and other components of nuclear power plants"*), *as well as compliance with requirements for CSS components, including leak-tightness requirements.*

9) How are your requirements expressed to address protection from internal and external hazards?

According to para. 8 of NP-010-16 "*Confinement safety systems and their elements shall perform their designed functions with due account to external and internal events, including shockwaves, jet streams, projectiles and force loads from connected pipelines. Durability and operational capacity of the confinement safety systems and their components in operational states and design basis accidents shall be justified."*

According to para. 9 of NP-010-16: *"Operational capacity of CSS components shall be ensured at lower temperatures and its effects, such as crystallization of water on confinement safety system surfaces."*

Requirements for containment are established in para. 21 of NP-010-16: *"NPP power unit shall provide for the reactor facility containment. The need for containment for other systems (components) containing radioactive substances shall be justified in the NPP design.*

The containment shall be able to perform the following functions under NPP normal operation and in case of abnormal operation, including accidents:

…

protection of systems and components enclosed in the containment, failure of which could result in discharge of radioactive substances out of the borders set by the design, in the amounts exceeding safe operation limits, against natural and man-induced hazards in cases stipulated by the NPP design."

In addition, para. 3.4 of NP-064-17 establishes requirements as follows: *"For each external hazard considered in the nuclear facility project, there must be prepared a separate list of structures, systems and components of the nuclear facility subject to analysis of resistance to this external impact."*

10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

According to para. 8 of NP-010-16 *"Confinement safety systems and their elements shall perform their designed functions with due account to external and internal events, including shockwaves, jet streams, missiles and force loads from connected pipelines. Strength and operational integrity of the confinement safety systems and their components in operational states and design basis accidents shall be justified. In addition, confinement safety system design should take into account temperature and its effects, such as crystallization of water on confinement safety system surfaces."*

Para. 24 of NP-010-16: *"NPP design should provide technical and organisational measures for containment leakage limitation in BDBA. Measures stated above shall be aimed to limit pressure and temperature inside localization zone, explosive mixtures detonation prevention, confinement protection from jet streams force loads and missiles and limitation of the radioactive releases into the environment. …"*

Para. 62 of NP-010-16: *"Operability of leak-tight pipeline penetrations with due regard to impacts from connected pipelines, as well as operability of containment structures with due regard to thermal impacts from leak-tight penetrations shall be justified in NPP design."*

Para. 87 of NP-010-16: *"The following shall be defined in NPP design:*

…

mechanical loads and temperature effects on reactor facility containment, resulted from burning of hydrogen-containing mixtures under accident conditions, as well as possible consequences of mechanical loads and temperature effects, resulted from burning of hydrogen-containing mixtures, on NPP systems and components, including building structures enclosed by reactor facility containment."

Para. 107 of NP-010-16: *"Pipelines, equipment, fixture components and other components of passive steam condensation system shall be designed for impact of air-steam mixture flow and other dynamic effects, arising of which is possible under NPP normal operation and in case of abnormal operation, including accidents."*

11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

Para. 1.2.4 of NP-001-15 lists barriers as follows:

"Physical barrier system of the NPP unit should include: primary coolant boundary, containment and biological shielding, and, usually, (optionally) fuel matrix and cladding."

Definitions of primary coolant boundary, containment and biological shielding are established in Annex N 2 "Terms and definitions" of NP-001-15:

Reactor coolant circuit (primary circuit) - the circuit together with the volume control system (if any) intended for the coolant circulation through the core in the operation modes and conditions established in the NPP design.

Containment – the set of the NPP power unit components (including structures) enclosing the space around the reactor plant or any other facility, which contains radioactive substances. This set of components is forming the boundary as provided by the NPP design in order to prevent releases of radioactive substances and ionizing radiation into the environment in the amounts, that exceed the established limits. (In accordance with Annex N 2 "Terms and definitions" of NP-001-15)

Biological shielding - barriers (including structures) intended to protect against ionizing radiation.

12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

Each confinement system reliability index should be calculated. Confinement system reliability indices should be taken into account when determining probability of the large radioactive release. (Para. 15 of NP-010-16)

13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/ss/in service)?

Specific requirements for confinement testing are established in chapter IX of NP-010-16. Para. 142 – 151 of NP-010-16 establish general requirements for containment and other CSS testing, including test types, CSS condition during testing, addressing of detected defects. Para. 152 – 157 set out requirements for strength testing of containment, such as overpressure value, parameters that must be recorded during testing, rules of containment strength criteria establishment. Leak-tightness testing requirements are established in para. 158 – 176. These requirements include requirements for testing frequency, methodology, leak-tightness criterion, containment and enclosed equipment conditions, frequency of parameters registrations, testing of containment components.

Inspection and examination requirements are established in chapter XI of NP-010-16. Para. 215 – 229 establish requirements for scope of examination, addressing CSS components that are inaccessible for external and/or internal examination, subdivision of technical examination, condition of CSS under examination, documentation of examination and its results.

In addition, examination, testing, inspection and maintenance requirements for metal elements and welding are established in federal rules and regulation NP-084-15 "*Rules for control of base metal, welded joints and surface welding during operation of equipment, pipelines and other components of nuclear power plants*", NP-089-15 *"Rules for design and safe operation of equipment and pipelines of nuclear power installations"*, NP-104-18 *"Welding and surface welding of equipment and pipelines of nuclear power installations"* and NP-105-18 *"Rules for metal control of equipment and pipelines of nuclear power installations during manufacture and installation"*.

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

According to para. 8 of NP-010-16 *"Confinement safety systems and their elements shall perform their designed functions with due account to external and internal events, including shockwaves, jet streams, projectiles and force loads from connected pipelines. Durability and operational capacity of the confinement safety systems and their components in operational states and design basis accidents shall be justified."*

In addition, federal rules and regulations «*Rules of Nuclear Power Plant Hydrogen Explosion Protection*» (NP-040-02) establish requirements aimed at mitigation of the consequences of DBAs and BDBAs accompanied by an explosion of hydrogen-containing mixtures formed in volumes limited by containment.

15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

Para. 24 of NP-010-16 *"NPP design should provide technical and organisational measures for containment leakage limitation in BDBA. Measures stated above shall be aimed to limit pressure and temperature inside localization zone, explosive mixtures detonation prevention, confinement protection from jet streams force loads and missiles and limitation of the radioactive releases into the environment.*

In case of use of Corium Collecting and Cooling Device, reliable subcriticality of the medium enclosed herein, shall be ensured.

Controlled radioactive release from containment into the environment is permitted only to prevent containment destruction and upon condition of undertaking population radiation protection measures (such as filtration of the radioactive release, public sheltering, evacuation, etc.)."

16) What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?

"Limiting" accident scenarios could be defined from requirements of para. 25-26 of NP-010-16:

25. The NPP design shall justify that the maximum value of overpressure (underpressure) in the environment enclosed in the containment, will not exceed design-basis pressure (underpressure) in case of design-basis accidents. Non-excess of design-basis temperature in case of design-basis accidents shall also be justified. …

26. Confinement safety systems reinforced concrete structures shall be designed in accordance with standards established in the federal standards and rules in the field of use of atomic energy (NP-031-01 *"Design provisions for aseismic nuclear power plants",* NP-64-17 *"Record of external natural and human-induced impacts on nuclear facilities",* PiNAE-5.6 *"Building design provisions for NPP with reactors of different types",* etc.). *The NPP design shall justify strength and serviceability of containment building structures.*

17) How do you articulate requirements for managing containment ageing and degradation?

Confinement systems equipment, pipelines and structures ageing management program should be developed by operational organisation in compliance with requirements of para. 212-214 of NP-010-16 and requirements established in federal rules and regulations «*Requirements for ageing management of equipment and pipelines of nuclear power plants. General provisions*» (NP-096-15). Residual life assessment should be carried out during periodic safety assessment of the NPP. Life extension of equipment, pipelines or structures of the confinement safety systems should be justified by operational organisation based on the results of ageing management program.

18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

Based on SMR pre-licensing carried out by SEC NRS, ageing for SMR containment provisions and confinement system does not differ from ageing of large power reactors. In addition, federal rules and regulations allow any confinement systems' lifetime as long as their reliability during this period is justified in NPP design.

19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

a. **What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?**
b. **Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?**

There are no specific requirements that take into account plant elevation at a site. Para. 121 of NP-010-16 establish requirements for the materials as follows: *"Choice of materials used for confinement safety system elements manufacturing shall take into account their operational conditions, physical, mechanical and technological characteristics for ensuring correct performance of confinement safety systems functions during their designed lifetime."*

20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

Annex N 3 "List of possible processes (sources) of hydrogen generation" of NP-010-16 lists possible processes (sources) that could lead to hydrogen generation in normal operation state and accident conditions for different reactor types (VVER, RBMK, BBN, etc.).

ANNEX E: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT – RSA (SOUTH AFRICA)

Terminology

a) “Means” or “Provisions” versus “System(s)”

These terms are not explicitly defined in the NNR regulations; however, when reading the National Nuclear Regulator Act, Act 47 of 1999 and other regulatory documents, these terms are understood as defined in IAEA glossary. For example, section 53(1)(a)(ii) of the NNR Act states “The regulator may reproduce or cause to be reproduced documents in its possession or under its control by electronic **means**...”. The action in this statement is “reproducing” and the result – although not explicitly stated – is to exercise regulatory control/powers. Section 19 of the NNR Act states “Despite the **provisions** [conditions or requirements] of any other law [legal documents], the Regulator may not be placed under judicial management or in liquidation except if authorised by an Act of Parliament adopted specially for that purpose”. The third definition of “Systems” is very similar to that of “Means”.

b) Containment

The NNR recognizes the containment as the last DiD barrier that performs the safety function of confining (or preventing the dispersion of) radioactive materials in the primary system.

c) Confinement

Is the “safety function” of limiting the release of radioactive material in trying to fulfil the ALARA principle.

d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

The NNR applies regulatory requirements on the applicant or licensee’s safety related programmes & processes, which requires the applicant or licensee to demonstrate safety, compliance with regulatory safety criteria & requirements as well as demonstrate good engineering practices and the use of codes and standards.

The NNR applies IAEA definitions in terms confinement & containment; and require the licensee or applicant to provide information regarding the type of nuclear installation, potential source terms, barrier concept/s with respect to confinement of fission, activation and/or radioactive products, heat removal, reactivity control etc.

Considering the ALARA requirement, it is the task of the designer to demonstrate whether a full pressure confinement building (ie. containment) or a confinement building with filtered depressurisation function is the best solution.

The NNR regulatory framework also entails position papers which requires the applicant or licensee to demonstrate the derivation of target safety goals which are required when establishing the design basis parameters for the design of nuclear installations against external events using the performance-goal based approach.

Regulatory documents require the applicant or licensee to demonstrate that safety functions shall be available as appropriate in normal operation, during and following AOOs, design base accidents, design base extension conditions and severe accidents to facilitate response to severe accidents.

Regulations set out the safety assessment requirements in general to be complied with by applicants or authorisation holders and specifically address safety analyses, both deterministic and probabilistic safety analysis; prior safety assessments; operational safety assessments; accident management; PSR; worker safety assessments; and public safety assessments.

Specific requirements on safety assessments for nuclear facilities and specifically addresses, amongst other internal and external hazards; events selection and classification etc. are also required.

A Regulatory Guidance document provides general guidance to current or prospective licence holders on the documented evidence that are acceptable to the regulator on safety assessments.

Specific Requirements

1. **How requirements and guidance are articulated:**
   a. **Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?**
   NNR requirements are written in such a way to identify key safety objectives to be met. The NNR develops requirements in line with international codes and standards, and provide guidance through Regulatory Guides (RG) to applicants or authorization holders on how to meet them, and then verify that all regulatory requirements are complied with throughout the lifetime of the regulated activity.
   The NNR licensing process requires the applicant to present a safety case to the NNR which is a structured presentation of documented information, analyses and intellectual arguments to demonstrate that the proposed design can and will comply with the NNR licensing requirements. The NNR regulatory philosophy is a hybrid of both process-based licensing & prescriptive. By process based, the NNR applies requirements on applicant's safety related programmes & processes. This non-prescriptive approach requires the applicant to demonstrate safety, compliance with overall regulatory safety goals & requirements as well as demonstrate good engineering and the use of codes and standards. The NNR looks at safety related aspects. The licensing philosophy is non-prescriptive concerning the adoption of codes and standards for design and operation. However, the applicant has to use internationally acceptable codes and standards; and has to justify that the selected codes and standards are sufficient to support the safety case. The NNR is also prescriptive is our regulations in terms of dose limits, operator control room, operator licensing.

b. **What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?**
As in the previous response, the NNR's licensing philosophy is a hybrid so it remains upon the applicant to present a safety case to the NNR and rigorously demonstrate that the proposed design can and will comply with the NNR overall licensing requirements. The NNR does not prescribe specific materials for the containment structure, however, the applicant must demonstrate that the chosen material meets the acceptance criteria for a functional containment.

2. **Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?**
The NNR's regulatory philosophy is non-prescriptive provided the applicant uses recognized international codes and standards in design, manufacturing and assembly.
The NNR has a specific Regulatory document that defines the general quality & safety management requirements needed to ensure reliability of products and therefore safety is taken into account. We also apply a grading system that applicants or licensees must adopt to classify SSCs with their importance to safety.

3. **Safety Classification**:
   a. **Under what conditions are containment provisions classified as safety systems?**
   The safety case must define the confinement system which may include various barriers. Containment provisions in LWR serve as a barrier of radiological confinement. In the same hand, radiological confinement is a fundamental safety function. As such, containment provisions are classified as important to nuclear safety systems.

   b. **Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?**
   According to NNR Regulatory Guidance documents on management of safety, the method for safety classification of SSC's for nuclear facilities should take account taken of factors such as the:
   i. Safety function(s) to be performed by the item;
   ii. Consequences of failure to perform the safety function;
   iii. Frequency at which the item will be called upon to perform a safety function; and
   iv. Time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function.

   Therefore, SSCs that support containment and confinement functions wherein the safety sub-function to be performed by the item is of lesser significance, with little consequences in the event that the item fails to perform the safety function, and infrequent need for the item to perform its safety function; will be classified at a lower safety level.

4. **Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and**

external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

Design basis internal and external events must be postulated that may compromise the confinement function.

The design must include both redundant and diverse means, i.e., DID to ensure that the confinement function is not compromised.

There are requirements that in line with principal radiation protection and nuclear safety requirements, which stipulate that the principles of DiD are to be applied in accordance with the appropriate international standards, so that there are multiple layers provided by the SSCs, and procedures, to ensure that the fundamental safety functions are met.

For the PBMR, regulatory license documents required the applicant to demonstrate that safety functions would be provided to ensure that the fundamental safety functions are maintained and to provide the required levels of DiD considering reliability requirements and NNR safety goals.

5. **How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?**

   The NNR does not have specific requirements for ensuring sufficient leak tightness, however, the NNR Regulations and Requirement documents outlines radiation dose and risk limits.

   In addition, the frequency consequence curve can be derived from risk analyses.

   An applicant or authorization holder can use these radiation dose and risk limits in conjunction with recognized international codes and standard to demonstrate in the safety case that small releases will still comply with Principal Radiation Protection and Nuclear Safety Requirements.

6. **Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other access ways?)**
   We have no specific requirements. The applicant is required to demonstrate, with reasonable assurance, that the design for construction and operation of the facility is adequate to protect the radiological health and safety of workers and to comply with the regulatory requirements during routine and non-routine operations, including anticipated events and accident conditions. NNR requirements document does refer to criteria for penetrations. The applicant is to use recognized international standards in developing/adopting a criteria for the safety and suitability of any kind of penetrations.

7. **Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)**
   See response to number 6.

8. **How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?**

We have no specific requirements. In NNR Position Paper (PP-0017, Design and implementation of digital instrumentation and control for nuclear installations), the NNR draws guidance on design and implementation of digital instrumentation and control for nuclear installations from various international nuclear safety agencies and organisations. Guidance on containment isolation systems is drawn from the USNRC's 10 CFR 50.34(f)(2)(xiv)
In 10 CFR 50.34(f)(2)(xiv) the requirements for containment isolation are articulated in a prescriptive manner to achieve the main safety goal, which is, to prevent or limit the escape of fission products that may result from postulated accidents.

9. How are your requirements expressed to address protection from internal and external hazards?

The NNR gives requirements that address protection from internal and external hazards. More relevant to SMRs, the regulation states that "For multiple unit facility sites, the design shall take due account of the potential for specific hazards giving rise to simultaneous impacts on several facilities on the site". The requirements of this regulation are also expressed in a non-prescriptive manner, in that, the Regulator does not make recommendations on means of achieving the goals set forward by the requirements but rather it (NNR) highlights key areas that the licensee must address.

It is expected that the applicant designer defines performance goals for SSC in line with the NNR overall safety goals. (NNR Position Paper, PP-0014 on Consideration of External Events for NI's).

10. How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

It is expected that the applicant designer defines performance goals for SSC in line with the NNR overall safety goals. See PP-0014 on Consideration of External Events for NI's.

Further the applicant/designer must justify the use of relevant industry standards commensurate with the reliability targets for the SSC being considered.

Where systems or functions that handle loads from arising from pressure, temperature, radiation, combustible gases, and mechanical impact are classified as accident mitigation measures. These systems and functions include but are not limited to containment isolation system, containment pressure indication, containment spray system, containment heat removal system(s) and containment atmosphere radionuclide mitigation system.

11. How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

The NNR regulations stipulate that the three fundamental safety functions are maintained. The number of barriers shall depend on the magnitude (risk) the radiological hazard and the consequences of failure and to ensure resilience, the regulations recommends that the multiple barriers should be independent of each other. The main function/task of containment/confinement barriers as stated in the definitions under "Terminology" is to prevent or control the release and the dispersion of radioactive substances. These barriers also serve the purpose of shielding primary SSCs against external missile strikes.

12. How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

The regulatory guide on management of safety states that the SSC should be designed, manufactured, installed and subsequently commissioned, operated and maintained to a level of quality and *reliability* commensurate with their classification.

On design and development, NNR regulatory guide states that "If possible the SSC should be of a design proven in previous equivalent applications, and should be consistent with the *reliability* goals determined for the respective plant SSC. Where new or innovative design or features are used, the authorisation holder should provide the results of the investigations on applicability of the codes and standards to the regulator. It should be demonstrated that the selected codes and standards are fully applicable to the plant SSC. In any other case a revised code, standard or specification should be developed and approved" here the Regulator is refraining from prescribing reliability standards but rather turns to more established international codes and standards; these have quantitative reliability requirements.

The NNR looks to in-service inspections as another tool to insure reliability of SSCs. NNR Requirements documents state:

An authorisation holder shall prepare and implement documented programmes for the regular and systematic maintenance, testing, surveillance, and inspection of systems, structures or components which are important to nuclear safety to ensure that their availability, reliability, and required functionality remain in accordance with the assumptions and intent of the design over the service lifetime of the nuclear facility.

The maintenance and inspection programme shall ensure the reliability and integrity of equipment and plant having an impact on nuclear safety and shall be commensurate with the radiation hazard associated with the nuclear facility.

13. How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/commissioning/in service)?

The NNR has recently issued regulations and guidance (RG-0027) on Ageing Management and LTO of NPPs. The Ageing Management Programmes for components must specify testing, etc. programmes. *See response to number 12*

14. How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

NNR Regulations has the following requirements:

c) The design of a facility shall take due account of internal hazards such as fire, explosion, flooding, missile generation, collapse of structures and falling objects, pipe whip, jet impact and release of fluid from failed systems or from other facilities on the site. Appropriate features for prevention and mitigation shall be provided to ensure that safety is not compromised.

d) The design of a facility shall include due consideration of those natural and human induced external events (i.e. events of origin external to the facility) that have been identified in the site evaluation process. Natural external events shall be addressed, including meteorological, hydrological, geological and seismic events. Human induced external events arising from nearby industries and transport routes shall be addressed.

h) The design shall be such as to ensure that items important to nuclear safety are capable of withstanding the effects of external events considered in the design, and if not, other features such as passive barriers shall be provided to protect the facility and to ensure that the required safety function will be performed.

15. How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

The NNR regulatory documents stipulate that the concept of DiD are to be applied to all safety related activities. Deterministic safety analysis should be used to assess the adequacy of the design and should cover both normal operations and abnormal behaviour and should be supported by appropriate probabilistic analysis to judge the significance of uncertainties, show that risks are balanced, and demonstrate compliance with numerical risk criteria. Deterministic safety analysis should be used to analyse AOO's, DBA's and DBEC's.

The NNR requires that probabilistic safety analysis must be performed to demonstrate compliance with the numerical risk criteria unless it can be justified that no credible accident conditions exist.

16. What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?

IE, PIE's and LBE must be defined by the applicant designer using a systematic approach. These are typically technology dependent.

17. How do you articulate requirements for managing containment ageing and degradation?

Regulatory Guidance documents on Ageing Management and LTO of NPPs (RG-0027)

furthermore, stipulates that "the authorisation holder should describe ageing management as part of the management system for the NPP; the authorisation holder should develop, implement and maintain an ageing management programme comprising the functions, duties and responsibilities for assuring the operability and technological conformance of SSCs important to nuclear safety throughout operating life of the facility".

18. Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

The recently issued Regulatory Guide, RG-0027 on Ageing Management and LTO of NPPs, provides for a generic approach applicable to all components, including SMR containment provisions.

For PBMR we had a specific licensing document which required the applicant to provide a safety case that demonstrated the adequacy of the facility design and operational procedures against the regulatory licencing requirements. The applicant was required to demonstrate compliance with the regulatory licensing requirements by way of a formal safety analyses, with reference to proven technology and in accordance with international practices. The analyses required both deterministic & probabilistic analyses.

There may be room to develop requirements with regards to how to address over reliance by the designers on their designs.

19. Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

No, aside from design basis flood levels to be defined considering site specific factors.

a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?

It would be site and design specific. We have not progressed to that stage in any application that have been processed. It is expected that soil structure interactions, geohydrological factors be considered in the design of the below grade construction, including water ingress, etc. therefore corrosion issues.

b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?

It would be site and design specific.

20. **Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group**.

Regulations No. R.266 Regulations on the Long-Term Operation of Nuclear Installations

Regulatory Guide, RG-0027 on Ageing Management and LTO of NPPs

Draft Specific Nuclear Safety Regulations: Nuclear facilities

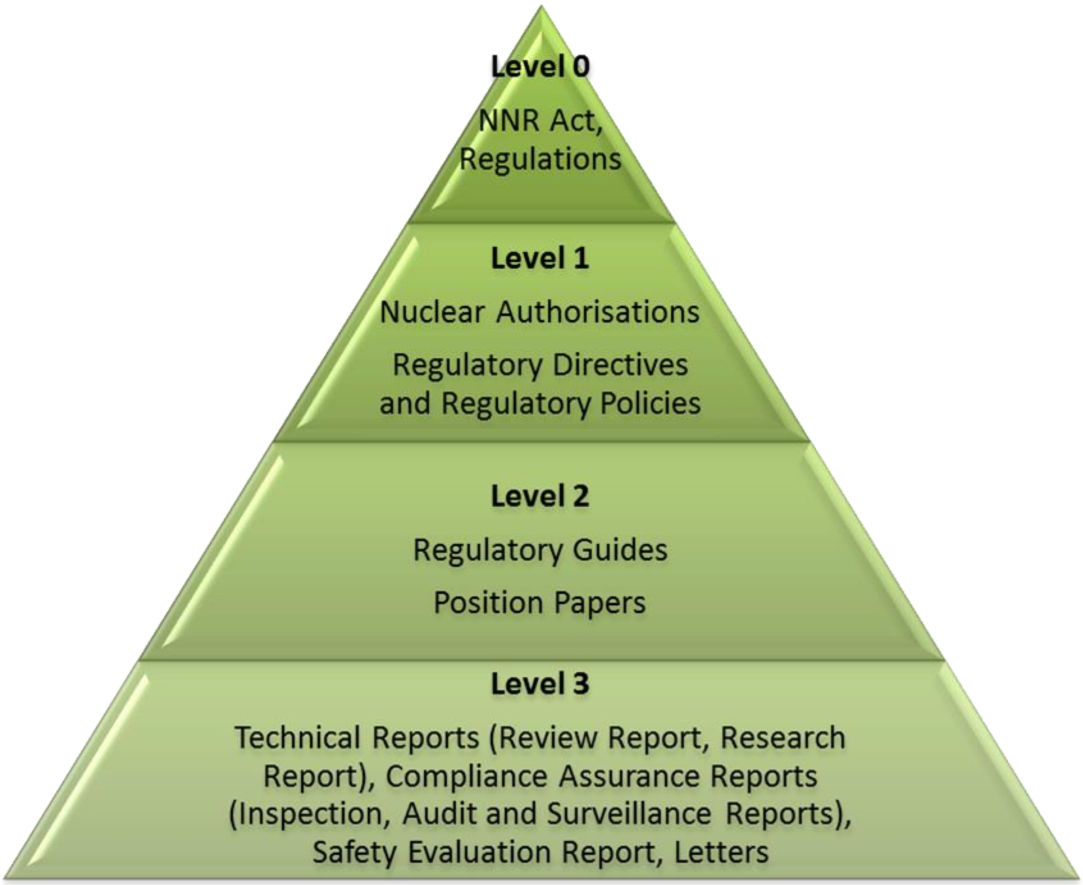
Regulatory Guide, RG-0019 on Guidance on safety assessments of nuclear facilities

Regulatory Guide, RG-007 on Management of Safety

Position Paper, PP-0014 on Consideration of External Events for NI's

Position Paper, PP-0017, Design and Implementation of digital instrumentation and control for Nuclear Installations

NNR Regulatory Framework



ANNEX F: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT- ONR (UNITED KINGDOM)

Terminology

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

a) “Means” or “Provisions” versus “System(s)”

Not explicitly defined in ONR regulatory guidance, although all three terms are used in the ONR Safety Systems Technical Assessment Guide (TAG) with their common English definitions.

“Safety system” is defined in ONR Safety Assessment Principles (SAPs) as:

“*A system that acts in response to a fault to protect against a radiological consequence.*”

b) Containment

Defined in ONR SAPs glossary:

“*Methods or physical structures designed to prevent the dispersion of radioactive material.*”

Also SAPS paragraph 520 states:

“*The term ‘containment’ encompasses a wide range of structures and plant items, from the massive buildings surrounding power reactors, to glove boxes and individual packages and containers. Containments often have associated systems, such as cooling systems and sprays, which are considered to be part of the containment system.*”

Relevant extract from ONR TAG NS-TAST-GD-020, “Civil Engineering Containments for Reactor Plants”:

“*Although related to confinement, containment is usually used to refer to methods or structures that perform a confinement function in facilities and activities, namely preventing or controlling the release of radioactive substances and their dispersion in the environment.*”

c) Confinement

Defined in ONR TAG NS-TAST-GD-020, “Civil Engineering Containments for Reactor Plants”:

“*Prevention or control of releases of radioactive material to the environment in operation or in accidents. Confinement is closely related in meaning to containment, but confinement is typically used to refer to the safety function of preventing the ‘escape’ of radioactive material, whereas containment refers to the means for achieving that function. Confinement in nuclear safety is the safety function that is performed by the containment.*”

d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework.)

No, but the nature of ONR’s goal-setting regulatory philosophy extends to the identification of safety functions and the SSCs that fulfil them. As such, ONR’s non-prescriptive approach would not preclude a ‘functional containment’ type justification for demonstrating the adequacy of the confinement function being developed by the licensee in their safety case.

Specific requirements

1) **How requirements and guidance are articulated:**
   a. **Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?**
   ONR as a goal-setting regulator outlines regulatory expectations in publicly available guidance such as the ONR SAPs and TAGs, rather than by setting prescriptive requirements. ONR expects that the safety functions to be delivered should be identified and categorised based on their significance with regard to safety. The SSCs identified to deliver the safety functions should be appropriately classified in accordance with the category of the safety function being delivered. This will help inform the performance, inspection and maintenance requirements for the design life.

   b. **What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to ~~for alternatives~~ provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?**
   The licensee is free to propose alternative means to achieve equivalent safety outcomes, provided they can demonstrate that the selected design option reduces the risk to as low as reasonably practicable (ALARP). The holistic design is assessed on a case-by-case basis in accordance with the safety functions assigned to SSCs claimed within the safety case. It is expected that multiple, independent and diverse arguments should provide a robust, multi-layered justification in which weaknesses in individual layers of the argument are offset by strengths in others, which could include the use of proven materials.

2) **Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?**
   ONR's regulatory framework is goal-setting and largely non-prescriptive and as such we would not typically prescribe specific SSCs. However, we recognise that there are proven means to reduce risks to ALARP in specific circumstances/applications, i.e. relevant good practice, and we expect the licensee to justify any significant deviation or shortfall against this in the safety case, in order to demonstrate that they comply with the legal duty that the design choices reduce the risk to ALARP.

3) **Safety Classification:**
   a. **Under what conditions are containment provisions classified as safety systems?**
   ONR expects that structures, systems and components (SSCs) that have to deliver safety functions should be identified and classified on the basis of those

functions and their significance to safety (as captured in SAP ECS.2). This also applies to the containment provisions.

b. **Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?**
As above, the safety classification is expected to be linked to the safety significance of the safety function being delivered, thus if the consequences of failure to deliver the safety function of 'confinement of radioactive materials' can be shown to be low, then the classification of the SSC(s) delivering that function may also be reduced.

4) **Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?**
SAP ECV.2 states that containment and associated systems should be designed to minimise radioactive releases to the environment in *normal operation, fault and accident conditions*, thus covering all levels of defence in depth.
In the above context, ONR SAPs define 'faults' as "Any unplanned departure from the specified mode of operation of a structure, system or component due to a malfunction or defect within the structure, system or component or due to external influences or human error". This includes internal and external hazards.

SAP FA.2 sets out the expectation that fault analysis should identify all initiating faults with the potential to lead to a significant dose of radiation or release of radioactive material, and FA.3 sets out the expectation that fault sequences following from the initiating faults to the potential consequences should be analysed. SAP FA.5 provides expectations for the inclusion of initiating faults in design basis analysis with an initiating event frequency (IEF) $\geq 10^{-5}$ pa and natural hazards with a predicted return frequency $\geq 10^{-4}$ pa. SAP FA.6 outlines a typical cut-off for design basis analysis of 10^{-7} pa.

EHA.1 sets out the expectation that an effective process should be applied to identify and characterise all external and internal hazards that could affect the safety of the facility. EHA.19 states that hazards whose associated faults make no significant contribution to overall risks from the facility should be excluded from the fault analysis. EHA.3 expects that for each internal or external hazard which cannot be excluded on the basis of either low
frequency or insignificant consequence (see Principle EHA.19), a design basis event should
be derived. As per SAP EHA.4, the thresholds set in Principle FA.5 for design basis events are 1 in 10 000 years for external hazards and 1 in 100 000 years for man-made external hazards and all internal hazards (see also paragraph 629). EHA.6 expects that the effects of internal and external hazards that could affect the safety of the facility should be analysed. The analysis should take into account hazard combinations, simultaneous effects, common cause failures, defence in depth and consequential effects. EHA.5 expects that the analysis of design basis events should assume the event occurs simultaneously with the facility's most adverse permitted operating state.

At DiD 3-4, EHA.18 expects that *for fault sequences initiated by internal and external hazards beyond the design basis should be analysed applying an appropriate combination of engineering, deterministic and probabilistic assessmen*ts and EHA.7 expects that *a small change in design basis fault or event assumptions should not lead to a disproportionate increase in radiological consequences.*

5) **How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?**
SAPs ECV.1: Prevention of leakage - "*Radioactive material should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.*"
SAPs ECV.2: Minimisation of releases - "Containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions."
SAPs ECV.6 and 7: Provide expectations for leak monitoring.
SAPS: ECE.22 - "*Civil engineering structures that retain or prevent leakage should be tested for leak tightness prior to operation.*"
TAG-020 states:
The concept of defence in depth should be applied to the civil structure design features that provide the safety functions, normally shielding and containment. The use of multiple engineered containment barriers with provision for preventing, collecting and monitoring minor leakage represent relevant good practice. The required leak tightness of the civil structure containment barriers should take account of:
- The degree of radiological contamination of the contents, including any contained liquids.
- The chemical properties of the contained liquids.
- The potential for the degradation of civil structure containment barriers by action of the contents.
- The environmental consequences of leakage of the contents.
- The potential for harm to operators and the public from leakage of a radiological or chemically harmful inventory.
-

6) **Do you have specific requirements/limitations for large penetrations (e.g. airlocks/hatches/ other accessways?)**
SAP ECV.5 states that the need for access by personnel to the containment should be minimised.
SAP ECE.20 states "*Provision should be made for inspection, testing and monitoring during normal operation aimed at demonstrating that the structure continues to meet its safety functional requirements. Due account should be taken of the periodicity of the activities.*"
SAP ELO.1 states "*The design and layout should facilitate access for necessary activities and minimise adverse interactions while not compromising security aspects*".
SAP ELO.4 states "*The design and layout of the site, its facilities (including enclosed plant), support facilities and services should be such that the effects of faults and accidents are minimised.*" For example, the design and layout should minimise the direct effects of initiating events, particularly from internal and external hazards, on structures, systems or components.

7) **Do you have specific requirements/limitations for other penetrations (e.g. pipe-runs, electrical/I&C cabling)?**
SAP ECV.3 states the expectation that the licensee will "*minimise the size and number of service penetrations in the containment boundary, which should be adequately sealed to reduce the possibility of radioactive material escaping via routes installed for other purposes.*" This is consistent with ELO.4 ONR's internal hazards TAG expects that penetrations on divisional / fire barriers (such as doors, dampers, cable and pipework penetrations, and so on) should be avoided where practicable. If this is not feasible, then the location of penetrations should be optimised. The penetrations should be readily identifiable and maintained at suitably frequent intervals to ensure the appropriate level of reliability.

ONR's internal hazards TAG, in the general considerations section paras. 5.7 to 5.9 outlines the expectation that:

…*for internal hazards that cannot be completely eliminated or prevented, the severity of the hazard can be reduced by a number of means. This can be done by favouring benign materials, fluids and operating envelope, for example by limiting combustible or flooding source inventories, or operating at lower pressures and temperatures*. Paragraph 5.8. emphasises that *limiting the consequences of hazards can be preferably ensured by good plant layout principles (SAP ELO.4), and by protecting the plant from the hazard loadings*. *The latter should be generally achieved by the provision of robust passive barriers, which withstand the maximum credible loadings and segregate safety systems that allow the continued delivery of nuclear safety functions*.
Paragraph 5.9. states that *the nuclear safety consequences of hazards can also be limited by ensuring that equipment important to safety withstands the hazard loadings by, for example, suitable qualification to the specific dynamic and environmental conditions of the hazard, ensuring sufficient separation or shielding*. **This extents to the design and protection of penetrations or any such features of the design.** Paragraph 5.9 finally concludes that *approaches based entirely on hazard distances or heavily reliant on SSC qualification may be challenging to support in the absence of suitable segregation*.
It is important to recognise the hierarchy of measures in para 155 and that the provision of one measure e.g. protecting a penetration does not per se justify not implementing other measures that may be practicable such as improving the layout to keep it further away from the hazards source.

8) **How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?**
SAP ECV.4 states that when considering secondary containment, the design should include appropriate means of isolation.
SAP ECV.10 covers controls on ventilation systems which sets out the expectation for isolation to protect against identified faults and SAP ECV.3 sets out the expectation that the use of ducts that need to be sealed by isolating valves under fault conditions should be avoided.
SAP EHT.4 states that isolation devices (on heat transfer systems) should be provided to limit any loss of radioactive fluid.
SAP ENM.3 - "*Temporary isolations should be effective and controlled by suitable management arrangements. Particular attention should be paid to situations in which*

ineffective or partially effective temporary isolations could lead to unintended transfers of nuclear matter, eg through leaking valves."

9) **How are your requirements expressed to address protection from internal and external hazards?**
TAG 020 sets out the expectation that a detailed schedule of loading for both serviceability and ultimate limit states such as normal operations, plant transients, faults and internal and external hazards should be prepared and that the design analysis covers potential failure modes for conditions arising from design basis faults and potential in-service degradation mechanisms.
Further guidance is provided in:
TAG 013 - External Hazards
TAG 014 - Internal Hazards
TAG 017 - Civil Engineering
General provisions on protection against internal hazards are described in the TAG 014 reference paragraphs included in question 7 above. In addition to this and the hierarchy of measures in SAPs para 155, the ONR Internal Hazards TAG provides specific expectations on safety measures/ protection for each internal hazard in turn (see TAG 015 section 5).
General provisions on protection against external hazards are described in TAG 013 and the supporting annexes on seismic hazards, meteorological hazards, coastal flood hazards and accidental aircraft crash hazard. More information on safety measures is included in TAG 013.

10) **How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?**
See answer to Q9.
Alternative technologies would not be excluded where they can be demonstrated to be resilient against the schedule of loading for normal operations and accident conditions.
In relation to hot gas releases, ONR internal hazards TAG provides general expectations and reference to RGP which recognises the goal setting nature of the regulatory framework, the flexibility in approach and the hierarchy of measures in EKP.3 and para 155 of the SAPs. As an example, TAG 014 expects that "*Efforts should be made to minimise the number and energy of the steam and hot gas release sources, and to place them in areas furthest away and preferably segregated from nuclear safety significant plant (by suitably qualified barriers and penetrations) so far as is reasonably practicable. In conjunction with this, there may be features to direct the gas via an engineered route away from release points out to open air via vents, louvres, or quick release dampers. In some cases, it has been necessary to qualify essential SSCs against the effects of the hot gas.*" Subsequent paragraphs (5.102 to 5.105) in TAG 014 refer to expectations in the prevention and protection against high temperature / pressure loadings. Similar guidance is provided for each internal hazard in turn in TAG 014.

Appendix 4 of TAG 013 sets out the expectations for "industrial hazards". These hazards arise either due to the conveyance of hazardous materials on adjacent transport routes (for example, pipeline, rail, road and sea) or adjacent permanent facilities (for example, quarries, tank farms etc).

11) **How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?**
No prescriptive requirements. On a case-by-case basis the licensee must adequately demonstrate the performance of the barrier(s), including for fault and hazard conditions, and that the design choices have reduced the risk to ALARP.
ECV.4 states: "*Where the radiological challenge dictates, waste storage vessels, process vessels, piping, ducting and drains (including those that may serve as routes for escape or leakage from containment) and other plant items that act as containment for radioactive material, should be provided with further containment barrier(s) that have sufficient capacity to deal safely with the leakage resulting from any design basis fault.*"

12) **How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?**
ONR sets out numerical targets for dose, frequency and risk in the SAPs. In general, the higher the frequency of the fault, the lower the target on-site and off-site doses. This ultimately drives reliability requirements for particular SSCs to be commensurate with the potential risk arising from failure of that SSC.

SAPs ECE.2 sets out the expectation that the safety case should consider the required resilience of civil engineering structures when subject to beyond design basis loadings during severe accidents.

ONR SAPs EDR.1 to EDR.4 cover general regulatory expectations around designing for reliability, including principles such as failure to safety, redundancy, diversity and segregation, common cause failure, and the single failure criterion.

13) **How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g. construction/commissioning/in service)?**
ONR SAPs EMT.1 to EMT.8 covers the general regulatory expectations for maintenance, inspection and testing.
SAP ECE.20 covers the expectations around inspection and testing of civil engineering structures (such as a typical PWR containment), with further guidance provided in TAG 020, which refers out to IAEA NS-G-2.6 for guidance on the frequency of inspection for PWR type reactor containments, and ASME XI B&PV code for pre-stressed and reinforced concrete containments.
The expectation for a periodic containment leak test in accordance with Appendix J to USNRC guide 10CRF50 for PWR containments is also set out in TAG 020.
SAP ECE.21 states: "*Pre-stressed concrete pressure vessels and containment structures should be subjected to a proof pressure test, which may be repeated during the life of the facility.*"
SAP ECE.22 states: "*Civil engineering structures that retain or prevent leakage should be tested for leak tightness prior to operation.*"

SAP ECV.7 states: “*Appropriate sampling and monitoring systems should be provided outside the containment to detect, locate, quantify and monitor for leakages or escapes of radioactive material from the containment boundaries.*”

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

See answers to Q4 and Q9.

SAPs ECV.3 paragraph 525. (h) states: “Where appropriate, the safety case should: provide for discharge routes, including pressure relief systems, with treatment system(s) to minimise radioactive discharges to acceptable levels. There should be appropriate treatment or containment of radioactive wastes generated by such systems. Should the pressure relief system operate, the performance of the containment should not be degraded”

SAPs ECV.3 paragraph 525. (i) states: “Where appropriate, the safety case should: justify the continuing safe functioning of the containment and its discharge routes in faults or accidents involving combustible, explosive and/or toxic gases”

SAP EHA.14 states: “*Sources that could give rise to fire, explosion, missiles, toxic gas release, collapsing or falling loads, pipe failure effects, or internal and external flooding should be identified, quantified and analysed within the safety case*”.

SAP EHA.8 paragraph 521 states: “*The direct and indirect effects of aircraft crashes on structures, systems and components needed to achieve a stable, safe state should be analysed. These should include effects relating to mechanical resistance, vibrations and structural and component integrity.*”

SAP EHA.8 paragraph 252 states: “*The analysis should include fire and explosion hazards deriving from aircraft crashes including fires caused by aircraft fuel, fire ball and pool fire combinations and other consequential fires due to the aircraft crash. Buildings (or parts of buildings) containing nuclear fuel or housing structures, systems and components needed to achieve a stable, safe state should be designed to prevent aircraft fuel from entering them.*”

SAPS EPS.3 and 4 set-out the expectation for adequate pressure relief systems, overpressure protection and periodic testing.

In addition, ONR expects the licensee to perform severe accident analysis to help identify any further safety measures that could reduce the risk of radiological consequences or mitigate them, so far as is reasonably practicable.

15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

ONR SAP EQU.1 sets out the expectation that qualification procedures should be applied to confirm that SSCs will perform their allocated safety functions in normal operations, fault and accident conditions (including severe accidents).

ONR SAP EHA.18: ”*Fault sequences initiated by internal and external hazards **beyond the design basis** should be analysed applying an appropriate combination of engineering, deterministic and probabilistic assessments*”.

ONR SAP EHA.7 ‘Cliff-edge effects’ states: “*A small change in design basis fault or event assumptions should not lead to a disproportionate increase in radiological* *consequences.*”. This drives the expectation that fault sequences beyond the design basis will be considered.

ONR SAP ECE.6 states: “*Load development and a schedule of load combinations, together with their frequencies, should be used as the basis for structural design. Loadings during normal operating, testing, design basis fault and accident conditions should be included. To preclude cliff edge effects, margins to failure should extend beyond design basis fault (or hazard) loadings by an amount consistent with assumptions in the severe accident analysis. Beyond design basis loading considerations should be included before the structural design is finalised. Special attention should be paid when assessing existing structures not designed in accordance with current standards or codes.*”

ONR SAP ECE.1 states: “*The required safety functions and structural performance of the civil engineering structures under normal operating, fault and accident conditions should be specified… Margins should be such that civil engineering structures will continue to provide their residual safety function(s) following the application of beyond design basis loads by either having sufficient design margins, or by failing in a manner that suitably limits the radiological consequences.*”

SAPs ECE.2 sets out the expectation that the safety case should consider the required resilience of civil engineering structures when subject to beyond design basis loadings during severe accidents.

16) **What is the approach for defining the “limiting” accident scenarios used in the containment design (e.g. for large LWRs this may be main steamline break/LOCA)?**

In-line with international guidance regarding the practical elimination of large and early releases, for any credible accident scenario that is not physically impossible or extremely unlikely to occur with a high degree of confidence, it should be demonstrated that sufficient confinement of radioactive materials is maintained to meet the numerical targets for dose and risk set out in the SAPs. The licensee must also demonstrate the risk from such sequences has been reduced to ALARP. For fault sequences with very low frequency of occurrence (e.g. beyond the design basis), it may be acceptable to perform this demonstration on a best estimate basis, with margin to cover uncertainties.

17) **How do you articulate requirements for managing containment ageing and degradation?**

General expectations for ageing and degradation are set out in ONR SAPs EAD.1 to EAD.5 covering aspects such as safe working life, lifetime margins, obsolescence, and periodic measurement of material properties and other relevant parameters. Further guidance is provided in ONR TAG 020, which makes reference to IAEA NP-T-3.5.

SAP ECE.2 identifies the expectation that the safety case will consider potential in-service degradation mechanisms.

SAP ECE.8 states that if elements cannot be inspected, the safety case should demonstrate with high confidence that the performance of these elements will remain adequate for the design life.

SAP ECE.21 states pre-stressed concrete pressure vessels and containment structures should be subjected to a proof pressure test, which may be repeated during the life of the facility.

SAP ECE.16 states "*The construction materials used should comply with the design methodologies employed, and be shown to be suitable for enabling the design to be constructed and then operated, inspected and maintained throughout the life of the facility*."

18) **Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?**
SMR operational design life (and construction / decommissioning lifetime) is typically shorter than a traditional PWR.

19) **Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?**
    a. **What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g. material types, siting restrictions etc)?**
    SAPs paragraph 263 and 264 state: "In line with Principle EKP.3 (defence in depth), consideration should be given to extreme hydrological phenomena. The design of all structures, systems and components needed to deliver the fundamental safety functions in any permitted operational states should be augmented by protection from water ingress and waterproofing as a redundant measure to provide a further barrier in the event of flooding of the site.
    All structures, systems and components vulnerable to failure from water intrusion, submergence or consequential effects that cannot be placed above the design basis flood level should be protected by engineered features designed to prevent water intrusion and submergence and protect against consequential effects"
    SAP ECE.11 states "The design should take account of the possible presence of naturally occurring explosive, asphyxiant or toxic gases or vapours in underground structures such as tunnels, trenches and basements."
    The siting ST.1 - ST.6 cover expectations around siting, including ST.4, which states "*The suitability of the site to support safe nuclear operations should be assessed prior to granting a new site licence*".
    The impact of below-grade construction on seismic response should be considered as expected by SAP EHA.9 which states: "The seismology and geology of the area around the site and the geology and hydrogeology of the site should be evaluated to derive a design basis earthquake (DBE)".
    ECE.5 and 10 consider geotechnics and groundwater, including the effects of climate change.
    SAP EHA.11 states: "*Facilities should be shown to withstand weather conditions that meet design basis event criteria. Weather conditions beyond the design basis that have the potential to lead to a severe accident should also be analysed.*" SAP paragraph 259 also states: "*The reasonably foreseeable effects of climate change over the lifetime of the facility should be taken into account, particularly during Periodic Safety Reviews.*"

    b. **Are there any specific technical criteria that would need to be addressed for below grade structures (e.g. ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?**

Yes, there are regulations covering work in confined spaces and restriction of exposure to ionising radiation, and radiological protection (RP) SAPs. There are conventional safety regulations covering these aspects.

Also regarding ability to inspect, SAP ELO.1 states: "*The design and layout should facilitate access for necessary activities and minimise adverse interactions while not compromising security aspects*". This would be expected to include consideration of access for inspection and maintenance activities. ONR Internal Hazards TAG 014 outlines expectations in relation to internal hazards that are transferable to this context. Flammable gas accumulation leading to fire and/or explosion may be exacerbated by increased use of below grade structures. Adequate ventilation is expected to reduce the potential of flammable gas accumulation. The DSEAR ACOP and guidance provide expectations on what constitutes adequate ventilation. Expectations regarding the prevention, control and mitigation of internal flooding are also available in TAG 014.

From an external hazards perspective, below grade structures may, for example, reduce the risk from some meteorological hazards. However, further understanding of the impact of below grade construction on seismic response and hydrogeology may be required.

20) **Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.**

ONR's expectations regarding defence in depth are captured in SAP EKP.3, which also states:

"*An important aspect of the implementation of defence in depth is the provision of multiple, and as far as practicable independent, physical barriers to the release of radioactive material to the environment, and to ensure the confinement of radioactive material at specified locations. The number of barriers will depend on the magnitude of the radiological hazard and the consequences of their failure.*"

ANNEX G: RESPONSE TO QUESTIONNAIRE ON CONTAINMENT - USNRC (UNITED STATES OF AMERICA)

Terminology:

Please describe your regulatory interpretation of the following terms, including formal definitions and where they are expressed in your framework:

- **a) “Means” or “Provisions” versus “System(s)”**
- **b) Containment**
- **c) Confinement**
- **d) Has your organization developed interpretations or guidance of specialized terminologies such as Functional containment? (Please list and describe including references to your regulatory framework)**

U.S. Nuclear Regulatory Commission (NRC) Response:

a) In general, the NRC does not use the terms “Means” or “Provisions” in regulations related to system(s) performance requirements. However, these terms may appear in some guidance documents to refer to abilities or ways, and acceptable options of required actions or conditions, respectively. The NRC uses the term structures, systems, and components (SSCs) to describe the design attributes for plant specific containment design characteristics for the purposes of fission product retention when performing their design basis function(s).

b) For light water reactor (LWR) designs:

10 CFR Part 50, Appendix J, II.A states that “‘Primary reactor containment’ means the structure or vessel that encloses the components of the reactor coolant pressure boundary, as defined in § 50.2, and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.” (See 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 16 regarding requirements for containments).

For non-light water reactor (Non-LWR) designs:

Non-light water reactor (LWR) technologies have operating conditions, coolants, and fuel forms that differ from LWRs. As described in SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors,” dated September 28, 2018, these differences may allow or possibly require different approaches to fulfilling the safety function of limiting the release of radioactive materials. This has led to describing a “functional containment” as a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment.

c) Generally, “confinement” is not a term that is used in NRC regulations with respect to reactor containment of fission products for plant normal operations or accident conditions. However, confinement is a term that can generally refer to structures, systems, and components (SSCs) that act as barriers between areas containing radioactive substances and the environment. (Also refer to SRM-SECY-03-0047, dated June 26, 2003 (ADAMS Accession No. ML031770124), for NRC Commission's position on "confinement").

d) The NRC has issued Regulatory Guide (RG) 1.233, which provides guidance for using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-LWR applicants who intend to use a licensing approach based on the Licensing Modernization Project (LMP) for advanced reactor technologies – an industry-led effort supported by the U.S. Depart of Energy (DOE) (Refer to Idaho National Laboratory report INL/EXT-18-46151, issued September 2018). The methodology provides one approach to establishing the systems, structures, and components (SSCs) that perform fundamental safety functions including retention of fission products. This constitutes "functional containment" approach for non-LWRs as discussed in SECY-18-0096. In addition, the staff is developing a new technology-inclusive and risk-informed regulatory framework for advanced reactors under 10 CFR Part 53. The staff has developed preliminary proposed rule language under which would define functional containment [§ 50.280] as "set of barriers taken together that effectively limit the physical transport and release of radionuclides to the environment across the full spectrum of events [§ 50.250 Anticipated operational occurrences and design basis accidents, § 50.260 Beyond-design-basis events, § 50.270 Severe accidents]."

Specific Requirements:

1) How requirements and guidance are articulated:

a. Please describe how your requirements are written to address structures that support containment functions? For example, are requirements and guidance written in such a way to identify key safety objectives to be met or prescribe the design and performance criteria of specific structures or both?

b. What degree of flexibility is provided to permit the proposal and demonstration of alternative ways to for alternatives provide confinement/containment functions (including but not limited to specific material requirements for the containment structure)?

NRC Response:

a) Requirements to support containment functions are written in the regulations. The regulations identify key safety objectives or criteria under which specific items are to be addressed in ensuring acceptable containment functions. For LWRs, regulations provide functional requirements and, regulatory guidance provide possible ways to meet the regulations. For example, 10 CFR Part 50, Appendix A, GDC 16—Containment design and GDC 50—Containment design basis, are applicable to containment design. GDC 50 states, in part:

> The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be

> designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

Another example is GDC 51—Fracture prevention of containment pressure boundary, which states in part:

> The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions: (1) its ferritic materials behave in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized.

NRC staff's comprehensive review guidance, such as the Standard Review Plan (SRP) (NUREG-0800), provides the guidance to follow in a more detailed and systematic manner to ensure all regulatory requirements are satisfactorily addressed by an applicant. Also, applicable Regulatory Guides (RGs) provide insights for applicants to understand what information the NRC staff will expect in some specific areas of a licensing application to ensure certain regulations are satisfactorily addressed. For LWRs, SRP Section 6.2.1.1.A, "PWR Dry Containments, Including Subatmospheric Containments," providing guidance to the NRC staff states that "To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a loss-of-coolant accident, or a steam or feedwater line break."

For non-LWRs using LMP-based licensing approach, the design requirements for SSCs that fulfil the fundamental safety function of fission production retention, thus supporting containment functions, would be performance-based and depend on the required function of the SSCs. As described in RG 1.233, SSCs that are intended to perform safety functions of preventing and mitigating Licensing Basis Events (LBEs) are divided in two broad categories – safety related (SR) and nonsafety-related with special treatment (NSRST); safety-significant SSCs include all those SSCs classified as SR or NSRST. Performance criteria for the reliability and capability of the SSCs fulfilling safety-significant functions for non-LWRs are defined; a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs are established. The safety classification of SSCs is made in the context of how the SSCs perform specific safety functions for each LBE in which they play a role in preventing or mitigating an event.

b) All RGs for LWRs contain a statement confirming the applicability of alternate ways in meeting regulations:

> The NRC issues regulatory guides to describe and make available to the public methods that the NRC staff considers acceptable for use in implementing specific parts of the agency's regulations, techniques that the staff uses in evaluating specific problems or postulated accidents, and data that the staff needs in reviewing applications for permits and

> licenses. Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in regulatory guides will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

For non-LWRs using LMP-based licensing approach, the requirements are performance based and allow significant flexibility for the applicant to propose which SSCs are credited with fulfilling the safety function of fission product retention. RG 1.233 describes its purpose as follows:

> The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

2) Under what conditions do you prescribe specific SSCs required to support reliable performance of the containment function?

NRC Response:

GDC 38—Containment heat removal, states that "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

GDC 41—Containment atmosphere clean-up, states that "Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained."

10 CFR 50.44 requires that all containments must have a capability for ensuring a mixed atmosphere. Combustible gases must be controlled to establish and maintain safe shutdown and containment structural integrity with systems and components capable

of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

For non-LWRs using LMP-based licensing approach, the NRC has issued RG 1.233 which provides guidance for using a technology-inclusive, risk-informed, and performance-based methodology to inform the licensing basis and content of applications for non-LWRs. The methodology includes defining performance criteria for the reliability and capability of the SSCs fulfilling safety-significant functions for non-LWRs including fission product retention.

The NRC has also issued RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," which provides guidance on how the GDC in 10 CFR 50, Appendix A, may be adapted for non-LWR designs. The guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria (PDC) for any non-LWR design. PDCs that are developed for a specific design would specify the conditions that are required to be addressed by the designs SSCs that are used to fulfil the safety function of fission product retention.

10 CFR 50.44 requires that, for future non water-cooled reactors, all containments must have: (1) information addressing whether accidents involving combustible gases and technically relevant for their design, and (2) if accidents involving combustible gases are found to be technically relevant, information (including a design-specific probability risk assessment) demonstrating that safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common defense and security.

3) Safety Classification:

   a. Under what conditions are containment provisions classified as safety systems?

   b. Under what conditions might systems that support containment and confinement functions be classified at a lower safety level?

NRC Response:

a) In nuclear processes, containment provisions that are designed for protection of the reactor and prevention of fission products release to the outside environment are called safety systems. Safety-Related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design basis events (Refer to 10 CFR 50.2). Containments, classified as safety related, are designed with capabilities to prevent, or mitigate the consequences of accidents by confining fission products that otherwise might be released to the atmosphere and result in potential significant offsite exposures.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria and includes assessing and classifying non-LWR SSCs as safety related, nonsafety-related with special treatment, or nonsafety-related with no special treatment. SECY-18-0096 describes an approach to "functional containment" for non-LWRs that may not rely on traditional containment structures to limit the physical transport and release of radioactive
material to the environment. The transport of fission products can be adequately modelled for all barriers and pathways to the environs, including the specific consideration of containment design.

b) SSCs can be classified at a lower safety level if they do not meet the definition for safety related SSCs provided in 10 CFR 50.2.

Risk-Informed Safety Class (RISC)-III SSCs (safety-related, low safety significance per 10 CFR 50.69), which are safety-related SSCs that a risk-informed process has determined as not important to safety, may be subject to a reduced level of quality verification (e.g., quality assurance requirements and containment penetration leak rate testing).

An example of SSCs being classified at a lower safety level is discussed in NUREG-0800 (SRP), "Introduction—Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition." (Revision 0, January 2014). A safety-significance categorization process classifies SSCs in one of four review levels that correlate to safety significance:

(1) A1—safety-related and risk significant;
(2) A2—safety-related and non-risk significant;
(3) B1—not safety-related and risk significant; and
(4) B2—not safety-related and non-risk significant.

SSCs are categorized as either safety-related or not safety-related using the criteria in 10 CFR 50.2, and as either risk significant or not risk significant using the process developed for the reliability assurance program. The SSCs within the scope of the reliability assurance program are identified by using a combination of probabilistic, deterministic, and other methods of analysis to identify and quantify risk, including probabilistic risk assessment, severe accident evaluation, assessment of industry operating experience, and expert panel deliberation.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria and includes assessing and classifying non-LWR SSCs as safety related (SR), NSRST, or nonsafety-related with no special treatment. None of the nonsafety-related with no special treatment SSCs are classified as safety significant, but they may have requirements to ensure that failures following a design-basis internal or external event do not adversely impact SR or NSRST SSCs in their performance of safety-significant functions.

4) Please describe, at a high level, the requirements and guidance that inform the development and demonstration of the containment design basis for internal and external events. Are there requirements on containment systems at DiD levels 1-4 for all plant states and if so, how are they expressed in requirements and guidance?

NRC Response:

GDC 4—Environmental and dynamic effects design bases, states, in part, “Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.”

GDC 2—Design bases for protection against natural phenomena, states, in part:

> Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

GDC 16 —Containment design establishes the fundamental requirement to design a containment that is essentially a leak-tight barrier against the uncontrolled release of radioactivity to the environment.

GDC 50 —Containment design basis, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the emergency core cooling system, the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters in assessing containment design margins.

GDC 51—Fracture prevention of containment pressure boundary:

> The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions: (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material

> properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

10 CFR 50.44 requires containments of LWRs have an inerted atmosphere or limit hydrogen concentrations in containment during and following an accident.

In addition to using a conditional containment failure probability of 0.1, the Commission approved the use of deterministic containment performance goal in the evaluation of the passive advanced LWRs (Staff Requirement Memorandum related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," July 21, 1993):

> The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

10 CFR 52.47(a)(23) states that an application must include the following information: "For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass."

NUREG-0800 (SRP), Section 6.2.1, provides guidance for review of containment functional design, and states, in part, that the containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long-term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require. This SRP Section guides the reviewer to perform reviews under its various subsections.

NUREG-0800 (SRP), Section 6.2.7 provides guidance for review of Fracture Prevention of Containment Pressure Boundary, and states, in part, that the reactor containment system design must include the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant to postulated accidents. This guidance ensures compliance with GDC 1, as it relates to the quality standards for design and fabrication; GDC 16, as it relates to the prevention of the release of radioactivity to the environment; GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions: (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized; as well as relevant requirements in 10 CFR 52.47(b)(1) and 10 CFR 52.80(a).

NUREG-0800 (SRP) Chapters 3 and 19 provide guidance for the review of containment and SSCs design basis for external events, such as, seismic, extreme winds, tornadoes, flooding, aircraft hazards, externally generated missiles, storm surge, tsunami, volcanism, etc.

10 CFR Part 52 identifies requirements on how applicants are expected to address defense-in-depth. Review guidance on defense-in-depth (DiD) for containment systems is described in NUREG-0800 (SRP), Chapter 19. SRP Section 19.0, states, in part, that the staff will determine whether the applicant has identified risk-informed safety insights based on systematic evaluations of the risk associated with the design., including the design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria that includes a framework to establish defense-in-depth that includes probabilistic and deterministic assessment techniques using a combination of plant capabilities and programmatic controls. This approach evaluates fission product retention for a spectrum of events including anticipated operational occurrences, design basis accidents and beyond-design-basis accidents including evaluation of uncertainties, and it includes an assessment of so-called "cliff-edge effects" for events of even lower likelihood. RG 1.232 provides the non-LWR methodology for developing PDC for any non-LWR design. PDCs that are developed for a specific design would specify internal and external events that are required to be addressed by the designs SSCs.

5) How is prevention and mitigation of small releases addressed? For example, do you have specific requirements for ensuring sufficient leak tightness?

NRC Response:

The description of the reactor containment leakage rate testing program is reviewed for conformance to 10 CFR Part 50, Appendix J, and GDC 52, 53, and 54. Appendix J includes two options, A and B, either of which can be chosen by an applicant or licensee for meeting the requirements of the appendix. Option A is "Prescriptive Requirements," and Option B is "Performance-Based Requirements." GDC 52, "Capability for Containment Leakage Rate Testing," relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the containment integrated leakage rate tests (up to the containment design pressure). GDC 53, "Provisions for Containment Testing and Inspection," relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate testing at the containment design pressure of penetrations having resilient seals and expansion bellows. GDC 54, "Piping System Penetrating Containment," relates to piping systems penetrating primary reactor containment being designed with a capability to determine if valve leakage rate is within acceptable limits.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria for small releases. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing licensing basis events (LBEs) which may include criteria for leak tightness or allowable leakage limits.

6) Do you have specific requirements/limitations for large penetrations (e.g., airlocks/hatches/ other accessways?)

NRC Response:

Pursuant to 10 CFR 50.55a requirements, the ASME Boiler and Pressure Vessel Code standards for Quality Group B apply to containment penetrations.

For airlocks, the lock structure and both doors must be designed and constructed for the same pressure as the containment vessel.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria for penetrations. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs.

7) Do you have specific requirements/limitations for other penetrations (e.g., pipe-runs, electrical/I&C cabling)?

NRC Response:

Pursuant to 10 CFR 50.55a requirements, the ASME Boiler and Pressure Vessel Code standards for Quality Group B apply to containment penetrations.

For pipe-runs,

- Prevent excessive pipe loads from being transferred to containment system;
- Prevent pipe from restraining the containment system during thermal or pressure expansions; and
- Minimize heat transfer into shell.

Electrical penetrations must be able to maintain containment integrity over long periods of time under normal operating conditions and over shorter periods under the much more severe conditions that would result if an accident occurred.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria for other penetrations. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs.

8) How are specific requirements for containment isolation articulated and what safety objectives are they required to addressed?

NRC Response:

GDC 54 —Piping systems penetrating containment, as it relates to the requirement that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety, and as it relates to designing such piping systems with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC 55 —Reactor coolant pressure boundary penetrating containment, as it relates reactor coolant pressure boundary piping penetrating containment.

GDC 56 —Primary containment isolation, as it relates to primary containment atmosphere isolation.

GDC 57 —Closed system isolation valves, as it relates to the requirement that lines penetrating the primary containment boundary that are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment.

These GDCs are to address that the primary reactor containment providing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria for containment isolation. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs. RG 1.232 provides a methodology for developing PDC for any non-LWR design. PDCs that are developed for a specific design would specify containment isolation capabilities that are required to be addressed by the designs SSCs.

9) How are your requirements expressed to address protection from internal and external hazards?

NRC Response:

In accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to withstand the differential pressure loadings that may be imposed because of pipe breaks within the containment subcompartments.

GDC 4 —As it relates to designing safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and as it relates to the requirement that these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.

GDC 2 —As it relates to designing safety-related SSCs to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform safety functions.

For non-LWRs using LMP-based licensing approach, RG 1.232 provides a methodology for developing PDC for any non-LWR design. PDCs that are developed for a specific design would specify internal and external events that are required to be addressed by the designs SSCs.

10) How do you articulate requirements for loads management (such as those arising from pressure, temperature, radiation, combustible gases, and mechanical impact) in a containment/confinement? To what degree do they permit the demonstration and use of alternative technologies?

NRC Response:

<u>Applicable requirements are listed below</u>:

GDC 38 —As it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

GDC 50 —As it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate and with sufficient margin.

The structural performance of the containment under severe accident loads reviewed by the staff encompasses: (1) the applicant's assessment of the Level C (or factored load) pressure capability of the containment in accordance with 10 CFR 50.44(c)(5); (2) the applicant's demonstration of the containment capability to withstand the pressure and temperature loads induced by the more likely severe accident scenarios as stipulated in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," Section I.J; (3) the applicant's containment structural fragility assessment for overpressurization; and (4) the applicant's assessment of the seismic capacity of the containment structure in meeting the expectation documented in SECY-93-087, Section II.N.

The RG 1.216 describes methods that the NRC staff considers acceptable for: (1) predicting the internal pressure capacity for containment structures above the design basis accident pressure, (2) demonstrating containment structural integrity related to combustible gas control, and (3) demonstrating containment structural integrity through an analysis that specifically addresses the Commission's performance goals related to the prevention and mitigation of severe accidents.

Any technology not using LMP-based licensing approach, must address all applicable regulations, including those listed above, and must include requests, for Commission approval, for exemption from those that may seem not applicable for the technology. Applicants for these technologies may propose alternatives for Commission consideration.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining performance criteria for load management. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs. RG 1.232 provides the non-LWR methodology for developing PDC for any non-LWR design. PDCs that are developed for a specific design would specify load management considerations that are required to be addressed by the designs SSCs.

11) How do you articulate requirements to ensure an appropriate number of and sufficient resilience of barriers that confine radioactive materials? Is a definition of tasks/functions of containment/confinement barrier(s) provided?

NRC Response:

Traditional large light reactors (LWRs) include protections against design-basis accidents that reflect the traditional approach for multiple barriers providing "defense in depth" to limit releases of radioactive material: fuel cladding, reactor coolant system pressure boundary, and containment. The containment structures for LWRs have been designed to control the leakage of radioactive materials following design-basis accidents that can damage the fuel cladding and pressure boundary. The performance criteria for LWR containments are defined as allowable leakage rates, which are determined from analyses performed to show that estimated radiation doses to members of the public resulting from a design basis accident are below the reference values cited by NRC regulations.

GDC 16 —Containment design, as it relates to establishing an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assuring that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

SECY-18-0096 describes an approach to “functional containment” for non-LWRs that may not rely on traditional containment structures to limit the physical transport and release of radioactive material to the environment. The transport of fission products can be adequately modelled for all barriers and pathways to the environs, including the specific consideration of containment design. A "functional containment" is a barrier, or a set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment. Figure 1, “Functional Containment Performance Criteria,” in this SECY provides a depiction of how the NRC staff's proposed methodology would help designers establish performance criteria for plant features using accepted event categories and fundamental safety functions such as controlling reactivity and reactor power, removing heat, and limiting the release of radioactive materials from a reactor facility.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for defining performance criteria for the reliability and capability of the SSCs fulfilling safety-significant functions for non-LWRs including fission product retention.

12) How is the reliability of systems addressed in your requirements? For example, do you have any quantitative reliability requirements for containment systems (active and passive)?

NRC Response:

For LWRs, SRP Chapter 19, Section 19.0, provides the following acceptance criteria:

- The risk associated with the design compares favourably against the Commission’s goals of less than 1 x 10^{-4} per year for core damage frequency and less than 1 x 10^{-6} per year for large release frequency.
- The design compares favourably against the Commission’s approved use of a containment performance goal, which includes: (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges; and (2) a

probabilistic goal that the conditional containment failure probability be less than 0.1 for the composite of all core damage sequences assessed in the probabilistic risk assessment.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining system reliability requirements. The methodology establishes a systematic approach to assessing and determining appropriate relationships between the needed capabilities and reliabilities for SSCs and the role of those SSCs in mitigating and preventing LBEs. The process includes assessing event sequences over a wide range of frequencies and establishing risk and safety function reliability measures.

13) How do you articulate containment-specific requirements for testing, examinations, inspections, and maintenance (e.g., construction/commissioning/in service)?

NRC Response:

For LWRs, Appendix J to 10 CFR 50 provides containment leakage testing requirements and SRP Chapter 14, Section 14.3, provides Inspections, Tests, Analyses, and Acceptance Criteria. Also, 10 CFR 50.55a specifies requirements for inservice inspection of containment.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for establishing the framework for defense-in-depth adequacy for containment-specific requirements for testing, examinations, inspections, and maintenance. Programmatic defense-in-depth includes measures to increase confidence in SSC performance during operation and throughout the life of a plant (e.g., quality assurance, testing, maintenance, and configuration control), operational procedures and training, and preparedness for emergency plan protective actions.

14) How are the effects of extreme conditions (e.g., explosions within the barrier) and environmental conditions due to accidents, including conditions arising from the external and internal events, required to be taken into account in the design of confinement provisions?

NRC Response:

- 10 CFR 52.47(a)(23) states that an application for licenses, certifications, and approvals for nuclear power plants must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information: “For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass;"

- 10 CFR 50.44 requires structural analyses to demonstrate maintenance of structural integrity following severe fuel damage accompanied by combustible gas events.
- 10 CFR 50.155(b)(2), "Extensive damage mitigation guideline" states that each applicant or licensee shall develop, implement, and maintain:

> Strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant impacted by the event, due to explosions or fire, to include strategies and guidelines in the following areas:
>
> (i) Firefighting;
>
> (ii) Operations to mitigate fuel damage; and
>
> (iii) Actions to minimize radiological release.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for establishing the framework for defense-in-depth adequacy for containment-specific performance-based requirements for extreme and beyond-design-basis events. RG 1.232 provides the non-LWR methodology for developing PDC for any non-LWR design. PDCs that are developed for a specific design would specify internal and external events that are required to be addressed by the designs SSCs.

15) How is resiliency of the design provisions beyond DBA addressed in your requirements? For example, do you have specific containment related requirements for DECs and for severe accidents?

NRC Response:

An applicant is expected to adequately address the Commission's objectives regarding the appropriate way to address consideration of severe accidents and the use of PRA in the design and operation of facilities under review. These objectives are outlined in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Section C.I.19.2.

For LWR designs, 10 CFR 52.47(a)(23) and 10 CFR 52.79(a)(38) state that a design certification (DC) application and a combined license (COL) application, respectively, must contain a final safety analysis report that includes a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

As stated in SRP Section 19.0, the NRC staff evaluates the applicants assessment of structural performance of the containment under severe accident loads which encompasses: (1) an assessment of the Level C (or factored load) pressure capability of the containment in accordance with 10 CFR 50.44(c)(5), (2) demonstration of the containment capability to withstand the pressure and temperature loads induced by the more likely severe accident scenarios as stipulated in SECY-93-087, Section I.J, (3) a containment structural fragility assessment for overpressurization, and (4) seismic high-confidence-of-low-probability-of-failure assessment of the containment in meeting the SECY-93-087, Section II.N, expectation.

SRP Section 19.0 identifies Commission direction and staff guidance published in multiple documents from which acceptance criteria have been developed. The NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of probabilistic risk assessment (PRA). The SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur. In particular, the SRM on SECY-93-087 provides direction about the treatment of external events in PRAs to support DC and COL applications.

RG 1.216 describes methods, in part, that the NRC staff considers acceptable for demonstrating containment structural integrity through an analysis that specifically addresses the Commission's performance goals related to the prevention and mitigation of severe accidents.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for using the probabilistic risk assessment in an expanded role to develop or confirm the events, safety functions, key SSCs, and adequacy of DiD; and provides a structured framework to risk-inform the application for the specific reactor design. The probabilistic risk assessment plays a key role in the methodology which uses a systematic process for identifying and categorizing event sequences as anticipated operational occurrences (AOOs), design-basis events (DBEs), or beyond-design-basis events (BDBEs). The methodology provides a way to assess a range of events to determine risk significance, support SSC classification, determine special treatment requirements, identify appropriate programmatic controls, and confirm the adequacy of DiD.

16) What is the approach for defining the "limiting" accident scenarios used in the containment design (e.g., for large LWRs this may be main steamline break/LOCA)?

NRC Response:

The NRC staff's overall review approach ensures that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents or initiating events. Initiating events are categorized according to expected frequency of occurrence and by type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Categorization of initiating events by type provides a basis for comparison between events, which makes it possible to identify and evaluate the limiting cases (i.e., the cases that can challenge the analysis acceptance criteria). Refer to SRP, Section 15.0, for additional details on a systematic approach the NRC staff adopts for verifying and evaluating limiting accident scenarios.

For LWRs, a loss of coolant accident (LOCA) or main steam line break (MSLB) would typically establish the peak containment pressure and temperature profile. The maximum hypothetical accident (as defined in 10 CFR Part 100 or 10 CFR 50.67) establishes a source term for evaluation of offsite consequences that relates to limits for containment leak tightness.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for the identification of events that could challenge key safety functions and layers of defense against the release of radioactive materials. A systematic process is used to categorize these events. The methodology provides the process for deriving the limiting design basis accidents from these events. RG 1.232 provides the non-LWR methodology for developing PDC for any non-LWR design. PDC are expected to include a criterion similar to the following:

> PDC 50 —Containment design basis:
>
> The containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. This margin shall reflect consideration of: (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

17) How do you articulate requirements for managing containment ageing and degradation?

NRC Response:

- 10 CFR 50.36(c) requires Technical Specifications to include surveillance requirements, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

- 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” specifies requirements for licensees to monitor the performance or condition of SSCs, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components can fulfil their intended functions.
- Overall containment structural integrity is assured by 10 CFR 50.55a requirements for containment inservice inspection and the Type A integrated leak rate testing requirements of 10 CFR Part 50, Appendix J.

For non-LWRs using LMP-based licensing approach, RG 1.233 provides a methodology for determining programmatic controls including monitoring, surveillance and maintenance for safety related and nonsafety-related SSCs with special treatment. These programmatic controls would consider failure modes and effects including ageing and degradation.

18) Have you seen any predictions or foresight of ageing for SMR containment provisions and systems (without going into specific technology necessarily)?

NRC Response:

The NRC staff are not aware of any concerns. However, for novel designs, attention should be paid to different operating conditions and environments such as new materials, high temperatures, effects of radiation, new construction techniques and corrosive environments.

19) Related to establishment of plant elevation at a site (above-ground, below ground, etc.), do you have specific requirements taking different elevations into account in the design of means of containment?

   a. What restrictions or conditions may be applicable for below-grade construction of containment structures (e.g., material types, siting restrictions etc)?

   b. Are there any specific technical criteria that would need to be addressed for below grade structures (e.g., ventilation of containments/shielding provided by the ground /ability to inspect/retrofit etc.)?

NRC Response:

NRC regulations do not distinguish based on the plant elevation. However, the seismic and hydrostatic loading on the containment structure and the ability to inspect will change based on the grade level and would have to be addressed in the design.

20) Please list your other regulatory requirements for confinement of radioactive materials which may be relevant to this Working Group.

NRC Response:

Additional requirements for SSCs performing a containment or confinement function, such as quality assurance (10 CFR 50 Appendix B), environmental qualification (10 CFR 50.49), reliability assurance program (Commission policy in the Staff Requirement Memorandum on SECY 95-132), and maintenance rule (10 CFR 50.65), would apply.