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| ONR Assessment Report  Generic Design Assessment of the Rolls-Royce SMR – Step 2 assessment of Mechanical Engineering |



ONR Assessment Report

**Project Name**: Generic Design Assessment of the Rolls-Royce SMR

**Report Title**: Step 2 assessment of Mechanical Engineering

**Authored by**: [Redacted]

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# Executive Summary

This report presents my mechanical engineering Step 2 assessment findings for the Rolls-Royce Small Modular Reactor (SMR) design. It forms part of the Office for Nuclear Regulation’s Generic Design Assessment. My assessment used Rolls-Royce SMR Limited's (the Requesting Party) Environment, Safety, Security and Safeguards case.

The objective of my assessment was to identify any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the design (subject to site specific assessment and licensing). My judgement informs the Office for Nuclear Regulation’s decision on its Step 2 statement.

My Step 2 assessment considered the generic Rolls-Royce SMR design safety. It examined the Requesting Party’s claims, arguments and supporting evidence. This enabled me to make a judgement on the Requesting Party’s Environment, Safety, Security and Safeguards case mechanical engineering adequacy.

My assessment conclusions are that the Requesting Party’s generic Environment, Safety, Security and Safeguards case has, for the purposes of Step 2 of the Generic Design Assessment, demonstrated how the:

* safety case is logically structured and accessible;
* limits and conditions of safe operations development approach aligns with relevant good practice;
* passive safety approach aligns with relevant good practice;
* defence in depth approach aligns with relevant good practice;
* categorisation and classification methodology considers principles of prevention;
* examination, inspection, maintenance and testing strategy aligns with relevant good practice; and
* equipment qualification approach aligns with relevant good practice.

Shortfalls have been identified. These relate to how the layout design meets regulatory expectations for:

* personnel access and egress;
* mechanical handling; and
* examination, inspection, maintenance and testing.

I judge that these matters can be resolved during Step 3. This is because the Requesting Party has the opportunity to address the shortfalls and present supporting evidence.

I also conclude that the Requesting Party has not yet shown that the:

* safety case has a completed claims, arguments and evidence structure and robust requirements traceability;
* containment heating, ventilation and air-conditioning system design requirements for mechanical engineering are complete;
* seismic performance requirements for mechanical engineering are understood.

Overall, based on my assessment to date, and subject to the provision and assessment of suitable and sufficient supporting evidence, I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic Rolls-Royce SMR design at Step 2.

# List of Abbreviations

ADS Automatic Depressurisation System

ALARP As low as reasonably practicable

AIV Automatic Isolation Valve

AR Assessment Report

ASME American Society of Mechanical Engineers

CAE Claims Arguments Evidence

CoFT Control of Fuel Temperature

CoR Control of Reactivity

CRDM Control Rod Drive Mechanism(s)

DAC Design Acceptance Confirmation

DBE Design Basis Earthquake

DOORS Dynamic Object-Orientated Requirements System

DiD Defence in Depth

DR Definition Review

DRP Design Reference Point

EAH Equipment Access Hatch

EBDV Emergency Blowdown Valve

ECC Emergency Core Cooling

EIMT Examination, Inspection, Maintenance and Testing

EQ Equipment Qualification

FCD Final Concept Design

FHM Fuel Handling Machine

FSF Fundamental Safety Function

HOW2 ONR’s Management System internal online portal

HSE Health and Safety Executive

HP High Pressure

HTOP High Temperature Operating Pressure

HVAC Heating Ventilation and Air Conditioning

IAEA International Atomic Energy Agency

IHP Integrated Head Package

LCU Local Cooling Unit

LP Low Pressure

LTOP Low Temperature Operating Pressure

LUHS Local Ultimate Heat Sink

MKoP Modular Kit of Parts

MOC Main Overhead Crane

MSCV Multistage Control Valve

MSD Mechanical Sequence Diagram

MSIV Main Steam Isolation Valve

OLM On-Line Maintenance

ONR Office for Nuclear Regulation

OPEX Operating Experience

PDHR Passive Decay Heat Removal

PGA Peak Ground Acceleration

PIE Postulated Initiating Event

PSA Probabilistic Safety Assessment

PSRB Product Safety Review Board

RCP Reactor Coolant Pump

RCS Reactor Coolant System

RD Reference Design

RGP Relevant Good Practice

RI Reactor Island

RPV Reactor Pressure Vessel

SAP Safety Assessment Principle(s)

SG Steam Generator

SMR Small Modular Reactor

SPC Seismic Performance Classification

SRV Safety Relief Valve

SSC Structure, System and/or Component

TAD Target Audience Description

TAG Technical Assessment Guide(s) (ONR)

TLA Through-Life Activities

TSC Technical Support Contractor

VFD Variable Frequency Drive

V&V Verification and Validation

WENRA Western European Nuclear Regulators’ Association

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# Introduction

1. This report presents the outcomes of my mechanical engineering assessment of the Rolls-Royce Small Modular Reactor (SMR) as part of Step 2 of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA). This assessment is based upon the information presented in version 2 of Rolls-Royce SMR Limited’s Environmental, Safety, Security and Safeguards (E3S) case chapters (refs [1], [2], [3], [4], [5], [6], [7], [8], [9]) and supporting documentation.
2. My assessment was undertaken in accordance with the requirements of the Office for Nuclear Regulation (ONR) Management System and follows ONR’s guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [10]). The ONR Safety Assessment Principles (SAPs) (ref. [11]), together with supporting Technical Assessment Guides (TAGs) (ref. [12]), have been used as the basis for this assessment.
3. This is a Major report (refer to NS-PER-GD-108 (ref. [13])).
   1. Background
4. The ONR’s GDA process (ref. [14]) calls for a step-wise assessment of the Requesting Party's (RP) submissions with the assessments increasing in detail as the project progresses. Rolls-Royce SMR Limited is the RP for the GDA of the Rolls-Royce SMR design.
5. In April 2022 ONR, together with the Environment Agency and Natural Resources Wales (NRW), began Step 1 of the GDA for the generic Rolls-Royce SMR design. Step 1, which is the preparatory part of the design assessment process and mainly associated with initiation of the project and preparation for technical assessment in later steps, was successfully completed in 12 months.
6. Step 2 commenced in April 2023. This is the first substantive technical assessment step. The focus of ONR’s assessments in this step is towards the fundamental adequacy of the design and safety and security cases, and the suitability of the methodologies, approaches, codes, standards and philosophies which form the building blocks for the design and generic safety and security cases. The objective is to undertake an assessment of the design against regulatory expectations to identify any fundamental safety or security shortfalls that could prevent ONR permissioning the construction of a power station based on the design.
7. Prior to the start of Step 2 I prepared a detailed Assessment Plan for mechanical engineering (ref. [15]). This has formed the basis of this assessment and was also shared with the RP to maximise openness and transparency.
8. This report is one of a series of Assessments which support ONR’s overall judgements at the end of Step 2 which are recorded in the Step 2 Summary Report (ref. [16]).
   1. Scope
9. The assessment documented in this report is based upon the E3S case for the Rolls-Royce SMR design as summarised in the E3S case chapters and supporting documentation.
10. The overall scope of the Rolls-Royce SMR GDA is described in (ref. [17]). Rolls-Royce SMR Limited has indicated that it intends to complete a three step GDA, with the objective of receiving a Design Acceptance Confirmation (DAC) from ONR and have aligned their GDA scope with this objective. The GDA scope defines the generic plant and layout and includes all systems, structures and components (SSC) that are identified as being important to safety, security and safeguards, all modes of operation, and all stages of the plant lifecycle.
11. My assessment included consideration of the following aspects, as defined in ONR’s GDA Technical Guidance section 3.13 (ref. [18]):

* key structure, system and/or component (SSC) functional requirements and design parameters;
* the RP’s examination, inspection, maintenance and testing (EIMT) approach with consideration of design for access and inspectability;
* demonstration that risks have been reduced as low as reasonably practicable (ALARP);
* the RP’s mechanical handling approach;
* application of operating experience (OPEX);
* defence in depth, redundancy, diversity, segregation and layout approach; and
* equipment qualification approach.

# Assessment standards and interfaces

1. For ONR, the primary goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a safety, security and safeguards case for the reactor technology being assessed.
2. ONR has a range of internal guidance to enable Inspectors to undertake a proportionate and consistent assessment of such cases. This section identifies the standards which have been considered in this assessment.
3. This section also identifies the key interfaces with other technical topic areas.
   1. Standards
4. The ONR Safety Assessment Principles (SAPs) (ref. [19]) constitute the regulatory principles against which the RP’s case is judged. Consequently, the SAPs are the basis for ONR’s assessment and have therefore been used for the Step 2 assessment of the Rolls-Royce SMR design.
5. The International Atomic Energy Agency (IAEA) safety standards (ref. [20]) and nuclear security series (ref. [21]) are a cornerstone of the global nuclear safety and security regime. They provide a framework of fundamental principles, requirements and guidance. They are applicable, as relevant, throughout the entire lifetime of facilities and activities.
6. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [22]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [23]).
7. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and expanded on in the TAGs (ref. [12]). The TAGs provide the principal means for assessing the mechanical engineering aspects in practice.
   * 1. Safety Assessment Principles (SAPs)
8. The key SAPs (ref. [11]) applied within my assessment are:

* FP.6 (Prevention of accidents). This is relevant to claims that risks have been reduced ALARP.
* SC.4 (Safety case characteristics). This is relevant to claims that the Rolls-Royce SMR design has applied conservative design principles.
* EKP.1 (Inherent safety) to EKP.5 (Safety measures). This is relevant where claims are made regarding conservative design principles.
* ECS.3 (Codes and standards). This is relevant to the RP’s claims that relevant good practice has informed the Rolls-Royce SMR design.
* EQU.1 (Qualification procedures). This is relevant to the RP’s claims that it has a robust approach to verification and validation.
* EMT.1 (Identification of requirements). This is relevant to the RP’s EIMT claims.
* EAD.1 (Safe working life) and EAD.2 (Lifetime margins). These are relevant to ageing and degradation claims.
* ELO.1 (Access). This is relevant to through life access and egress claims.

1. A list of the SAPs used in this assessment is recorded in Appendix 1.
   * 1. Technical Assessment Guides (TAGs)
2. The following TAGs have been used as part of this assessment:

* NS-TAST-GD-005 - Regulating duties to reduce risks to ALARP (ref. [24])
* NS-TAST-GD-006 - Design Basis Analysis (re. [25])
* NS-TAST-GD-009 - Examination, Inspection, Maintenance and Testing of Items Important to Safety (ref. [26])
* NS-TAST-GD-022 - Ventilation (ref. [27])
* NS-TAST-GD-036 - Redundancy, Diversity, Segregation and Layout of Structures, Systems and Components (ref. [28])
* NS-TAST-GD-056 - Nuclear Lifting Operations (ref. [29])
* NS-TAST-GD-057 - Design Safety Assurance (ref. [30])
* NS-TAST-GD-067 - Pressure System Safety (ref. [31])
* NS-TAST-GD-094 - Categorisation of Safety Functions and Classification of Structures, Systems and Components (ref. [32])
* NS-TAST-GD-096 - Guidance on the Mechanics of Assessment (ref. [10])
  + 1. National and international standards and guidance

1. The following international standards and guidance have been used as part of this assessment:

* IAEA, SSR-2/1 Safety of Nuclear Power Plants: Design (ref. [33])
* IAEA, SSG-56 Design of Reactor Coolant System and Associated Systems for Nuclear Power Plants (ref. [34])
* IAEA, SSG-61 Format and Content of the Safety Analysis Report for Nuclear Power Plants (ref. [35])
* IAEA, SSG-67 Seismic Design for Nuclear Installations (ref. [36])
* IAEA, SSG-69 Equipment Qualification for Nuclear Installations (ref. [37])
* IAEA, SSG-70 Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants (ref. [38])
* IAEA, NS-G-1.13 Radiation Protection Aspects of Design for Nuclear Power Plants (ref. [39])
* IAEA, SRS-46 Assessment of Defence in Depth for Nuclear Power Plants (ref. [40])
  1. Integration with other assessment topics

1. I have worked closely with other topics as part of my mechanical engineering assessment. Similarly, other assessors sought input from my assessment. These interactions are key to the success of GDA to prevent or mitigate any gaps, duplications or inconsistencies in ONR’s assessment.
2. The key interactions with other topic areas were:

* Civil engineering. This related to the RP’s consideration of large item removal from the containment building.
* Internal hazards. This related to the RP’s approach to heating ventilation air conditioning (HVAC), mechanical handling and drop loads.
* Fault studies. This related to the RP’s approach to safety function categorisation and SSC classification.
* Nuclear site health and safety. This related to the RP’s application of principles of prevention at the early design stage.
* Structural integrity. This related to the RP’s approach to ageing and degradation management.
* Human factors. This related to the RP’s approach to HVAC main control room habitability.
* Electrical engineering. This related to the RP’s HVAC design and reactor coolant pump (RCP) variable frequency drive (VFD).
* Control and instrumentation. This related to the RP’s HVAC design.
* Radiation protection. This related to the RP’s use of cobalt-based materials.
* Project. This related to RO-RRSMR-001: Development of the generic E3S case.
  1. Use of technical support contractors

1. During Step 2 I have not engaged Technical Support Contractors (TSCs) to support my assessment of the mechanical engineering aspects of the Rolls-Royce SMR design.

# Requesting party’s submission

1. Rolls-Royce SMR Limited submitted a series of E3S chapters, or summary reports, and other supporting references, which outline the E3S case for the generic Rolls-Royce SMR design. This section presents a summary of the RP’s safety case for mechanical engineering. It also identifies the documents submitted by the RP which have formed the basis of my mechanical engineering assessment of the Rolls-Royce SMR.
   1. Summary of the Rolls-Royce SMR design
2. The generic Rolls-Royce SMR design is a three loop Pressurised Water Reactor (PWR) with a target electrical power output of 470 MWe (from a thermal power of 1,358 MWth) and a design life of 60 years for non-replaceable components.
3. The Rolls-Royce SMR design has been developed by the RP based upon well-established PWR technology, in use all over the world. Innovation comes in the form of its modular approach to construction which would see the majority of the power station built in factory conditions and assembled on-site.
4. The reactor itself is of a typical PWR design, including a steel Reactor Pressure Vessel (RPV) holding fuel assemblies, Steam Generators (SG), Reactor Coolant Pumps (RCP) and piping, all held within a steel containment vessel. The reactor is equipped with a number of supporting systems for normal operations and a range of safety measures are present in the design to provide cooling, control criticality and contain radioactivity under fault conditions. Passive safety features are preferred to active components, reflecting the RP’s design philosophy.
5. There are several novel aspects I consider relevant to my mechanical engineering assessment. These include:

* compact layout;
* modular design using a standardised “kit of parts”;
* reactor island aseismic bearing;
* passive safety system use; and
* innovative spent fuel route mechanical handling.
  1. E3S case approach and structure

1. Rolls-Royce SMR Limited has chosen to develop its cases in a holistic manner, as an Environment, Safety, Security and Safeguards (E3S) case. The overall objective for the E3S case is to demonstrate that the design will ‘protect people and the environment from harm’.
2. This means that, although the case made for each of the E3S purposes (i.e. environment, safety, security and safeguards) will inevitably be different at the top level, it will draw upon common evidence outputs (as well as other non-common outputs) to substantiate each of the purposes. This is claimed to offer benefits in terms of clarity, integration and understanding the impacts from any changes to the case.
3. The E3S case is being developed using a three tier hierarchy and incorporating a Claim, Argument and Evidence (CAE) structure with the highest-level claims being derived from the RP’s own E3S principles. The highest level of the three tiers is the RP’s Tier 1 E3S chapters, with the lower tiers providing more detailed arguments and evidence. This is illustrated in Figure 1.

****

Figure 1: Claim, Argument and Evidence (CAE) structure within the E3S hierarchy (ref. [1])

1. The structure of the E3S case largely aligns with the IAEA guidance for safety cases, SSG-61 (ref. [35]), supplemented to include UK specific expectations and expanded to include the other E3S purposes.
   1. Summary of the requesting party’s E3S case for mechanical engineering
2. The E3S case comprises 33 chapters. The RP’s fundamental E3S case objective is to demonstrate that its Rolls-Royce SMR design reduces overall safety risks ALARP (ref. [2]). No single E3S chapter explicitly captures the mechanical engineering safety case claims. Instead, these claims are spread across several chapters. These include:

* E3S case chapter 3 (ref. [2]). This contains the RP’s top level claim 3. This states that “suitable E3S design principles and associated methods, approaches, and requirements are established for the Rolls-Royce SMR to achieve the E3S fundamental objective.”
* E3S case chapter 5 (ref. [3]). This introduces the Rolls-Royce SMR design for its reactor coolant system (RCS). E3S chapter 6 (ref. [4]) introduces the Rolls-Royce SMR design for its safety systems. E3S chapter 9a (ref. [5]) introduces the Rolls-Royce SMR design for its auxiliary systems. These chapters claim that the associated SSCs are conservatively designed to reduce risks ALARP.
* E3S case chapter 15 (ref. [7]). This presents the RP’s safety analysis approach. It claims that its safety analysis has informed the design and demonstrates that adequate defence in depth (DiD) has been applied to deliver fundamental safety functions.
* E3S case chapter 16 (ref. [8]). This presents the RP’s operational limits and conditions. The RP claims that the operational limits and conditions for safe operations are defined by the Rolls-Royce SMR design and E3S analysis.
* E3S case chapter 24 (ref. [9]). This presents the RP’s ALARP summary. Here the RP claims that its Rolls-Royce SMR design reduces nuclear and nuclear site health and safety risks ALARP.
  1. Basis of assessment: requesting party’s documentation

1. The principal documents forming the basis of my mechanical engineering assessment of the E3S case are:

* E3S tier 1 chapters applicable to mechanical engineering (refs. [1], [2], [3], [4], [5], [6], [7], [8], [9])
* HVAC system design description (ref. [41])
* RCP system design description (ref. [42])
* RCP requirements specification (ref. [43]).
* The fault schedule (refs [44] and [45])
* Modular kit of parts strategy (ref. [46])
* Engineering management plan (ref. [47])
* Emergency core cooling (ECC) safety measure design description (ref. [48])
* Mechanical sequence diagrams (MSDs) (ref. [49])
* Architectural and layout summary report (ref. [50])
* EIMT strategy (ref. [51])
* ALARP summary report (ref. [52])
* Design definition for conveyance of fuel assemblies / internals within storage areas (ref. [53])
* Reactor island mechanical handling design summary report (ref. [54])

# ONR assessment

* 1. Assessment strategy

1. My Step 2 assessment plan (ref. [15]) outlines my targeted scope. It describes specific mechanical engineering themes. I sought to assess these themes in the RP’s E3S case by targeting SSCs important to nuclear safety. These themes are:

* Environment, Safety, Security and Safeguard Case Adequacy
* Design Parameters
* Passive Safety
* Modularisation
* Containment
* Defence in Depth and Redundancy, Diversity and Segregation
* Design Safety Assurance
* ALARP and Relevant Good Practice
* Categorisation of Safety Functions and Classification of Mechanical Structures, Systems and Components.
* Examination, Inspection, Maintenance and Testing
* Equipment Qualification

1. My assessment has been against the RP’s E3S case defined by its GDA design reference point one (DRP1) report (ref. [55]). The DRP1 report is consistent with the RP’s reference design 7 (RD7) engineering milestone, which represents a fixed design baseline. Hence, my SSC assessment sample is taken from submissions supplied against this baseline. The RP issued revised tier 1 chapters in May 2024. I have sampled these submissions and am satisfied that they reflect the Rolls-Royce SMR design that I have assessed.

## Assessment

### E3S case

1. My Step 2 plan (ref. [15]) included an adequacy assessment of the RP’s E3S case. Specifically, I sought to judge whether the generic E3S case:

* has appropriate structure; and
* is developing adequately during GDA.

1. I assessed the RP’s introduction to its E3S case (ref. [1]). I noted that the RP’s E3S case uses a hierarchical structure to support accessibility (see Figure 1 of this report). I am satisfied that the RP’s E3S case is logical and accessible. This is because it is arranged in a hierarchical structure that seeks to provide a claims, arguments and evidence trail. This aligns with my expectations against ONR SAP SC.2 (Safety case process outputs) and paragraphs 79 to 86 inclusive (ref. [11]).
2. To evaluate the RP’s E3S case development adequacy, I assessed several tier 1 chapters (see paragraph ‎35 of this report). I recognise that the RP is developing its E3S case and Rolls-Royce SMR design concurrently. I also note that the RP is working towards a concept design. Hence I consider it reasonable that the E3S case is not yet fully complete. During Step 2 I judge it appropriate that the RP provides adequate assurance that mechanical engineering design considerations have informed the facility layout. This is because mechanical engineering design rules should support demonstration that the concept design reduces nuclear and nuclear site health and safety risks ALARP. I judge that early consideration of layout supports the GDA objective to de-risk potential for future licensing activities. This is consistent with the stated objectives of a Step 2 assessment (GDA Guidance to Requesting Parties, Appendix 3 (ref. [18]) and aligns with my Step 2 expectations of ONR SAP SC.4 (Safety case characteristics) (ref. [60] [11]).
3. Hence, I targeted:

* The main control room and safeguard building HVAC SSCs. This is because its sizing may affect the layout of the Rolls-Royce SMR design, and it used what I consider a novel approach to deliver endurance period cooling. The HVAC design is not yet at a final concept stage and the RP has not yet completed its systematic hazard identification (see section ‎4.2.2.2 of this report). Hence, the RP has not yet completed its HVAC design requirements identification and provided assurance regarding the system sizing.
* Seismic performance requirements. The RP plans to determine some SSC seismic performance requirements during GDA Step 3 (see section ‎4.2.2.8 of this report). I consider that the development of seismic performance requirements may result in design and layout changes.
* The RP’s structured decomposition of requirements for traceability. I sought to identify the ‘golden thread’ through the its E3S case CAE. I noted that traceability between EIMT requirements and evidence is incomplete (see section ‎4.2.7.4 of this report). I consider that the development of EIMT requirements may result in design and layout changes.
* The RP’s nuclear site health and safety approach for mechanical engineering SSCs. During Step 2 the RP was developing its product safety review processes. This included a product safety review board (see paragraph ‎294‎ of this report).

1. Hence, I considered it unclear how the RP’s E3S case development adequately demonstrated that it has identified all significant design requirements appropriate during Step 2. This does not align with my expectations against ONR SAP SC.4 (Safety case characteristics) (ref. [11]).
2. Regulatory Observation RO-RRSMR-001 (ref. [56]) was raised by ONR to seek an E3S case development strategy, and improvements in its implementation. RO-RRSMR-001 has two actions:

* RO-RRSMR-001.A1 – E3S case strategy and forward plans. This sought evidence relating to how the E3S case and design development is linked.
* RO-RRSMR-001.A2 – E3S processes, governance and oversight. This sought evidence regarding E3S case organisational arrangements.

1. In response to RO-RRSMR-001, the RP made several submissions as per its resolution plan (ref. [57]). I sampled its nuclear assurance internal inspection (ref. [58]) and noted its key findings included that:

* there is not yet a defined, repeatable, systematic set of processes by which E3S principles and hazards become safety requirements in its requirements management system (DOORS); and
* the digital tools require improved E3S case requirements traceability.

1. An intervention was arranged to discuss RO-RRSMR-001 (ref. [59]). From a mechanical engineering perspective it was unclear:

* how the RP manages its E3S case development including its CAE traceability;
* when the RP will have a complete set of claims for its design; and
* how its project safety review board is used to:
  + ensure design decisions reduce risks ALARP; and
  + resolve conflicting nuclear safety requirements.

1. I conclude that the RP has:

* adequately demonstrated that its E3S case is logically structured and accessible.
* not yet adequately demonstrated that its E3S case has:
  + applied mechanical engineering design rules to identify safety parameters appropriate for Step 2 that may affect the Rolls-Royce SMR design layout.
  + a completed claims, arguments and evidence structure; and
  + adequate safety requirements traceability.

1. For Step 2 I am satisfied that these shortfalls can be resolved. This is because the RP has recognised the gap and is seeking to close RO-RRSMR-001 in accordance with its resolution plan (ref. [57]).
2. During Step 3 I will seek assurance that the E3S case is sufficiently complete and supports the closure of RO-RRSMR-001. I consider this a residual matter.

### Design parameters

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s E3S case design parameters.
2. My assessment sought to judge whether appropriate Step 2 design parameters, including limits and conditions, were identified and where relevant substantiated. This includes the architecture and key SSC sizing.
3. To judge the RP’s design parameter adequacy I assessed the E3S case Chapter 16: Operational Limits and Conditions (ref. [8]). I also targeted the following given their relevance to nuclear safety:

* heating, ventilation and air-conditioning (HVAC) systems;
* reactor coolant pump (RCP);
* reactor coolant pressure safety relief valves (SRV);
* control rod drive mechanisms (CRDMs);
* emergency core cooling (ECC) safety measure emergency blowdown valves (EBDVs);
* passive decay heat removal (PDHR) safety measure;
* main steam isolation valves (MSIVs); and
* seismic analysis and withstand.

#### Operational limits and conditions

1. I assessed the RP’s E3S Case Chapter 16: Operational Limits and Conditions (ref. [8]). This presents evidence to support its claim that the Rolls-Royce SMR design has adequate limits and conditions.
2. I noted that the RP:

* identifies appropriate limits and conditions RGP (section 16.0.4 of (ref. [8]));
* states that its design principles (ref. [60]) identify the requirement for operational limits and conditions; and
* states that development of limits and conditions is ongoing (section 16.2.3 of (ref. [8]).

1. I recognise that the Rolls-Royce SMR design and E3S case are being developed to a final concept. Hence, I am satisfied that not all limits and conditions are identified at this stage. For Step 2 I conclude that the RP has adequately demonstrated that its operational limits and conditions development approach considers RGP.
2. During Step 3 I will seek assurance that the RP has identified and adequately justified appropriate generic Rolls-Royce SMR design limits and conditions.

#### Heating Ventilation And Air-Conditioning design parameters

1. My HVAC system sample included SSCs that:

* are the principal means of delivering category A nuclear safety functions (nuclear safety class 1); and
* fulfil important confinement and cooling roles.

1. I assessed the HVAC system design description (ref. [41]). This considers the reactor island (RI) HVAC systems.
2. Within the HVAC system, I targeted the reactor island safeguard building and the control room sub-systems. This is because the:

* reactor island safeguard building has nuclear safety functions to maintain safety system SSCs at qualified temperatures (section 2.8 (ref. [41]); and
* control room has nuclear safety functions to enable human and SSC performance in several plant conditions (section 2.11 (ref. [41]).

1. For both targeted HVAC sub-systems I noted that several design attributes were incomplete. For example, the local cooling unit (LCU) load and airflow requirements for both targeted sub-systems were not defined. Because of the HVAC design immaturity and as there is no reference plant to provide assurance, I considered the basis for the HVAC capacity and sizing unclear.
2. I raised RQ-01272 (ref. [61]) to seek clarification regarding the RP’s HVAC E3S case maturity, development and performance requirements. The RP responded stating that its:

* HVAC design decision records (for example its HVAC design underpinning) are subject to review and are likely to result in changes;
* HVAC hazard identification remained outstanding; and
* requirements management tool (DOORS) is not populated with HVAC requirements.

1. I recognise that the RP intends to develop its HVAC design during GDA. Based on its design immaturity assessed during Step 2, I judge that the RP has not yet demonstrated that its HVAC design is adequate. This does not satisfy my Step 2 expectations against

* ONR SAP SC.4 (Safety case characteristics) (ref. [11]). This relates to identification of limits and conditions in the interests of safety.
* Ventilation TAG NS-TAST-GD-022 (paragraph 131 of (ref. [27]). This relates to HVAC system sizing to achieve the required airflow and depressions in normal, fault and accident situations.
* IAEA Safety of Nuclear power Plants: Design (SSR-2/1) (ref. [33]) regulation 73. This relates to HVAC system capacity to achieve safety claims.

1. For Step 2 I conclude that adequate HVAC design parameter demonstration remains outstanding.
2. During Step 3 I will seek assurance that the RP’s HVAC design and associated limits and conditions are adequate.

#### Reactor coolant pump design parameters

1. The reactor coolant pump (RCP) has a nuclear safety function to supply primary coolant flow in several plant conditions.
2. I assessed the RP’s RCP design parameters in its E3S case including:

* E3S case chapter 5 (section 5.5 of (ref. [3]));
* RCP system design definition (ref. [42]); and
* RCP requirements specification (ref. [43]).

1. I considered it unclear:

* how the RCP limits and conditions had been identified for normal, fault and accident conditions;
* whether the RCP limits and conditions were complete;
* how the RCP limits and conditions provided adequate margin to allow for uncertainties;
* why seismic requirements had not been identified; and
* which RCP seals had nuclear safety claims.

1. I raised RQ-01024 and RQ-01181 (ref. [61]) to seek clarification. The RP responded that:

* The fault schedule (ref. [44] [45]) is used to derive the RCP functional requirements as defined in its C3.2.2-3 engineering process.
* The RCP parameters relate to the thermal hydraulic duty for normal operation. This defines the minimum flow required to remove heat from the reactor core.
* The extent of seismic work undertaken to date is dependent upon perceived scale of impact on the SSC or the layout. RCP seismic performance clarifications (SPCs) are allocated in accordance with the RP’s seismic classification methodology (ref. [62]).
* RCP seals providing a nuclear safety function will be identified in future E3S case revisions.
* RCP design parameter configuration control is achieved through its requirements management system in accordance with:
  + process C3.1.1 (Define and manage requirements);
  + the configuration management plan [63]; and
  + change control during GDA (see RQ-01009 (ref. [61])).

1. I consider that the:

* Fault schedule (ref. [44] [45]) identifies fault and accident performance requirements. I note that the RCP duty requirements (best estimate flow rate and maximum pressure head) are included in the RCP requirements specification (see PT100-R-1251 and PT100-R-1909 of (ref. [43])). This aligns with my expectations against ONR SAP EHT.1 (Design) and EHT.2 (Coolant inventory and flow) (ref. [11]).
* RCP seismic performance requirements are not yet complete. This is because the:
  + horizontal peak ground acceleration (PGA of 0.3g in requirement PT100-R-1497 (ref. [43])) is referenced rather than the acceleration transferred to the RCP through the aseismic bearing.
  + Vertical PGA has not been specified (see PT100-R-1473 of the requirements specification (ref. [43])).

Note: The RP has recognised that it needs to specify and share these RCP seismic loads with the supplier (PT100-R-1497 (ref. [43])). For Step 2 this does not align with my expectations against ONR SAP EHA.9 (Earthquakes) (ref. [11]). Demonstration that specific seismic withstand has been achieved is not expected during Step 2. However, I do expect that adequate seismic provision is considered early in the design phase in accordance with RGP (section 4.2 of IAEA SSG-67 (ref. [36])).

1. Hence, I conclude that the RP has:

* adequately demonstrated its RCP design parameters:
  + link to the fault schedule; and
  + will be included in its requirements management system.
* not yet adequately demonstrated that its RCP design parameters include all those that may affect layout feasibility.

1. For Step 2 I am satisfied. This is because I consider that the RP has the opportunity to demonstrate its RCP design parameter adequacy during Step 3. This can be achieved through its RCS gated review process and supplier engagement.
2. During Step 3 I will seek assurance that the RP’s RCP design parameters:

* are suitable and sufficient against the reference design; and
* provide adequate margin to allow for uncertainties.

#### Primary circuit safety relief valve design parameters

1. The primary circuit SRVs:

* have a category A safety function and are nuclear safety class 1 SSCs; and
* fulfil an important confinement and overpressure relief role.

1. I used IAEA guidance to inform my judgement including:

* SSR2/1 – Safety of Nuclear power Plants: Design (ref. [33]). Requirement 15 expects SSC design limits and physical parameters to be specified for all operational states.
* SSG-56 – Design of Reactor Coolant System and Associated Systems for Nuclear Power Plants (ref. [34]). This expects SSC design limits and criteria to be linked to fuel criteria.

1. I raised RQ-01182 and RQ-01183 (ref. [61]) to seek clarification regarding the SRV design parameters and design inputs. There is OPEX of similar valve types using cobalt-based materials to improve the properties of valve seating faces. I therefore requested further information on whether the RP intends to use cobalt-based materials, as it is relevant when considered against the RP’s EIMT regime. This is because it effects personnel dose during EIMT activities in the vicinity of the SRVs and RCS.
2. The RP outlined its arrangements for controlling cobalt-based material use in the RCS. This is achieved through an entry in its requirement management database which states:

* that the use of cobalt-based material in the RCS must be justified by an E3S assessment; and
* endorsed by the Chief Engineer.

1. This satisfies my expectations against ONR SAP RP.7 (Hierarchy of control measures) (ref. [11]) for restricting exposure to radiation, and demonstrates the RP is controlling risks when developing its design parameters.
2. During Step 3, I will seek assurance that the RP’s requirement management process adequately considers EIMT dose uptake when assessing its cobalt based materials use. This aligns with Radiation Protection Aspects of Design for Nuclear Power Plants (NS-G-1.13) (ref. [39]). This expects the source terms to be minimised by selection of materials to reduce the nuclide concentration.
3. I assessed the RP’s SRV system design description (ref. [64]) and design requirements document (ref. [65]). I noted that the RP has:

* Referred to established codes and standards. This is evidenced through the RP’s responses to RQ-01182 and RQ-01183 (ref. [61]) These requested further information on design process arrangements for systems including classified pipework and valves. This satisfies my expectations against ONR SAP ECS.3 (Codes and standards) (ref. [11]).
* Recognised potential pressurised pipe systems design faults (for example flow flashing). The RP intends to analyse and remove faults from the design. This satisfies my expectations against ONR SAP FP.3 (Optimisation of protection) [11].
* An appropriate approach for reducing the nuclear safety class 1 and 2 pipework which may be embedded in concrete, walls or ceiling materials. This reduces areas inaccessible for through life EIMT. This aligns with my expectations against ONR SAP EMT.5 (Procedures) (ref. [11]).

1. For Step 2 I am satisfied that the RP has adequately demonstrated the fundamental SRV design parameters.

#### Control rod drive mechanism design parameters

1. The CRDMs:

* have a category A nuclear safety function and are a nuclear safety class 1 SSC; and
* fulfil an important reactivity control function.

1. I assessed the RP’s CRDM requirements specification (ref. [66]), component safety classification refinement (ref. [67]) and initial verification strategy (ref. [68]). I was satisfied that the RP adequately demonstrated:

* the basis of its design requirements;
* its safety categorisation of CRDM functions;
* a CRDM design which accommodates a boron free primary circuit chemistry;
* its approach to CRDM cooling; and
* use of appropriate standards to justify fundamental EIMT activities.

1. I judge that the RP’s approach aligns with my expectations against ONR SAPs (ref. [11]):

* ECS.3 (Codes and standards); and
* ERC.1 (Design and operation of reactors).

1. From the RP’s submissions I noted the CRDMs are:

* intended to be supplied by an established CRDM manufacturer with pedigree delivering to the United States nuclear industry.
* being designed against the American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section III 2021 for safety classified 1 components. This provided me with assurance that the design is aligned to recognised industry codes and standards. This satisfies my expectations against ONR SAP ECS.3 (Codes and standards) (ref. [11]).

1. I assessed the RP’s decision record regarding its use of cobalt-based materials in the CRDMs (ref. [69]). The significance of using a cobalt-based material is that when such a component containing this material is worn through operation, the released material can make its way into the primary coolant, where it can become radioactively ‘activated’. This can increase personnel dose during EIMT.
2. I raised RQ-01170 (ref. [61]) to clarify the RP’s decision-making for the use of cobalt-based materials . The RP stated that it has not yet tested known alternatives to cobalt-based materials. I liaised with ONR’s radiation protection specialism and concluded that I would require further engagement with the RP on its decision making process. The use of cobalt-based materials may be the correct design solution because of its material properties. However, I consider that the RP has not yet demonstrated that its use of cobalt-based materials reduces risks ALARP.
3. For Step 2 I am satisfied. This is because I consider the RP has opportunity to adequately demonstrate its decision-making process during Step 3.
4. During Step 3 I will seek assurance that the RP’s optioneering decision-making regarding the use of cobalt-based materials reduces risks ALARP.

#### Emergency core cooling safety measure emergency blowdown valves design parameters

1. The emergency core cooling (ECC) safety measure emergency blow down valves (EBDVs):

* are the principal means of delivering category A nuclear safety functions (nuclear safety class 1); and
* fulfil an important confinement and cooling role.

1. I consider that the EBDVs are a novel arrangements of existing valve technology. Figure 2 shows an EBDV arrangement schematic.

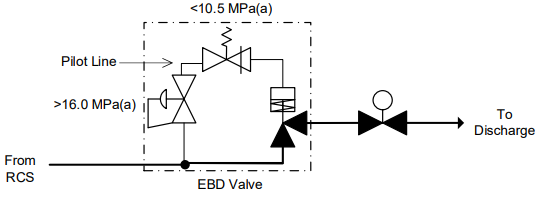


Figure 2: Emergency blowdown valve schematic (ref. [48])

1. The RP uses EBDVs during high pressure and low pressure ECC. The normally closed valve on the discharge line is:

* a multistage control valve (MSCV) on the high-pressure lines; and
* an automatic isolation valve (AIV) on the low-pressure lines.

1. Figure 3 shows the RP’s EBDV arrangement in its high (HP) and low-pressure (LP) automatic depressurisation system (ADS) lines.

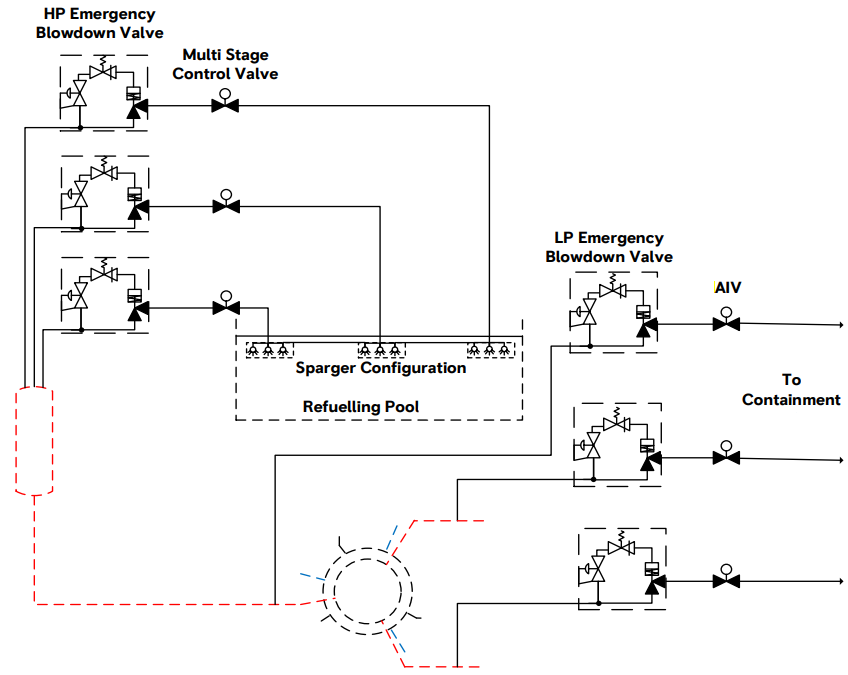
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Figure 3: Emergency blowdown valve arrangement in high and low pressure automatic depressurisation system lines (ref. [48])

1. I assessed the safety measure design description for the ECC (ref. [48]) and the ADS system design description (ref. [70]). I consider that these include appropriate EBDV limits and conditions including:

* high pressure set points;
* low pressure set points;
* high and low-pressure line sizes;
* MSCV opening times;
* EBDV leakage rates; and
* AIV opening parameters.

1. Hence, for its EBDVs during Step 2, I consider that the RP has adequately demonstrated:

* Appropriate pressure relief limits and conditions. This aligns with my expectations against:
  + ONR SAP EPS.3 (Pressure relief) (ref. [11]); and
  + ONR Pressure Systems Safety TAG NS-TAST-GD-067 (paragraph 51 (ref. [31])).
* How the EBDVs support fundamental safety functions. This aligns with my expectations against:
  + ONR SAP ERC.1 (Design and operation of reactors) (ref. [11]); and
  + IAEA SSG-70 Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants (ref. [38]) (section 4.3 and Appendix I.21).

1. For Step 2 I conclude that the RP has adequately demonstrated the ECC EBDV design parameter adequacy.

#### Passive decay heat removal safety measure design parameters

1. The PDHR safety measure is the principal means of delivering category B nuclear safety functions (nuclear safety class 2) and fulfils an important cooling role.
2. I used RGP to inform my judgements:

* ONR SAPs (ref. [11]):
  + SC.4 (Safety case documentation). This considers that a safety case should be accurate, objective and demonstrably complete for its intended purpose.
  + ERL.4 (Margins of conservatism). This considers ‘Where safety-related systems and/or other means are claimed to reduce the frequency of a fault sequence, the safety case should include a margin of conservatism to allow for uncertainties.’
  + EKP.5 (Safety measures). This considers that safety measures should be identified to deliver the required safety function(s).
* ONR TAG NS-TAST-GD-057 (Design Safety Assurance) (ref. [30]).

1. I raised RQ-01212 (ref. [61]) to obtain further information regarding the PDHR design parameters. This included the basis of its thermal performance. I sought clarification as to how the 18 MW heat removal value had been determined, and whether adequate margin is available to meet the system’s safety functional requirements should fault conditions make duty heat exchangers unavailable. The RP’s response provided me with assurance that:

* analytical methods were used when determining the PDHR performance requirements;
* these methods include a margin in addition to the base requirements; and
* design margins considered RGP and operating experience (OPEX).

1. For Step 2 I judge that the RP has adequately demonstrated appropriate PDHR design parameters.

#### Seismic analysis and withstand

1. I consider seismic analysis an important input for SSC design parameters. This is because an SSC’s seismic performance requirements can affect key safety functional requirements and layout feasibility.
2. The RP’s design incorporates an aseismic bearing supporting the reactor island (RI). The aseismic bearing is designed to reduce seismic accelerations transferred to the RI. Though these accelerations will be reduced, RI seismic analysis is required to demonstrate that SSCs on the RI can perform their design basis earthquake (DBE) safety functions.
3. The RP has produced a seismic performance classification methodology (ref. [62]). This document outlines how the RP intends to apply SSC seismic classifications. The document also outlines the RP’s design basis earthquake (DBE) assumptions (and more severe earthquakes) and assumed ground accelerations. The adequacy of assumptions underpinning these accelerations will be assessed by ONR hazard specialists. For Step 2 I am satisfied that the RP has in place a classification scheme that references RGP.
4. The RP has not yet adequately demonstrated how its SSCs have been designed to deliver their safety functions during a DBE. The RP states that seismic classification assessments will be developed at a later point. No schedule for this analysis was identified in the RP’s submissions. I judge this to be a gap against RGP expectations including:

* IAEA Seismic Design for Nuclear Installations SSG-67 (ref. [36]). This expects:
  + a seismic design process (section 3.5) that:
    - determines the preliminary design of structural elements based on codes and standards and providing adequate reinforcement detailing;
    - verifies that the seismic demand does not exceed the seismic capacity defined in the preliminary design and adjusts the design if necessary; and
    - assesses seismic margins.
  + that the layout of the installation should be established early in the design stage and should aim to achieve the most suitable seismic design solution (section 4.2).
* ONR SAP EHA.9 (Earthquakes) (ref. [11]). This expects SSCs to be designed and assessed for seismic requirements.

1. The design of several systems and layouts has been matured to a final concept. This is beyond the initial design concept and is maturing towards the detail design stage. Following this, designs can be released for manufacture. I recognise that seismic analysis, and subsequent design modifications, can be undertaken at any stage. However, RGP recognises that undertaking seismic analysis later in the development stage may have a significant effect on the design and layout.
2. I raised RQ-01273 (ref. [61]) to clarify:

* when the RP’s seismic analysis will inform the design;
* how it would enable meaningful assessment of the design and layout requirements during GDA; and
* how its approach will avoid fundamental design and layout revisions.

1. The RP responded that:

* mechanical SSCs will have seismic performance classifications (SPC) allocated in accordance with its performance classification methodology (ref. [62]) during Step 3;
* where seismic analysis is required to demonstrate that the SSC meets its SPC, this may be completed after GDA Step 3;
* where physical testing is required, this may be performed later but a confidence statement would be provided after GDA; and
* the extent of seismic work undertaken to date is dependent on the significance of the SSC loading or layout.

1. The RP outlined work that may be undertaken early in Step 3 to avoid fundamental design and layout revisions. This included:

* where loading is not expected to be significant then a judgement, argument and supporting evidence would be provided for an SSC using best available information;
* where loading is more significant, then more detailed calculations will be undertaken, for example, certain pipe systems and layouts; and
* supplier engagement and comparison with existing designs to provide confidence that seismic qualification can be achieved.

1. This approach may satisfy the overall scope of my assessment for GDA. However, a satisfactory outcome relies on how the RP intends to scope its seismic work, the type of seismic analysis and the outputs that come from the seismic analysis. Further engagement and discussion with the RP on these matters is required in Step 3.
2. My Step 2 judgement is that the RP’s approach does not align with my expectations against:

* IAEA SSG-67 (ref. [36]) section 4.2. This expects that “the layout of the installation should be established early in the design stage of the installation and should aim to achieve the most suitable solution for the seismic design”.
* ONR SAP SC.4 (Safety case characteristics) (ref. [11]). This expects that a safety case should provide sufficient information to demonstrate that engineering rules have been followed during the design process.

With reference to IAEA SSG-67 (ref. [36]) and ONR SAP EHA.9 (Earthquakes) (ref. [11]), I judge that the design process has not yet been adequately influenced by seismic design rules, and this is my expectation at this stage of the RP’s design and safety case development.

1. I have assessed the RP’s approach the seismic design. As discussed in paragraph ‎104 of this report, I am concerned that future seismic analysis has the potential to challenge design decision justifications made by the RP. However, I am satisfied that the RP has the opportunity to demonstrate the adequacy of its seismic design during Step 3.
2. During Step 3 I will seek assurance that the RP’s seismic approach, analysis and outputs are adequate.

#### Design parameter summary

1. I conclude that the RP has:

* adequately demonstrated:
  + how its limits and conditions development approach considers RGP; and
  + SSC manufacturer engagement.
* not yet adequately demonstrated that key SSC design parameters are considered in the design.

1. The demonstration that design parameters important to safety are adequately identified in the E3S case is a GDA expectation. I am satisfied that the RP has the opportunity to demonstrate that key SSC design parameters are identified during Step 3. This is because it is expected that key SSCs will have reached final concept maturity during Step 3.
2. During Step 3 I will seek assurance that the key Rolls-Royce SMR design parameters are complete. I consider this a residual matter.

### Passive safety

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s approach to passive safety.
2. My assessment sought to judge whether the RP’s mechanical engineering passive safety claims are supported by appropriate safety analysis, design substantiation and experimental work.
3. To judge the RP’s passive safety approach, I assessed the following SSCs given their relevance to nuclear safety:

* control rod drive mechanisms (CRDMs); and
* PDHR safety measure.

#### Control rod drive mechanism (CRDM)

1. The CRDMs:

* Are the principal means of delivering a nuclear safety function (category A and nuclear safety class 1).
* Control reactivity and are utilised as part of the means of delivery for the SCRAM safety measure. This is the primary means of delivering the control of reactivity (CoR) fundamental safety function (FSF).

1. To improve my understanding of the CRDM passive safety approach outlined in the RP’s submissions, a technical meeting was held with the RP (ref. [71]).
2. The CRDMs are passive because when the electrical supply to the CRDMs is interrupted a mechanism releases the control rods which then fall under gravity into the core. This reduces reactivity and the reactor is shut down. This aligns with my expectations against ONR SAPs (ref. [11]):

* EKP.2 (Fault tolerance);
* EKP.5 (Safety features) the hierarchy of safety characteristics outlined in paragraph 115; and
* EDR.1 (Failure to safety).

1. This mechanism for CoR and for shutting down a reactor is consistent with extensive OPEX and is the primary means of emergency CoR for the majority of civil nuclear reactors.
2. For my Step 2 fundamental assessment I consider that the RP has adequately demonstrated its CRDM mechanical engineering passive safety mechanism and approach aligns with RGP and is derived from OPEX.

#### Passive decay heat removal safety measure

1. The PDHR safety measure is the principal means of delivering category B nuclear safety functions (nuclear safety class 2) and fulfils an important cooling role.
2. I assessed the PDHR safety measure design description (ref. [72]) and requirements (ref. [73]). I engaged with the ONR fault studies and human and organisation factors specialists and raised RQ-01212 (ref. [61]) to obtain clarification regarding the RP’s passive systems approach. I sought to understand the design of the cross-connection pipework between the local ultimate heat sinks (LUHS). Specifically, why the cross-connection valves are manual and not operated automatically. I considered ONR SAP EKP.5 (Safety features) to be applicable (ref. [11]). This expects safety features to adopt a hierarchy of responses, with passive means being at the top of that design hierarchy.
3. The RP responded that it did not intend to specify automatic valves for its design because:

* Following a fault that places a demand on LUHS cooling, the valves only need to be opened after a minimum of 24 hours. This extends PDHR cooling to 72 hours in the unlikely event that only a single PDHR cooling train was successfully aligned.
* The valves are located outside of the containment structure and operator access is available.
* Manual valves remove some failure modes associated with electrically actuated valves. For example spurious C&I actuation.
* The detriments of manual valves are not evaluated to outweigh the benefits. For example, placing a demand on the operator post-fault and the need to ensure the valves remain accessible post-fault.

1. I considered the basis of the RP’s evaluation to be unclear. During Step 3 I intend to seek assurance that the RP’s approach to automatic valves in passive systems is appropriate.
2. I noted that the RP references IAEA TECDOC 1624 – Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants (ref. [74]). This was used to baseline the RP’s passive system definition and passive categorisation. I consider that the RP has made appropriate reference to RGP to inform its passive safety systems design.
3. For Step 2 I conclude that the RP has:

* adequately demonstrated:
  + passive safety RGP consideration; and
  + that it has designed the PDHR with the aim of delivering a passive safety response in its various modes in accordance with RGP.
* not yet adequately demonstrated its evaluation relating to manual or automatic valve selection.

1. For Step 2 I am satisfied. This is because I consider the RP has the opportunity to demonstrate its PDHR passive safety adequacy during Step 3.

#### Passive safety summary

1. For Step 2 I conclude that the RP has adequately demonstrated that its passive safety approach aligns with RGP.
2. During Step 3 I will seek assurance that the RP has adequately applied its passive safety design principles to its SSC designs.

### Modularisation approach

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s E3S case modularisation design approach.
2. My assessment sought to judge whether there are mechanical engineering related safety implications and whether these are adequately identified. Further modularisation assessment is included in the Step 2 project assessment summary (ref. [16]).
3. To assess the adequacy of the RP’s modularisation approach I sampled its:

* modular kit of parts strategy (ref. [46]);
* modularisation kit of parts barriers design definition (ref. [75]); and
* modularisation kit of parts primary structure standard frame design definition (ref. [76]).

1. I recognise that the RP has proposed a modular kit of parts (MkoP) approach to support manufacturing and build certainty. From its MkoP, the RP intends to undertake manufacture, assembly and testing in a controlled factory environment. These modules can then be delivered to site.
2. From my assessment I noted that:

* seismic requirements for mechanical SSCs located in modules or module clusters will be agreed during the next phase (section 4.11.17 of the MkoP barriers design definition (ref. [75])).
* mechanical handling and transport loads are not defined (section 4.14.2 of the MkoP barriers design definition (ref. [75])).
* EIMT requirements have not yet been identified (section 4.15.4 of the MkoP barriers design definition (ref. [75])).

1. I recognise that the modularisation design is being developed and transport is outside the GDA scope. Hence, I do not expect all design inputs to be identified during GDA. However, I consider it reasonable that transport, mechanical handling and seismic loads should inform the modularisation design of mechanical SSCs. This aligns with my expectations against ONR SAPs (ref. [11]):

* EMT.1 (Identification of requirements);
* EQU.1 (Qualification procedures) paragraph 175; and
* EHA.9 (Earthquakes) paragraph 255.

#### Modularisation approach summary

1. I conclude that the RP’s E3S case has:

* adequately demonstrated:
  + its modularisation approach;
  + that its modularisation strategy is developing.
* not yet shown how its modularisation approach considers mechanical SSC:
  + seismic requirements;
  + mechanical handling and transport loading; and
  + EIMT requirements.

1. For Step 2 I am satisfied. This is because I consider that the RP has the opportunity to demonstrate that mechanical engineering modularisation design parameters are identified during Step 3 as its modularisation design matures.
2. Hence, during Step 3, I will seek assurance that the RP’s E3S case has adequately considered mechanical SSC modularisation:

* seismic requirements;
* transport loads; and
* EIMT requirements.

1. I consider these residual matters.

### Containment

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s approach to containment.
2. My assessment sought to judge whether the safety requirements for containment penetrations have been identified, justified and could be achieved by the Rolls-Royce SMR design.
3. To assess the adequacy of the RP’s approach to containment I sampled the following given their relevance to nuclear safety:

* HVAC system primary containment design; and
* containment equipment access hatch (EAH).

#### Heating ventilation and air conditioning

1. The RP’s HVAC primary containment system has important safety functions during several plant modes. I assessed the HVAC system design description (ref. [41]) to identify HVAC primary containment safety claims.
2. I noted that several primary containment HVAC system key design parameters remain outstanding (table 3 of (ref. [41])). The RP also states that its supporting calculations relate to a previous RI layout (section 8.3.2 of (ref. ( [41])). I consider that this demonstrates that the RP’s primary containment HVAC design is immature when assessed against ONR SAP ECV.10 (Ventilation system safety functions) (ref. [11]) expectations.
3. I raised RQ-01272 (ref. [61]) to seek clarification. The RP responded that its HVAC hazard identification is outstanding and that its requirements management tool (DOORS) is not populated with HVAC requirements.
4. I recognise that the RP intends to develop its containment HVAC design during GDA. This design will be reflected in its E3S case scheduled for issue to ONR during Step 3. Hence, for Step 2, I conclude that the RP has not yet demonstrated that its primary containment HVAC design is adequate. I am satisfied that the RP can address its HVAC primary containment design gaps during GDA.
5. During Step 3 I will seek assurance that the RP’s HVAC primary containment design is adequate.

#### Containment equipment access hatch

1. The containment equipment access hatch (EAH) is part of the RP’s primary containment. The EAH is a significant penetration and hence a potential weakness in the primary containment civil structure. The EAH has a category A safety function and nuclear safety class 1 designation.
2. I assessed the RP’s containment system design description (ref. [77]) and safety measure design description (ref. [78]). I noted that the EAH:

* has an elastomer seal;
* uses a double seal arrangement to allow interspace testing;
* has a hinged opening;
* leak testing is expected to be conducted after each outage; and
* is being designed to nuclear codes as it forms part of the containment vessel.

1. I consider that this aligns with my expectations against ONR SAPs (ref. [11]):

* ECS.3 (Codes and standards);
* ECV.3 (Means of confinement); and
* EMT.6 (Reliability claims).

1. I did not identify EAH seal EIMT considerations including:

* Ageing and degradation mechanisms. This does not align with my expectations against ONR SAP EAD.1 (Safe working life) (ref. [11]).
* Replacement periods. This does not align with my expectations against ONR SAP EMT.1 (identification of requirements) (ref. [11]).

1. I consider that this is consistent with the immaturity of the RP’s EIMT requirements development. The RP stated that its through life activity (TLA) modules are scheduled for completion in Step 3 and beyond. My EIMT assessment can be found in section ‎4.2.10 of this report.

#### Containment summary

1. I conclude that the RP has:

* adequately demonstrated consideration of EAH and primary containment building leak testing.
* not yet adequately demonstrated:
  + the HVAC primary containment design; and
  + EAH containment EIMT requirements.

1. For Step 2 I am satisfied. This is because I consider that during Step 3, the RP has the opportunity to demonstrate that its mechanical engineering containment requirements are adequately identified and implemented.
2. During Step 3 I will seek assurance that the RP’s mechanical engineering containment design requirements are adequately identified and implemented. I consider this a residual matter.

### Defence in depth and redundancy, diversity, segregation and layout

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s approach to defence in depth (DiD), redundancy, diversity, segregation and layout (RDS&L).
2. My assessment sought to judge whether:

* adequate safety function DiD is provided; and
* the generic design approach incorporates adequate RDS&L.

#### Defence in depth approach

1. To assess the RP’s DiD approach I assessed its E3S case objectives and design rules for SSCs (ref. [2]). I noted that the RP claims it applies DiD principles against postulated initiating events (PIEs) by providing practicable independent measures. The RP presents five independent levels of DiD.
2. I consider that the RP’s DiD framework aligns with my expectations against:

* ONR SAP EKP.3 (Defence in depth) (ref. [15]); and
* IAEA SRS-46 Assessment of Defence in Depth for Nuclear Power Plants (ref. [40]).

1. I assessed the RP’s approach to level 1 DiD. This is because international consensus expects adequate DiD should prevent faults from occurring in the first instance (ONR SAP EKP.3 (Defence in depth) paragraph 149 (ref. [11])). I noted its claimed approach is that:

* conservative design is applied;
* SSCs with nuclear safety class 1 and 2 are designed to nuclear specific codes and standards where available; and
* SSCs with nuclear safety class 3 should be designed to nuclear specific or appropriate industrial code and standards.

1. I am satisfied that for Step 2, the RP’s level 1 DiD approach aligns with my expectations against:

* ONR SAP ECS.3 (Codes and standards) (ref. [11]); and
* ONR TAG NS-TAST-GD-094 (ref. [32]) annex 2.

1. Hence, for my Step 2 assessment I judge that the RP has adequately demonstrated that its DiD approach aligns with RGP.
2. During Step 3 I will seek assurance that SSCs claimed as providing different DiD levels are independent and designed to appropriate codes and standards.

#### Redundancy, diversity, segregation and layout

1. To assess the adequacy of the RP’s RDS&L approach I assessed the RP’s design rules. I also targeted the following systems given their nuclear safety importance:

* reactor coolant pressure safety relief valves (SRVs); and
* PDHR safety measure.

#### Redundancy, diversity, segregation and layout design rules

1. To assess the RP’s RDS&L approach I assessed its E3S case objectives and design rules (ref. [2]) and E3S design principles (ref. [60]). I noted that its nuclear safety class 1 SSCs are claimed to be designed with:

* common cause failure (CCF) resilience;
* single failure criterion resilience;
* redundancy when in the worst permitted plant configuration during outages; and
* on-site essential service diversity.

1. This aligns with my expectations against ONR SAPs (ref. [11]):

* EDR.2 (Redundancy, diversity and segregation);
* EDR.3 (Common cause failure); and
* EDR.4 (Single failure criterion).

1. I note that the RP does not apply the same design principles for its nuclear safety class 2 SSCs. The ONR fault studies specialists have identified this may result in a potential shortfall against RGP (ref. [79]). This is because the RP’s approach does not require its nuclear safety class 2 SSCs to accommodate the worst normally permitted configuration for EIMT. During Step 3 I will seek assurance that application of the RP’s design rules (ref. [2]) and design principles (ref. [60]) results in adequate mechanical engineering SSC redundancy and diversity.

#### Reactor coolant pressure safety relief valves

1. I assessed the RP’s RCS SRV design definition (ref. [64]) and design requirements documents (ref. [80]). I considered the RP’s application of single failure criterion, common cause failure and redundancy to be unclear. I raised RQ-01182 and RQ-01183 (ref. [61]) to seek clarification.
2. The RP stated that:

* its RDS&L approach is described in its E3S objectives and design rules (ref. [2]) and its E3S design principles (ref. [60]);
* risks associated with common cause failure and the necessary mitigations have not yet been analysed;
* it intends to undertake this work following component selection; and
* common cause failure analysis may influence the RCS SRVs design requirements.

1. ONR fault studies specialists have assessed the fundamental adequacy of the RP’s approach and I engaged with them on this matter. With the exception of the matter discussed in paragraph ‎169, for Step 2 I judge that the RP’s fundamental approach aligns with my expectations against ONR SAPs (ref. [15]):

* EDR.2 (Redundancy, diversity and segregation);
* EDR.3 (Common cause failure); and
* EDR.4 (Single failure criterion).

1. During Step 3 I will seek assurance that RDS&L principles are adequately applied to mechanical SSCs (for example, the SRVs) once the RP’s common cause failure analysis is completed.

#### Passive decay heat removal safety measure

1. The PDHR safety measure is the principal means of delivering category B nuclear safety functions (nuclear safety class 2) and fulfils an important cooling role.
2. Supported by ONR fault studies specialists (ref. [81]), I consider that the PDHR and ECC operational hierarchy is as follows:

* The PDHR:
  + is the first protective safety function for frequent faults; and
  + provides a significant role in delivering heat removal for control of fuel temperature (CoFT) during fault conditions.
* The ECC:
  + is the second protective safety function for frequent faults;
  + provides the principal means of delivering heat removal safety function for CoFT during fault conditions; and
  + provides the first protective safety function for loss of coolant accidents providing the principal means of delivering heat removal for CoFT.

1. I considered the RP’s PDHR and ECC segregation approach to be unclear. This is because the PDHR and the ECC have common components. Segregation between independent systems to reduce common cause failure likelihood is an expectation of IAEA SSG-56 – Design of Reactor Coolant System and Associated Systems for Nuclear Power Plants (ref. [53]).
2. I raised RQ-01212 (ref. [61]) to seek clarification regarding the RP’s PDHR and ECC segregation approach. The RP stated that:

* Segregation between redundant SSC trains is provided.
* For each train there are shared SSCs where an alternative is not practicable. For example a single local ultimate heat sink (LUHS) is shared by a single train of ECC and PDHR safety measures.
* No design basis faults have been identified that can disable either or both the PDHR and the ECC in their entirety.
* The overall layout and internal hazard assessments are being developed.
* Segregation requirements and hazard prevention/protection design will be confirmed as the layout matures.

1. I assessed the RP’s PDHR safety measure design description (ref. [72]) and noted that the:

* rationale for sharing SSCs between the PDHR and ECC is outlined; and
* reference to a decision file supporting architecture decisions is provided (section 8.9.6 of (ref. [72])).

1. I consider that the RP’s rationale demonstrated adequate redundancy, diversity and segregation consideration in its architecture design at a system level. For my Step 2 assessment this aligns with my expectations against:

* ONR SAP EDR.2 (Redundancy, diversity and segregation) (ref. [11]); and
* ONR TAG NS-TAST-GD-036 – Redundancy, Diversity, Segregation and Layout of Structures, Systems and Components (ref. [28]).

1. During Step 3 I will seek assurance that:

* the RP’s application of RDS&L is adequate at a component level; and
* its shared architecture arrangement is adequate from a mechanical engineering perspective.

#### Defence in depth, redundancy, diversity, segregation and layout summary

1. I conclude that the RP has:

* adequately demonstrated that its DiD approach aligns with RGP and has given some consideration to CCF and SFC; and
* not yet shown how CCF and SFC risks have been addressed in its RCS safety relief valves design.

1. For Step 2 I am satisfied. This is because I consider that during Step 3, the RP has the opportunity to demonstrate adequacy of its DiD application, CCF and SFC implementation.
2. During Step 3 I will seek assurance that the RP’s mechanical engineering DiD, CCF and SFC approach is adequately implemented. I consider this a residual matter.

### Design assurance

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s design assurance approach.
2. My assessment sought to judge whether the RP’s design safety assurance arrangements ensure safety principles are integral to the generic design. Hence, to assess the RP’s design assurance approach I assessed the RP’s engineering management plan (ref. [47]). I also assessed the following SSCs given their nuclear safety relevance:

* reactor coolant pump (RCP);
* emergency blowdown valve (EBDV); and
* mechanical handling.

#### Design assurance arrangements

1. The RP is developing its E3S case concurrently with its design (E3S chapter 1: Introduction (ref. [1]) section 1.1.3). I assessed the RP’s engineering management Plan (ref. [47]) to assess its design development configuration control. This is because design configuration control supports adequate design assurance (ONR Design Safety Assurance TAG NS-TAST-GD-057 (ref. [30]) paragraph 63).
2. The engineering management plan (ref. [47]) describes the RP’s approach to:

* configuration management (section 6.13);
* engineering change control (section 6.15); and
* engineering management processes (section 11).

1. The RP uses a gated design review process. I assessed the RP’s engineering management design definition review (DR) process (C3.2.1-2 section 2.7.1 of (ref. [47])) to inform my judgement. I noted that the gated reviews align with key design quality hold points including:

* preliminary concept definition (PCD); and
* full concept definition (FCD).

1. The Chief Engineer (or a delegate) chairs the DR and assesses and confirms design definition maturity. Figure 4 shows the RP’s DR gates mapped against SSC maturity.

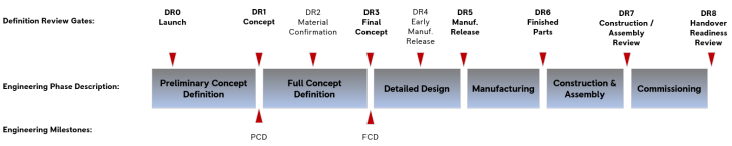
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Figure 4: Design reference gates and structure, system and component maturity (ref. [47])

1. I am satisfied that the RP’s gated review approach supports an iterative design process with quality hold points. For Step 2, this satisfies ONR’s design safety assurance TAG NS-TAST-GD-057 (ref. [30]) paragraph 57 and 63.
2. I noted that the RP’s design process includes optioneering (process C3.2.2-2). Process C3.2.2-2 is described in the RP’s key structural integrity processes for GDA (ref. [82]). It is stated that this process is applied for all key design decisions. Concept assessment is part of this optioneering. Figure 5 shows the RP’s concept assessment flow diagram.

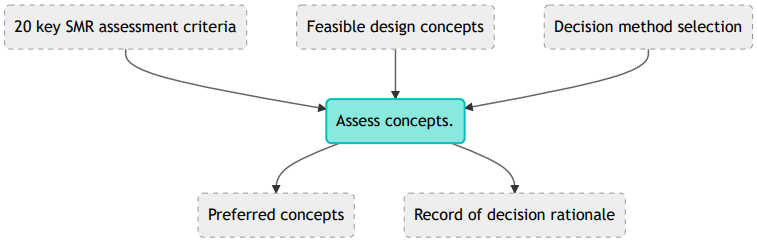


Figure 5: Optioneering process concept assessment flow diagram (ref. [82]) I sampled the RP’s ‘20 key assessment criteria’ identified in Figure 8

1. I assessed the RP’s 20 key assessment criteria identified in Figure 5. This is shown as an input to assessing concepts. These 20 key assessment criteria are found in the RP’s E3S case Chapter 1 (section 1.3.2 of (ref. [1])) and its decision record template (TS-DD-02) (ref. [83]). The criteria relate to:

* Safety
* Environment
* Security
* Cost
* Programme
* Market

1. Weighting for the key assessment criteria is discussed in the RP’s key objectives and assessment criteria (ref. [84]). I considered it unclear how the assessment criteria weighting demonstrated appropriate priority was given to safety. This is because I considered the aggregation of cost, programme and market factors versus safety to be in the order of 3:1. This does not align with my expectations against appendix 3 of ONR TAG NS-TAST-GD-005 – Regulating duties to reduce risks to ALARP (ref. [24]). This outlines optioneering expectations including that:

* priority should be given to achieving an overall balance of risk that is ALARP; and
* evaluation should prioritise options that minimise risk over life activities.

1. Supported by ONR’s nuclear site health and safety inspector I raised RQ-01152 (ref. [61]) to seek clarification. The RP responded stating that:

* its weighting for safety related questions were at the highest level;
* the scoring informs the overall decision such that unsafe options will be rejected; and
* further development of its processes relating to nuclear site health and safety were being implemented.

1. I recognise that the RP’s weighting, applied to the four safety related assessment criteria, is at the highest level. While I note that the 20 key assessment criteria output forms only part of its option decision making process, in my opinion this does not demonstrate that safety is prioritised relative to non-safety related criteria. This is because I consider the aggregation of cost, programme and market factors versus safety factors to be in the order of 3:1. This may lead to optioneering outcomes that do not reduce safety risks ALARP. I judge that this is a shortfall against Appendix 3 of TAG NS-TAST-GD-005 (ref. [24]).
2. I conclude that the RP has:

* adequately demonstrated that its engineering management plan and supporting arrangements describes its:
  + design review gated process; and
  + optioneering approach.
* not yet shown that its optioneering methodology, when applied, will lead to outcomes that appropriately prioritise safety.

1. For Step 2 I am satisfied. This is because during Step 3 the RP has the opportunity to adequately demonstrate its design optioneering process reduces risks ALARP.
2. During Step 3 I will seek assurance that the RP’s optioneering outcomes adequately demonstrate safety is prioritised and reduce mechanical engineering risks ALARP.

#### Reactor coolant pump

1. I assessed the RCP optioneering decision making process (ref. [85]). This was to inform my judgement regarding the application of the RP’s design assurance arrangements. This is because the RP’s RCP design includes an underslung arrangement with the RCP directly coupled to its related steam generator (SG).
2. I considered it unclear how:

* The optioneering scoring allocation is appropriate. This is because there were several examples where scoring allocated to alternative RCP configurations was not adequately justified. This does not align with my expectations against ONR TAG NS-TAST-GD-005 – Regulating duties to reduce risks to ALARP (Appendix 3, section 2 of (ref. [24])).
* Radiological safety has informed the optioneering output. This is because no source term had been provided to enable a meaningful comparison of personnel dose during EIMT. This does not align with my expectations against ONR SAP RP.7 (Hierarchy of control measures) (ref. [11]).
* OPEX had informed option selection. This is because I noted there to be a wide variation on available OPEX between options but all options were allocated the same score.

1. I raised RQ-01152 (ref. [61]) to seek clarification. The RP responded stating that:

* Mechanical handling complexities have been considered.
* Scoring allocation justification will be provided in a subsequent RCP optioneering decision record.
* Source term data was not available when the optioneering was completed and so was not included.
* The in-containment layout will need to be finalised to assess RCP EMIT dose.
* OPEX from hermetically sealed pumps is limited with one significant failure related to an AP1000 pump. The root cause is unclear although it is considered to relate to motor failure.

1. For Step 2 I conclude that the RP has not yet shown that its RCP optioneering output reduces risks ALARP. This is because it has not yet adequately demonstrated how:

* Radiation dose implications informed the optioneering. This is because it is unclear how the extant design reduces dose risks ALARP during operations and EIMT. This does not align with my expectations against ONR SAP RP.7 (Hierarchy of control measures) (ref. [11]).
* EIMT activities have informed the selected option. This is because the RCP EIMT activities have not yet been defined in the E3S case. This does not align with my expectations against ONR SAP EMT.1 (Identification of requirements) [11]. See section ‎4.2.10.2 of this report for RCP EIMT assessment.
* Scoring applied to various RCP options is justified. This does not align with my expectations against design assurance expectations in ONR TAG NS-TAST-GD-057 – Design Safety Assurance (paragraph 35 of (ref. [30])).

1. For Step 2 I am satisfied. This is because I consider that the RP has the opportunity during Step 3 to show that its RCP optioneering outcome reduces risks ALARP.
2. During Step 3, through its RCS gated review , I will seek assurance that the RCP optioneering process adequately considers:

* through life EIMT radiation dose implications;
* EIMT nuclear site health & safety; and
* appropriate scoring of all options.

1. I assessed the RCP requirements specification (ref. [43]) and noted that the RCP uses a variable frequency drive (VFD) during start up and shutdown. It was unclear whether all RCP VFD faults (for example overspeed) had been considered in the E3S case.
2. I collaborated with my electrical engineering colleague to raise RQ-01207 (ref. [61]) to clarify the RP’s RCP VFD E3S case claims. The RP responded stating that:

* the RCP VFD is intended to operate during modes 4a and 4b (Start up and shutdown);
* analysis has not been undertaken to identify the potential impact that a failure of the VFD may have on plant safety; and
* supplier input has enabled population of its requirements management in relation to the VFD.

1. I consider that the use of VFDs presents additional failure modes that have not yet been identified and mitigated. This does not align with my expectations against ONR SAP EHT.2 (Coolant inventory and flow) and EHT.4 (Failure of heat transport system) (ref. [11]).
2. For Step 2 I am satisfied. This is because during Step 3 the RP has the opportunity to show that its RCP system design review includes VFD:

* ES3 case requirements;
* interdependencies; and
* interdependency risk and mitigation plans.

#### Emergency blowdown valve

1. I targeted the EBDV to inform my Step 2 design safety assurance assessment. This is because the RP has designed the EBDV using existing valve technology applied in a novel arrangement. The EBDV schematic is shown in Figure 2 of this report.
2. The RP’s E3S case states that the EBDV operation requires continued leakage past the valve seat or a dedicated leak path to activate (section 3.2.12 of its safety measure design description for the emergency core cooling (ref. [48])). This is because the EBDV will open (to relieve pressure) when the relief valve spring overcomes the force generated by the fluid pressure in the pilot line.
3. I consider that the EBDV operational basis is unclear. This is because the E3S case does not describe how the pilot line pressure will be relieved to enable RCS depressurisation. This does not align with my expectations against ONR SAP EPS.4 (Overpressure protection) paragraph 277 (ref. [11]). This expects that where a combination of relief valves and an active protection system are used, then the safety case for the whole system is required.
4. For Step 2 I am satisfied. This is because during Step 3 the RP has the opportunity to implement its design safety assurance arrangements and show how the EBDV intent is achieved.
5. During Step 3 I will seek assurance that the RPs EBDV operational basis is adequate.

#### Mechanical handling

1. I targeted the E3S case reactor island mechanical handling approach to inform my judgement of the RP’s application of design assurance. This is because I consider that the reactor island mechanical handling solutions may affect the layout.
2. I assessed the RP’s mechanical sequence diagrams (MSDs) (ref. [49]) and noted personnel access provisions were unclear. I considered it unclear how access for mechanical handling operations and EIMT had been considered. I consider this a gap against:

* My expectations of ONR SAPs (ref. [11]):
  + EMT.1 (Identification of requirements). This expects EIMT activities to be identified in the E3S case.
  + ELO.1(Access). This expects the design and layout should facilitate access for necessary activities and minimise adverse interactions.
* Lifting Operations and Lifting Equipment Regulation guidance (ref. [86]). This expects:
  + means of access to lifting equipment (paragraph 65 of (ref. [86]));
  + loads should not be carried or suspended over areas occupied by people (paragraphs 159 and 230 of (ref. [86]) and associated guidance);
  + that the risk of clashes between lifting equipment and other objects to be minimised ((paragraph 265 and 266 of (ref. [86]) and associated guidance); and
  + design for EIMT (regulation 9 of (ref. [86]) and supporting guidance).

1. I raised RQ-01271 (ref. [61]) to seek clarification. The RP responded that the:

* architectural and layout summary report (ref. [50]) explains how access and egress is being considered;
* personnel requirements are provided in its target audience description (TAD) (ref. [87]); and
* EIMT strategy (ref. [51]) refers to through life activities (TLAs).

1. I assessed these submissions and noted that the:

* Architectural and layout summary report (ref. [50]) explains that the access strategy is being developed to comply with expectations for EIMT, operations and human factors. In my opinion this does not yet adequately demonstrate that the layout has reduced mechanical handling risks ALARP.
* TAD (ref. [87]) provides generic guidance relating to human performance (for example manual handling limits) and access and egress. In my opinion this does not yet demonstrate how these considerations have been incorporated into the layout.
* EMIT strategy (ref. [51]) does not yet adequately demonstrate that personnel EIMT requirements have been considered and implemented. This is because it provides the RP’s strategic approach but does not demonstrate how specific EIMT requirements will be achieved.

1. For Step 2 I conclude that the RP has not yet adequately demonstrated that its mechanical handling design has been informed by through-life personnel access and egress requirements. This includes operations and EIMT activities. This does not align with my Step 2 expectations against ONR SAPs (ref. [11]):
   * EMT.1 (Identification of requirements). This expects EIMT activities to be identified in the E3S case.
   * ELO.1(Access). This expects the design and layout should facilitate access for necessary activities and minimise adverse interactions.
2. I am content that the RP can address these gaps during GDA as its mechanical handling design progresses.
3. During Step 3 I will seek assurance that application of the RP’s design arrangements results in adequate personnel provision for mechanical handling operations and EIMT.

#### Design assurance summary

1. I conclude that the RP has:

* adequately demonstrated that its engineering management plan includes:
  + early design optioneering; and
  + gated design reviews.
* not yet adequately demonstrated:
  + that its optioneering methodology prioritises safety;
  + how its RCP optioneering outcome reduces risks ALARP;
  + how its gated review identifies and manages E3S case requirement interdependencies; and
  + how its layout achieves adequate mechanical handling, personnel access for operations and EIMT.

1. For Step 2 I am satisfied that the RP can address these design assurance shortfalls as its design matures during Step 3.
2. During Step 3 I will seek assurance that the RP’s design assurance application reduces operational and EIMT risks ALARP. I consider this a residual matter.

### Reducing risks to ALARP and use of relevant good practice

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s approach to reducing risks As Low As Reasonably Practicable (ALARP).
2. My assessment sought to judge whether the generic design has applied relevant good practice (RGP) towards reducing overall risks ALARP.
3. To assess the RP’s approach to reducing risks ALARP, I assessed the RP’s ALARP summary report (ref. [52]) and its mechanical handling approach. This is because I consider that the mechanical handling approach can affect nuclear safety.

#### ALARP summary report

1. The RP’s ALARP summary report (ref. [52]) explains the RP’s approach to reducing risks ALARP. I note that this report:

* Provides a summary of the design concept optioneering undertaken.
* References supporting design decisions records.
* Recognises that additional design refinements will be considered to demonstrate risks are reduced ALARP. For example, the RP states that whilst its MSIV reflects RGP and OPEX, it will consider whether introducing redundancy improves plant safety.
* References the use of ice stores as the selected means to supply chilled water to its HVAC nuclear safety class 1 cooling during accident conditions.

1. I assessed the RP’s selection of ice stores for its HVAC cooling. This is because I considered it a novel design. Through the course of my assessment, I identified an inconsistency in the RP’s approach for design decision-making. This is because:

* the RD7 HVAC submissions and the ALARP summary report (ref. [52]) state that ice stores will provide cooling to essential systems during challenging plant conditions; and
* during a HVAC technical meeting (ref. [88]) the RP informed me that it had reconsidered this design decision; this was confirmed through RQ-01272 (ref. [61]).

1. The RP’s rationale for this change to its stated ALARP position was a result of a design decision review. This undermined my confidence in the adequacy of the RP’s design decision making process and its demonstration that risks are reduced ALARP.
2. I consider that the inconsistency between the RP’s RQ response and its tier 1 HVAC submissions may be the result of the HVAC design maturity.
3. In Step 3 I will engage with the RP further to obtain assurance around decision making as it relates to optioneering and its design decision making arrangements.
4. For Step 2, I judge that the ALARP summary report demonstrates that the RP recognises that its design requires further iteration to show risks have been reduced ALARP. This aligns with ONR:

* TAG NS-TAST-GD-005 – Regulating duties to reduce risks ALARP (Appendix 3 paragraph 7 of (ref. [24])); and
* SAP MS.3 (Decision making) (ref. [11]). This states decisions made at all levels in the organisation affecting safety should be informed, rational, objective, transparent and prudent.

1. During Step 3 I will seek assurance that the RP’s optioneering output and decision-making process reduces risks ALARP.

#### Mechanical handling

1. I assessed the RP’s approach to reducing nuclear lifting risks ALARP. I considered its approach to:

* reducing nuclear lifts; and
* steam generator (SG) removal.

#### Reducing nuclear lifts

1. I sampled the RP’s spent fuel cask conveying system. This is because I considered that fuel route operations can influence the:

* fundamental layout; and
* mechanical handling requirements.

1. I assessed the RP’s design definition for conveyance of fuel assemblies / internals within storage areas (ref. [53]) and its ALARP summary report (ref. [89]).
2. I noted that the RP:

* Considered several cask movement options (section 3.10.6 of the ALARP summary report (ref. [89]));
* sought to avoid complex lifting solutions;
* sought to eliminate significant cask lifts; and
* recognises other hazards will be introduced by adopting ‘zero-lift’ cask handling (i.e. requirement for cask pit flooding).

1. For the spent fuel cask movements, the RP has:

* considered options for reducing cask loading and handling operations. This aligns with my expectations against ONR:
  + SAP EKP.1 (Inherent safety) (ref. [11]). This expects an inherently safe design that avoids radiological hazards.
  + TAG NS-TAST-GD-056 – Nuclear Lifting Operations (ref. [29]). This expects a safety case to demonstrate nuclear lifting risks have been reduced ALARP.
* sought to reduce cask lifting activities. This aligns with ONR SAPs (ref. [11]):
  + EKP.1 (Inherent safety).
  + ELO.4 (Minimisation of the effects of incidents). This expects a facility’s layout to minimise the effect of faults and accidents.
* Not yet demonstrated how its cask movement solution has considered personnel access, egress and nuclear site health and safety.

1. Based on my Step 2 spent fuel cask conveying system assessment, I am satisfied that the RP has considered reducing spent fuel cask nuclear lifts. Nuclear site health and safety aspects are assessed in section ‎4.2.7.4 of this report.
2. During Step 3 I will seek assurance that the fuel route and cask loading mechanical handling designs reduce nuclear lifting risks ALARP.

#### Steam Generator removal

1. I assessed the RP’s RI mechanical handling design summary report (ref. [54]). I noted that the RP states that it is considering SG removal options. The RP’s extant solution is to:

* introduce a suitable opening in its RI hazard shield; and
* lift the SG through this opening.

1. The RP recognises that optioneering and design development is required to underpin its final concept design.
2. For Step 2 I judge that the RP has adequately demonstrated that it is considering SG mechanical handling options.
3. During Step 3 I will seek assurance that its SG removal strategy adequately demonstrates that nuclear lifting risks are reduced ALARP.

#### Reducing risks ALARP and use of relevant good practice summary

1. I conclude that the RP has:

* adequately demonstrated that it:
  + is working to reduce risks ALARP, and that production of supporting evidence is ongoing;
  + is considering further design revisions as the Rolls-Royce SMR design and E3S case matures; and
  + has reduced nuclear cask lifts.
* not yet adequately demonstrated that some design decisions are robust.

1. For Step 2 I am satisfied. This is because I consider that during Step 3 the RP has the opportunity to demonstrate that its design decisions are robust and that it is working towards reducing risks ALARP as its design matures.
2. During Step 3, I will seek assurance that the Rolls-Royce SMR design reduces mechanical engineering risks ALARP. I consider this a residual matter.

### Categorisation of safety functions and classification of structures, systems and components

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s categorisation and classification approach as applied to mechanical systems.
2. My assessment sought to judge whether the RP has applied an adequate categorisation and classification methodology to its design. This is because an adequate methodology supports correct SSC nuclear safety classification. This is important as it informs a number of other important safety related activities (for example, EIMT, quality assurance, etc).
3. To assess the RP’s categorisation and classification approach I assessed its categorisation and classification methodology. I also targeted the following systems given their contribution to nuclear safety:

* reactor coolant pump (RCP); and
* mechanical handling

#### Methodology

1. I assessed the RP’s categorisation and classification methodology. I considered that the methodology:

* Adequately described the RP’s graded approach to categorisation of safety functions and SSC safety classification. This aligns with my expectations against:
  + ONR SAP ECS.2 (Safety classification of structures, systems and components) (ref. [11]);
  + ONR TAG NS-TAST-GD-094 – Categorisation of Safety Functions and Classification of Structures, Systems and Components (ref. [32]); and
  + IAEA Safety of Nuclear power Plants: Design (SSR-2/1) (ref. [33]).
* Was unclear regarding its application of principles of prevention. This is because I considered it defaulted to assigning the higher classification to protective safety systems in the first instance. RGP for SSC classification expects that all practicable steps must be taken to prevent and mitigate nuclear or radiation accidents in the first instance. For example ONR:
  + SAP FP.6 (Prevention of accidents) (ref. [11]);
  + SAP EKP.3 (Defence in depth) (ref. [11]); and
  + TAG NS-TAST-GD-094 Categorisation of Safety Functions Classification of Structures, Systems and Components (ref. [32]) section 5.7.4.1.

1. I raised RQ-01071 (ref. [61]) to seek clarification. In addition, a technical meeting (ref. [90]) was held to discuss and improve clarity of the RP’s categorisation and classification methodology.
2. The RP revised its methodology to include a process for prioritising the principle of prevention over protection (section 3.4 of [91]). The RP refers to this as up-rating SSC classifications. I consider this to be a positive addition by the RP due to its impact on improving the safety of the overall Rolls-Royce SMR design.
3. The RP recognises other factors may influence an SSC’s final classification. These include when:

* an SSC’s classification aligns with RGP;
* it is reasonably practicable to up-rate a preventative SSC’s classification; and
* probabilistic safety assessment (PSA) identifies an SSC as being of greater importance to safety.

1. I judge that for Step 2 the RP’s SSC up-rating of classification arrangements are appropriate. This is because it aligns with my expectations against ONR:

* SAP FP.6 (Prevention of accidents) (ref. [11]);
* SAP ECS.2 (Safety classification of structures, systems and components) (ref. [11]);
* SAP EKP.3 (Defence in depth) (ref. [11]); and
* TAG NS-TAST-GD-094 – Categorisation of Safety Functions Classification of Structures, Systems and Components (ref. [32]).

1. During Step 3 I will seek assurance on the adequacy of implementation of the RP’s SSC classification up-rating arrangements.

#### Reactor coolant pump

1. I assessed the RP’s RCP component safety classification (ref. [92]). I considered it unclear how the RCP classification aligned with my expectations against ONR SAP ECS.2 (Safety classification of structures, systems and components) (ref. [11]). This is because it was unclear:

* How the RCP coast down function classification was appropriate. This is because a nuclear safety class 2 allocation was made against a safety category A function.
* Which seals contribute to the RCP category A safety function to contain and confine primary circuit coolant.

1. I raised RQ-01181 (ref. [61]) to seek clarification. The RP responded that its E3S case will be revised to identify RCP:

* coast down performance as nuclear safety class 1; and
* mechanical seals with nuclear safety claims.

1. I consider that the RP’s response aligns with my expectations against ONR SAP ECS.2 (Safety classification of structures, systems and components) (ref. [11]).
2. Hence, for Step 2 I judge that the RP has adequately demonstrated that its RCP safety classification is appropriate.

#### Mechanical handling

1. To assess the RP’s classification approach I assessed the RI mechanical handling design summary report (ref. [54]). I recognised the RP’s intent to appropriately classify its mechanical handling equipment but considered it unclear how drop load consequences had informed the RP’s mechanical handling classification basis.
2. I raised RQ-01180 (ref. [61]) to seek clarification. The RP responded that:

* initial studies have been undertaken to identify drop hazards, consequences and safety functional requirements;
* candidate mechanical handling solutions have been proposed by the RP during the design development optioneering; and
* definition of claims, arguments and evidence (including classification) are expected during Step 3.

1. I assessed the RP’s mechanical classification approach in its design definition report for conveyance of fuel assemblies / internals within the storage area (ref. [93]). I noted that the RP explains that because of the design maturity, safety function categorisation has not been assigned (section 3.5.10 of (ref. [93])). I note that this is consistent with mechanical handling SSCs identified in the reactor area (section 3.5.10 of (ref. [53])).
2. The RP claims that it has applied a conservative approach to its concept mechanical handling SSCs. For example, the RP has assigned its RI main overhead crane (MOC) and in-containment fuel handling machine (FHM) as nuclear safety class 1 based on the expected consequences of structural failure and collapse (section 3.7.4 of (ref. [53])). I consider that this provides some assurance that the RP is considering mechanical handling faults in its developing design.
3. I assessed the RP’s fault schedule (ref. [44] [45]). It is evident that the RP has begun including drop loads in its fault schedule (mechanical handling section) but that this work is incomplete.
4. In accordance with ONR SAP EKP.2 (Fault tolerance) (ref. [11]), during GDA I expect the mechanical handling drop loads to be identified through a systematic approach. I consider that this is important where there are complex mechanical handling interactions with or around other SSCs (TAG NS-TAST-GD-056 paragraph 41 of (ref. [29])). Consequences from such systematic hazard identification is expected to inform the RP’s mechanical handling SSC classification.
5. Hence, for Step 2 I judge that the RP has not adequately demonstrated its mechanical handling SSC classification approach.
6. I am satisfied that the RP can present its mechanical handling classification approach during Step 3. This is because:

* I expect the RP’s mechanical handling design and safety case development to progress to final concept design during Step 3; and
* I have seen some evidence that the RP is developing its drop load hazard analysis to inform its mechanical handling SSC classification.

1. During Step 3 I will seek assurance that the RP’s mechanical handling classification is adequate. I also intend to assess the RP’s approach to faults within the probability region of 1 to 10-3 per year. This is because the RP’s categorisation and classification methodology indicates a reduced need for a diverse safety function for faults with lower radiological consequences. The ONR fault studies specialism has indicated that this may be a shortfall against the expectations that two lines of protection are provided for frequent faults (ONR TAG NS-TAST-GD-006 Design Basis Analysis (ref. [25])). Further information is available in ONR fault studies RQ-01149 (ref. [61]).

#### Categorisation of safety functions and classification of structures, systems and components summary

1. I conclude that the RP has:

* adequately demonstrated that:
  + it’s categorisation and classification methodology considers principles of prevention; and
  + the RCP has been correctly classified.
* not yet adequately demonstrated its mechanical handling SSC classification is underpinned.

1. For Step 2 I am satisfied. This is because I consider that the RP has:

* recognised that its mechanical handling SSC classification remains outstanding; and
* the opportunity to demonstrate its mechanical handling SSC classification during Step 3.

1. During Step 3 I will seek assurance that the RP’s mechanical handling classification is adequate. I consider this a residual matter.

### Examination, inspection, maintenance and testing

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s examination, inspection, maintenance and testing (EIMT) approach.
2. To assess the RP’s EIMT approach I assessed its EIMT strategy (ref. [51]). I also targeted the following given their nuclear safety relevance:

* reactor coolant pump (RCP); and
* integrated head package (IHP) mechanical handling.

#### Examination, inspection, maintenance and testing strategy

1. The RP’s EIMT strategy (ref. [51]) informs its SSC EIMT application approach. I assessed the EIMT strategy and considered it unclear regarding its approach to:

* ageing and degradation management;
* on-line maintenance (OLM);
* reducing EIMT risks early in the design;
* identifying through-life activities (TLAs) to support its design readiness hold points; and
* preventing risk reduction options from being foreclosed.

1. I raised RQ-01304 (ref. [61]) for clarification. The RP responded that:

* its ageing management plan (ref. [94]) defines its approach to ageing and degradation;
* identification of EIMT activities to monitor SSC degradation will be identified in equipment qualification (EQ) activities;
* no on line maintenance (OLM) has been identified in designs released to final concept review;
* it is developing an OLM assessment guide to inform where it is acceptable; and
* its design for nuclear site health and safety process (C3.2.2-4) manages risks through the design process and focuses on eliminating concept stage risks.

1. In my opinion, the RP has not yet adequately demonstrated that its C3.2.2-4 process manages risk reduction during concept design.
2. I assessed the RP’s ageing management plan (ref. [94]) and noted that:

* ageing and degradation mechanisms are extensively described (table 1);
* Non-metallic reactor island components will be included in issue 2 (section 1.2.14); and
* appropriate codes and standards relating to credible degradation mechanisms are identified.

1. Hence, for the EIMT strategy I conclude that the RP has adequately demonstrated that its ageing management approach aligns with my expectations against ONR SAPs (ref. [11]):

* ECS.3 (Codes and standards);
* EAD.1 (Safe working life); and
* EAD.2 (Lifetime margins).

1. During Step 3 I will seek assurance that the RP’s:

* OLM approach is adequate.
* Ageing management plan revision includes appropriate non-metallic SSCs.
* Nuclear site health and safety process (C3.3.3-4) application reduces EIMT risks ALARP.

#### Reactor coolant pump

1. I assessed the RP’s RCP EIMT activities. This is because the RCP is a hermetically sealed pump and is directly coupled underneath the associated SG. I consider this arrangement to have potentially challenging EIMT implications. This is because:

* The RCP bearings are lubricated by the reactor’s primary coolant. This may result in higher EIMT personnel radiation dose than other RCP designs. This is because in other RCPs, where the motor and pump can be removed from above the impeller, the thrust bearing is oil lubricated.
* The RCP’s are located in voids underneath the SGs. Adequate personnel access and egress and EIMT space will be necessary.

1. I assessed the RP’s RCP EIMT approach in the E3S case Chapter 5: reactor coolant system (ref. [3]). I considered it unclear:

* what RCP EIMT was required; and
* how RCP EIMT would be undertaken

1. I raised RQ-01023 (ref. [61]) to seek clarification. The RP responded that the RCP’s EIMT:

* design was at DR1 maturity (preliminary concept);
* access and egress had not yet been defined; and
* the radiation source term had not yet been defined.

1. I assessed the RP’s layout summary report (ref. [95]) and the RI mechanical handling design summary report (ref. [54]) and noted:

* that the RCP is expected to require in-situ EIMT;
* the radiation source term had not yet been defined.
* each RCP removal and replacement is expected once during the plant lifecycle;
* the removal sequence requires each RCP to be:
  + supported and lowered from the SGs;
  + traversed to the extraction pits at the containment sump’s eastern end; and
  + lifted to the containment ground floor level before being removed through the equipment access hatch (EAH).

1. Figure 6 shows the RCP plan and side elevation relative to the SG sump.

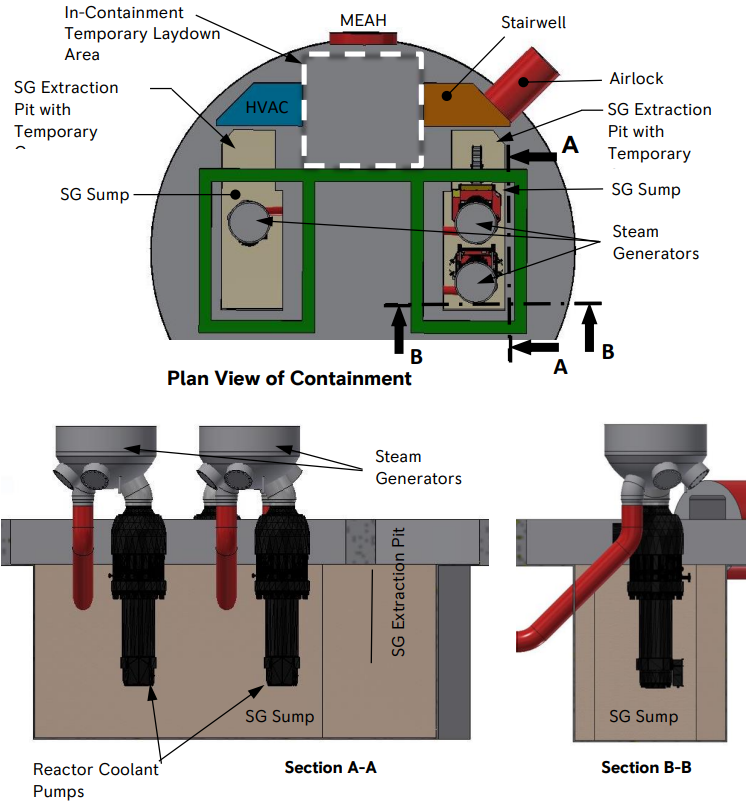


Figure 6: Reactor coolant pump location in the steam generator sump (ref. [54])

1. I assessed the RP’s RCP mechanical handling considerations. These are presented in the reactor island mechanical handling design summary report (ref. [54]). I noted that the RP recognises:

* space constraints posed by the containment building sump profile (paragraph 6.1.5);
* space constraints are because of the sump volume being minimised to achieve required ECC water height (paragraph 6.1.6);
* the sump would be a confined operational space for RCP EIMT (paragraph 6.1.6);
* that where two RCPs are located in the same SG sump (section A-A in Figure 6) both would require removal if EIMT was required on the one furthest from the extraction pit;
* personnel access and egress other than via the EAH is not possible because floor plates are removed for RCP lifting (paragraph 6.1.20); and
* the proposed concept requires a feasibility review following determination of dose uptake.

1. I consider that the RP has not yet adequately demonstrated how its RCP EIMT approach meets regulatory requirements. This is because it is unclear:

* How safe personnel access and egress during EIMT is achieved. This does not align with my expectations against ONR:
  + SAP ELO.1 (Access) (ref. [11]). This expects a facility’s layout to make provision for EIMT activities and personnel access and egress.
  + TAG NS-TAST-GD-009 – Examination, Inspection, Maintenance and Testing of Items Important to Safety (ref. [26]). This expects EIMT access provision.
* How the RCP EIMT activities have reduced personnel dose uptake ALARP. This is because I have not seen evidence that the RP has used a baseline source term to inform its RCP EIMT activities. This does not align with my expectations against ONR SAP RP.7 (Hierarchy of control measures) (ref. [11]).
* How crushing hazards are avoided during RCP inspection or removal and replacement. I consider this a shortfall against LOLER expectations (ref. [86]). This expects:
  + means of access to lifting equipment (paragraph 65 of (ref. [86]) and associated guidance);
  + loads should not be carried or suspended over areas occupied by people (paragraph 230 of (ref. [86]) and associated guidance);
  + minimising risk of clashes between lifting equipment and other objects (paragraph 265, 266 of (ref. [86]) and supporting guidance); and
  + design for EIMT (regulation 9 of (ref. [86]) and supporting guidance).

1. Hence, for Step 2 I conclude that for the RCP EIMT, the RP has:

* adequately demonstrated:
  + it has recognised some radiological and nuclear site health and safety risks associated with its design;
  + interdependence between RCP EIMT and containment building sump volume; and
  + that it is developing its design and E3S case.
* not yet adequately demonstrated:
  + what EIMT is required;
  + when EIMT is required; and
  + how RCP EIMT can be achieved safely with the extant containment building layout.

1. I am satisfied that the RCP EIMT design is at an early stage and is being developed to a final concept. I note that the RP has recognised its need to consider EIMT dose during design development. Hence, I consider that the RP has the opportunity to demonstrate that its design reduces nuclear site health and safety and radiological safety risks ALARP during Step 3.
2. During Step 3 I will seek assurance that the RP’s RCP EIMT requirements satisfy regulatory expectations.

#### Integrated head package

1. I sampled the IHP mechanical handling to inform my EIMT assessment. This is because the IHP mechanical handling requires:

* lifting activities adjacent to radiological material; and
* an IHP lift gate in the containment building layout.

1. I assessed the RP’s E3S Case Chapter 5: reactor coolant systems and associated systems (ref. [3]), the IHP lift gate (PT536) component definition (ref. [96]) and the mechanical sequence diagrams (MSDs) (ref. [49]). I noted that the:

* IHP is removed during refuelling (section 5.3.6.1 of (ref. [3])); and
* IHP lift gate is required to retain coolant and provide shielding during an outage (PT536-D-25 in (ref. [96])).
* IHP lift gate is included in the design to accommodate the maximum lift height that can be currently achieved by the main overhead crane (MOC). I understand that the current MOC is unable to lift the IHP onto its laydown area without the IHP lift gate being in its lowered position. I note that this introduces:
  + additional mechanical handling activities; and
  + a refuelling pool leak path.
* Mechanical handling EIMT laydown areas overlap with the EAH opening. I consider that this may introduce additional personnel and EIMT equipment access and operation risks.
* Personnel access and egress provisions were not yet identified.

1. I considered it unclear how the IHP lift gate and MOC designs reduce containment building mechanical handling and nuclear site health and safety risks ALARP. This is because the RP has not demonstrated why introduction of the IHP lift gate reduces risks when compared with increasing the MOC lift height
2. I note that the RP has developed a draft product safety philosophy (ref. [97]). This refers to a product safety review board (PSRB) that is arranged periodically to assess areas of design concern. This has not been implemented in the RP’s processes to date. Hence I am unable to conclude whether the proposed process is an effective means of demonstrating adequate resolution of conflicting design requirements. However, I recognise that the RP is developing its design governance processes to manage mechanical engineering design conflicts. The RP has indicated (ref. [98]) that the IHP lift gate and MOC interaction is an example of where its PSRB will be applied. I intend to engage with the RP on this matter during Step 3.
3. For Step 2 I conclude that the RP has not yet adequately demonstrated that its IHP lift gate and MOC design reduce risks ALARP. This is because I consider it unclear how the:

* MOC design and lift height restrictions reduces risks ALARP regarding:
  + mechanical handling; and
  + EIMT personnel access and egress.
* RP’s PSRB will:
  + be initiated (i.e. what process initiates a PSRB);
  + resolve conflicts between technical and/or E3S requirements (i.e. what terms of reference will show safety is integral to the design).

1. This does not align with my Step 2 expectations against:

* HSE Management of Health and Safety at Work Regulations (ref. [99]). Regulation 4 expects principles of prevention to be applied in designs.
* ONR SAP MS.3 (Decision making) paragraph 71 (ref. [11]).
* ONR SAP ELO.1 (Access) (ref. [11]);
* ONR SAP ELO.4 (Minimisation of the effects of incidents) (ref. [11]);
* ONR TAG NS-TAST-GD-056 – Nuclear Lifting Operations (paragraph 17 of (ref. [29])).

1. I recognise that the IHP EIMT design is maturing from a preliminary concept level and that the identified shortfalls can be addressed as it progresses to final concept during Step 3. I note that the RP has recognised its need to improve its design governance. This is evidenced through its PSRB process introduction. Hence, for Step 2 I am satisfied that the RP has the opportunity to demonstrate that its IHP EIMT design reduces nuclear site health and safety and radiological safety risks ALARP during Step 3.
2. During Step 3 I will seek assurance that its IHP lift and MOC risks are reduced ALARP.

#### Examination, inspection, maintenance and testing summary

1. I conclude that the RP has:

* adequately demonstrated that:
  + its EIMT strategy identifies applicable RGP; and
  + it is developing its nuclear site health and safety arrangements.
* not yet shown how its layout reduces EIMT risks ALARP.

1. For Step 2 I am satisfied that the RP has the opportunity to demonstrate that its EIMT design reduces nuclear site health and safety and radiological safety risks ALARP during Step 3.
2. During Step 3 I will seek assurance that the RP’s layout reduces EIMT risks ALARP. I consider this a residual matter.

### Equipment qualification assessment

1. My Step 2 plan (ref. [15]) included an assessment of the RP’s equipment qualification (EQ) approach.
2. I assessed the RP’s E3S objectives and design rules for SSCs (ref. [2]) to inform my judgement on its EQ approach. I considered it unclear how the RP’s EQ arrangements adequately identified:

* mission times;
* qualified life requirements;
* seismic qualification requirements;
* harsh and mild environmental conditions;
* how EQ is proportionate to an SSC’s classification; and
* configuration control arrangements.

1. I raised RQ-01028 (ref. [61]) to seek clarification. The RP stated that:

* its design verification process (internal process C3.1.2 and C3.2.3) aligns with international guidance (IAEA SSR-2/1 section 6.43 (ref. [33]));
* SSC mission times are predominantly derived by fault analysis;
* qualified life requirements are considered with age related degradation mechanisms that inform EIMT;
* harsh environment conditions will be defined by environmental analysis;
* the flow of environmental requirements into SSC’s requirements specifications is being developed;
* EQ methods include:
  + physical testing;
  + analysis; and
  + OPEX.
* a verification and validation approach was being developed and would be submitted for assessment.

1. I assessed the RP’s validation and verification approach document (ref. [100]). I considered it unclear how the RP’s verification and validation approach considered SSC ageing and degradation.
2. I raised RQ-01252 (ref. [61]) to seek clarification. The RP responded stating that:

* SSC service conditions will be established during design activities;
* service conditions will be assessed to determine whether they are ageing stressors;
* ageing stressors will inform whether accelerated ageing is required during an SSC’s EQ activities; and
* nuclear industry codes and standards (for example ASME QME-1) will guide EQ activities.

1. I consider that this aligns with my expectations against:

* IAEA Guidance SSG-69 – Equipment Qualification for Nuclear Installations (ref. [37]).
* ONR SAPs (ref. [11]):
  + EQU.1 (Qualification procedures);
  + EAD.1 (Safe working life); and
  + EAD.2 (Lifetime margins).

1. For Step 2 I conclude that the RP’s EQ approach adequately demonstrates activities:

* are proportionate to an SSC’s classification;
* consider environmental conditions, including harsh environments;
* are informed by ageing and degradation mechanisms; and
* consider appropriate EQ methods to demonstrate SSCs will deliver their safety function.

1. During Step 3 I will seek assurance that the RP’s:

* verification and validation approach, when applied, delivers adequate SSC EQ; and
* environmental conditions are identified and inform SSC qualification activities.

# Conclusions

* 1. Conclusions

1. This report presents the Step 2 mechanical engineering assessment for the GDA of the Rolls-Royce SMR design. The focus of my assessment in this step was towards the fundamental adequacy of the design and safety case. I have assessed the Tier 1 E3S chapters and relevant supporting documentation provided by Rolls-Royce SMR Limited to form my judgements. I targeted my assessment, in accordance with my assessment plan (ref. [15]), at the content of most relevance to mechanical engineering against the expectations of ONR’s SAPs, TAGs and other guidance which ONR regards as relevant good practice.
2. Based on my Step 2 assessment, I have concluded that the RP has adequately demonstrated:

* That its E3S case is logically structured and accessible. I also conclude that the RP has not yet shown that its E3S case has:
  + applied mechanical engineering design rules to identify safety parameters that may affect the Rolls-Royce SMR design layout.
  + a completed claims, arguments and evidence structure; and
  + adequate safety requirements traceability.
* How its limits and conditions development approach and manufacturer engagement considers RGP. I also conclude that the RP has not yet shown that it has identified key design parameters for mechanical engineering SSCs. This includes HVAC and seismic performance requirements.
* That its passive safety approach aligns with RGP.
* That its modularisation approach and strategy is developing. I also conclude that the RP has not yet shown for mechanical engineering SSCs how its modularisation approach considers seismic loads, mechanical handling loads, transport loads and EIMT requirements.
* Consideration of EAH and primary containment building leak testing. I also conclude that the RP has not yet adequately demonstrated its HVAC primary containment design or EAH EIMT requirements
* That its DiD approach aligns with RGP. I also conclude that the RP has not yet shown how CCF and SFC risks have been addressed in its SRV design.
* That its engineering management plan is appropriate because it requires early design optioneering and controls output through its gated reviews. I also conclude that the RP has not yet shown how its:
  + optioneering methodology, when applied, will lead to outcomes that appropriately prioritise safety.
  + RCP optioneering outcome reduces risks ALARP;
  + gated review identifies and manages E3S case requirement interdependencies; and
  + layout achieves adequate mechanical handling personnel access for operations and EIMT.
* That it is working to reduce risks ALARP. This has included recognition that further Rolls-Royce SMR design revisions are necessary and consideration of nuclear cask lifts reduction. I also conclude that the RP has not yet shown that its design decisions are robust.
* That its categorisation and classification methodology considers principles of prevention and that its RCP has been correctly classified. I also conclude that it has not yet shown that its mechanical handling SSCs are correctly classified.
* That its EIMT strategy identifies RGP and that it is developing its nuclear site health and safety arrangements. I also conclude that it has not yet shown how its layout reduces EIMT risks ALARP.
* That its EQ approach aligns with RGP. This is because I consider that it:
  + is proportionate to an SSC’s classification;
  + considers environmental conditions, including harsh environments;
  + is informed by ageing and degradation mechanisms; and
  + considers appropriate methods to demonstrate SSCs will deliver their safety function.

1. Overall, based on my Step 2 assessment to date, and subject to the provision and assessment of suitable and sufficient supporting evidence, I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic Rolls-Royce SMR design.
   1. Recommendations
2. My recommendation is as follows:

* Recommendation 1: ONR should consider the outcomes from my assessment as part of the decision to progress to Step 3 of GDA for the generic Rolls-Royce SMR design.

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Appendix 1 – Relevant SAPs considered during the assessment

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| SAP No. | SAP Title |
| FP.3 | Optimisation of protection |
| FP.6 | Prevention of accidents |
| MS.3 | Decision making |
| SC.2 | Safety case process outputs |
| SC.4 | Safety case characteristics |
| EKP.1 | Inherent safety |
| EKP.3 | Defence in depth |
| EKP.5 | Safety measures |
| ECS.2 | Safety classification of structures, systems and components |
| ECS.3 | Codes and standards |
| EQU.1 | Qualification procedures |
| EDR.1 | Failure to safety |
| EDR.2 | Redundancy, diversity and segregation |
| EDR.3 | Common cause failure |
| EDR.4 | Single failure criterion |
| ERL.4 | Margins of conservatism |
| EMT.1 | Identification of requirements |
| EMT.5 | Procedures |
| EMT.6 | Reliability claims |
| EAD.1 | Safe working life |
| EAD.2 | Lifetime margins |
| ELO.1 | Access |
| ELO.4 | Minimisation of the effects of incidents |
| EHA.9 | Earthquakes |
| EPS.3 | Pressure relief |
| EPS.4 | Overpressure protection |
| ECV.3 | Means of confinement |
| ECV.10 | Ventilation system safety functions |
| ERC.1 | Design and operation of reactors |
| EHT.1 | Design |
| EHT.2 | Coolant inventory and flow |
| EHT.4 | Failure of heat transport system |
| RP.7 | Hierarchy of control measures |
| DC.1 | Design and operation |