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



ONR Assessment Report

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

**Report Title**: Step 2 assessment of Structural Integrity

**Authored by**: [Redacted]

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# Executive Summary

This report presents the outcomes of my structural integrity 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 the Rolls-Royce SMR Limited’s Environmental, Safety, Security and Safeguards (E3S) case chapters and supporting documentation.

ONR’s GDA process calls for a step-wise assessment, which increase in detail as the project progresses. The focus of my assessment in this step was towards the fundamental adequacy of the Rolls-Royce SMR design and safety case, 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.

I targeted my assessment, in accordance with my assessment plan, at the content of most relevance to structural integrity against the expectations of ONR’s Safety Assessment Principles (SAPs), Technical Assessment Guides (TAGs) and other guidance which ONR regards as relevant good practice.

My assessment focused on the structural integrity aspects within the Rolls-Royce SMR design E3S case, including safety claims, component classification approach, design concepts and structural integrity specific analysis approaches and methodologies.

Based upon my assessment, I have concluded the following:

* The E3S case explicitly considers structural integrity, with traceable links between safety functional requirements and structural integrity controls applied to component design and manufacture. Structural integrity requirements are well substantiated, based on relevant Operational Experience (OPEX) and international guidance.
* The RP’s approach to component classification is reasonable, accounting for direct and indirect consequences of gross failure, which aligns with ONR expectations for beyond-code demonstration. I consider aspects of the RP’s approach to use multiple beyond-code classifications requires further assessment, to understand how the applied structural integrity controls are proportionately defined by the consequences and tolerability of gross failure.
* The RP has a good understanding of ONR expectations for highest reliability components through a conservative avoidance of fracture demonstration (AoFD) approach, providing beyond code substantiation.
* The RP has selected a well-established set of nuclear specific codes and standards to inform the design of the most safety significant components. I am satisfied at a high level that the RP’s approach incorporates suitable structural analysis, taking into account OPEX and environmental effects, where relevant.
* The RP’s approach to component design considers access for inspectability. Whilst this approach aligns with ONR expectations, it is heavily reliant on plant layout, which was not sufficiently developed at this stage of GDA to enable assessment. I consider this requires further assessment at a component-specific level when the E3S case is more developed, to confirm this approach is appropriately implemented across the Rolls-Royce SMR design.
* The containment vessel design is based on an established nuclear code which at a high level meets ONR expectations. The detailed substantiation of structural integrity controls requires further consideration, regarding the use of code cases and the containment vessel support structure.
* The Rolls-Royce SMR design uses a seismic isolation system safety feature, which directly influences the structural integrity design aspects of primary circuit, secondary circuit and safety system components. It is therefore important that adequate substantiation of the seismic isolation system design is provided to satisfy ONR expectations, which is necessary to underpin structural integrity claims.
* The RP’s materials selection approach is based on proven materials identified from codes and standards, with provisions to optimise the material properties specific to the operating environment. I consider this approach is broadly aligned with ONR expectations. The RP has established a robust approach for the management of ageing and degradation mechanisms, which aligns with ONR expectations. The RP’s approach has identified gaps in OPEX related to material compatibility with the proposed Rolls-Royce SMR design primary coolant chemistry regime, and is taking action to address them.
* The maturity and development status of the Rolls-Royce SMR design has a degree of uncertainty affecting structural integrity demonstration, including substantiation of the compact plant layout, component classification, safety analysis and component design optimisation. Whilst I do not consider this to be inhibitive for assessment within Step 2, further detailed information is required, mostly at a component-specific level, to support the E3S structural integrity claim.

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.

# List of Abbreviations

ALARP As low as is reasonably practicable

AMP Ageing Management Plan

AoFD Avoidance of Fracture Demonstration

AP Assessment Plan

AR Assessment Report (Step 2)

ASME American Society of Mechanical Engineers (design code)

ASTM American Society for Testing and Materials

BPVC Boiler & Pressure Vessel Code

CAE Claims, Arguments and Evidence

CH Closure Head

CHS Conventional Health and Safety

CoR Control of Reactivity

CoRM Confinement of Radioactive Material

CSR Component Substantiation Reports

CVS Containment Vessel Structure

DAC Design Acceptance Confirmation

DEC Design Extension Condition

DfX Design for X

DTA Defect Tolerance Assessment

DTAG Defect Tolerance Assessment Guidance

E3S Environmental, Safety, Security and Safeguards

EAF Environmentally-Assisted Fatigue

EEP Expert Elicitation Panel

ENIQ European Network for Inspection Qualification

EPRI Electric Power Research Institute

FCG Fatigue Crack Growth

FMEA Failure Modes and Effects Analysis

FSF Fundamental Safety Functions

GDA Generic Design Assessment

GR Gated Review

HLSF High Level Safety Functions

HR High Reliability

IAEA International Atomic Energy Agency

IPT Integrated Project Team

ISI In-service Inspection

ITPIA Independent Third Party Inspection Agent

IVR In-vessel Retention

KOH Potassium hydroxide

LDS Limiting Defect Size

LiOH Lithium hydroxide

MSQA Management for Safety and Quality Assurance

NC Non-nuclear Classified

NDE Non-Destructive Examination

NRW Natural Resources Wales

ONR Office for Nuclear Regulation

OPEX Operational Experience

PDI Performance Demonstration Initiative

PE(S)R Pressure Equipment Safety Regulations

PSI Pre-service Inspection

PSSR Pressure Systems Safety Regulations

PWHT Post Weld Heat Treatment

PWR Pressurised Water Reactor

RCP Reactor Coolant Pump

RF Reserve Factor

RGP Relevant Good Practice

RP Requesting Party

RPV Reactor Pressure Vessel

RQ Regulatory Question

RT Radiographic Testing

SAP Safety Assessment Principle(s)

SC Subclaim

SG Steam Generator

SIR Structural Integrity Requirements

SMR Small Modular Reactor

SQEP Suitably Qualified and Experienced Persons

SSC Structure, System and Component

TAG Technical Assessment Guide(s) (ONR)

TAGSI Technical Advisory Group of structural integrity

TB Technical Basis

TJ Technical Justifications

TSC Technical Support Contractor

USFTT Upper Shelf Fracture Toughness Testing

USNRC United States Nuclear Regulatory Commission

UT Ultrasonic Testing

VHR Very High Reliability

WENRA Western European Nuclear Regulators’ Association

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# Introduction

1. This report presents the outcomes of my structural integrity 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] and [9]) and supporting documentation.
2. Assessment was undertaken in accordance with the requirements of the 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-TAST-GD-108 (ref. [13])).

## Background

1. 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.
2. 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.
3. 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.
4. Prior to the start of Step 2, I prepared a detailed Assessment Plan for structural integrity (ref. [15]). This has formed the basis of this assessment and was also shared with the RP to maximise openness and transparency.
5. 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]).

## Scope

1. 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.
2. 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 that are identified as being important to safety, security and safeguards, all modes of operation, and all stages of the plant lifecycle.
3. However, given the step-wise assessment during GDA, information has not been submitted for all aspects within the GDA scope during Step 2. The following aspects of the E3S case are therefore out of scope of this assessment:

* Detailed evidence at the plant component-level. Step 2 has focused on the RP’s high level E3S case arrangements, approaches, methodologies and strategies, which summarise the claims and arguments level of a hierarchical safety case. Detailed evidence to demonstrate how such approaches and requirements are implemented for the Rolls-Royce SMR design are considered more appropriate for a Step 3 assessment.
* Information related to activities to be established as part of licensing, such as in-service examination, inspection, maintenance and testing regimes and strategies. My assessment has only considered whether the fundamental Rolls-Royce SMR design has been developed to enable such activities, to ensure that a licensee can meet ONR expectations during licensing and operation.

1. My assessment has considered the following aspects, as defined in section 3.20 of ONR’s GDA Technical Guidance (ref. [18]):

* The RP’s overall approach to structural integrity, including multi-discipline and cross-discipline interactions with other topic areas.
* Structural integrity claims on structures and components, including highest reliability claims.
* The proposed codes and standards.
* Materials testing and surveillance activities with the emphasis on structures and components underpinned by highest reliability claims and safety significant SSC.
* The structural integrity safety case strategy, including the approach to providing the beyond design code compliance justifications for highest reliability claims.
* The basis for an avoidance of fracture demonstration in support of highest reliably claims.
* Manufacturing, pre and in-service inspection (examination) and testing strategies.
* Design for access and inspectability.
* Materials selection and manufacturing (including fabrication) techniques along with the identification of through-life degradation mechanisms and an outline of the mitigation strategies to underpin the proposed design life.
* Broad consideration of how structural integrity claims demonstrate risk is reduced As Low As Reasonably Practicable (ALARP).

# 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 preliminary 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.

## Standards

1. The ONR SAPs (ref. [11]) 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.
2. The International Atomic Energy Agency (IAEA) safety standards (ref. [19]) and nuclear security series (ref. [20]) 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.
3. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [21]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [22]).
4. 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 structural integrity aspects in practice.

### Safety Assessment Principles (SAPs)

1. The key SAPs (ref. [11]) applied within my assessment are: EMC.1 to EMC.34 on the integrity of metal components and structures (EMC.1 to EMC.3 are relevant to highest reliability claims); ECS.1 to ECS.3 on safety classification; and EAD.1 to EAD.4 on ageing and degradation.
2. A list of the SAPs used in this assessment is recorded in Appendix 1.

### Technical Assessment Guides (TAGs)

1. The following TAGs have been used as part of this assessment:

* ONR-TAST-GD-005 – Revision 12, January 2024. Regulating duties to reduce risks to ALARP (As Low as Reasonably Practicable) (ref. [23])
* NS-TAST-GD-016 – Revision 7, April 2020. Integrity of Metal Structures. Systems and Components (ref. [24]).
* ONR-TAST-GD-051 – Revision 7.1, December 2022. The Purpose, Scope and Content of Safety Cases (ref. [25]).
* ONR-TAST-GD-067 – Revision 3.2, June 2023. Pressure Systems Safety. (ref. [26]).
* ONR-TAST-GD-094 – Revision 2, July 2019. Categorisation of Safety Functions and Classification of Structures, Systems and Components (ref. [27]).
* NS-TAST-GD-096 – Revision 1.2, December 2022. Guidance on Mechanics of Assessment (ref. [10]).

### National and international standards and guidance

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

* IAEA, Safety of Nuclear Power Plants: Design, Specific Safety Requirements SSR-2/1, Rev. 1 (ref. [28])
* IAEA, Format and Content of the Safety Analysis Report for Nuclear Power Plants, Specific Safety Guide No. SSG-61 (ref. [29])
* IAEA, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants, Specific Safety Guide SSG-56 (ref. [30])
* IAEA, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, Specific Safety Guide No. SSG-48 (ref. [31])
* The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Sections II, III and XI (ref. [32]).
* European Methodology for Qualification of Non-Destructive Testing. ENIQ Report No. 61 Issue 4 March 2019 (ref. [33]).

## Integration with other assessment topics

1. I have worked closely with other topics as part of my structural integrity 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:

* The structural integrity assessment provides input to the categorisation of safety functions and the classification of structures, systems and components (SSCs) aspects of the fault studies assessment. The fault studies inspectors provide advice on the structural integrity claims needed to support the overall safety case for the plant. This work is being led by ONR's fault studies discipline.
* The structural integrity assessment provides input to the missile generation, pipe-whip and internal flooding aspects of the internal hazards assessment. The results of the RP's indirect consequences assessment inform the structural integrity classifications. This work is being led by ONR's internal hazards discipline.
* The structural integrity assessment provides input on the containment structure, as a metallic, freestanding pressure vessel. This interaction has commenced during GDA Step 2, with some initial considerations of the aspects of civil building structure interfaces with plant important to safety, such as the containment structure support and steel framed mechanical-electrical plant modules. This work was undertaken in conjunction with ONR's civil engineering inspector with respect to the jurisdiction of physical metal supports and the importance of the seismic isolation system on nuclear pressure vessel and piping design.
* The chemistry discipline provided input to materials selection and the assessment of potential through-life degradation aspects of the structural integrity assessment.
* In addition to the above there have been interactions between structural integrity and several other technical areas, for example, fuel and core, electrical engineering and management for safety and quality assurance (MSQA). These interactions are important to ensure consistency across ONR's assessment of specific technical aspects of the Rolls-Royce SMR design and are discussed where relevant.

## Use of technical support contractors

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

# Requesting party’s submission

1. Rolls-Royce SMR Limited submitted a series of E3S case 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 structural integrity. It also identifies the documents submitted by the RP which have formed the basis of my structural integrity assessment of the Rolls-Royce SMR design.

## Summary of the Rolls-Royce SMR design

1. 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.
2. 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.
3. The reactor itself is of a typical PWR design, including a steel Reactor Pressure Vessel (RPV) holding fuel assemblies, Steam Generators (SG), pressuriser, Reactor Coolant Pumps (RCP) and reactor coolant/steam 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.
4. The RP’s design reference point (DRP) for step 2 GDA has been defined within its GDA Design Reference Report (DRP1) (ref. [34]). This document provides details of the baseline design for GDA step 2, outlining the physical system descriptions and requirements that form the design at that point in time.

## E3S case approach and structure

1. Rolls-Royce SMR Limited has chosen to develop its cases in a holistic manner, as an E3S case. The fundamental 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 case 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 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.

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**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. [29]), supplemented to include UK specific expectations and expanded to include the other E3S purposes.

## Summary of the requesting party’s E3S case for structural integrity

1. The Rolls-Royce SMR design demonstration of structural integrity for safety-classified metallic pressure boundary components and their supports is presented in the E3S case under Chapter 23 – Structural Integrity (ref. [8]). This is an additional chapter added by the RP to the IAEA approach in SSG-61 (ref. [29]), to meet UK context.
2. Chapter 23 contains Claim 23: “structural integrity of SSCs is justified and the risk of structural failure is minimised to ALARP”. Chapter 23 describes the requirements, approaches and methodologies necessary to substantiate the structural integrity of safety-classified metallic pressure boundary components and their supports. This includes lower pressure components (such as atmospheric tanks) and the reactor pressure vessel internals which support the reactor core. Structural integrity considers the confine fluid function of pressure retaining components.
3. Other functional requirements (other than structural integrity) are addressed under other chapters of the E3S case, which in accordance with the IAEA approach described in SSG-61 (ref. [29]), partition up the key SSCs as per the following, described by the RP as “engineering chapters”:

* Chapter 4: Reactor (Fuel and Core)
* Chapter 5: Reactor Coolant System & Associated Systems
* Chapter 6: Engineered Safety Features
* Chapter 7: Instrumentation & Control
* Chapter 8: Electrical Power
* Chapter 9a: Auxiliary Systems
* Chapter 9b: Civil Engineering Works and Structures
* Chapter 10: Steam & Power Conversion Systems
* Chapter 11: Management of Radioactive Waste

1. These chapters are used to identify those SSCs that fall within the scope of Claim 23, and require a structural integrity substantiation as part of the E3S case.
2. The key technical aspects of structural integrity covered by Chapter 23 are broadly grouped under three subclaims (SCs):

* SC 23.1: Achievement of Reliability
* SC 23.2: Demonstration of Reliability
* SC 23.3: Maintenance of Reliability

1. Each of these SCs are broken down further into technical arguments and reasoning to underpin the subclaim. This is typically provided through concise justifications, with reference to more substantial technical documents such as requirements, methodologies, approaches and technical basis providing rationale for the approach taken. The arguments also cite other sources of information considered, including relevant good practice (RGP), operational experience (OPEX), international guidance, nuclear design codes and relevant standards.
2. To accompany Chapter 23, the RP’s reasoning and approach for developing the claims, subclaims and arguments are provided in a Tier 2 document ‘Structural Integrity Requirements’ (ref. [35]).

### SC 23.1 - Achievement of reliability

1. This subclaim explains how achievement of high-quality design and manufacture comprises several further subclaims covering key structural integrity aspects such as component classification, use of appropriate design and construction codes and standards, structural assessment, materials selection, consideration of ageing and degradation mechanisms, and robust examination during manufacture. Additional provisions are given within the E3S case so that design and manufacture of components are optimised to ensure structural integrity risks are reduced ALARP.
2. The key aspects of this subclaim include:

* An approach of grading structural integrity controls for SSCs, based on the RP’s approach to safety categorisation and classification which is presented within a referenced document SMR0004589 Issue 1 ‘Categorisation and Classification Method’ (ref. [36]). This approach uses ‘failure modes and effects analysis’ to understand the complete consequences of gross failure of sub-components and regions, including accounting for any secondary consequences (internal hazards resulting from the initial failure). Two higher reliability classifications are proposed in (ref. [8]):
  + Very High Reliability (VHR) – where structural failure could lead to either an off-site release of dose exceeding 100 mSv or no physical barrier intact to confine any substantial relocation of radioactive material. The RP considers it is not reasonably practicable to provide control of the resulting conditions either within or beyond the design basis.
  + High Reliability (HR) – where structural failure could lead to relocation of radioactive material, but with off-site doses limited to less than 100 mSv. The RP considers it is not reasonably practicable to provide control of the resulting conditions within the design basis; however, it is reasonably practicable to provide beyond design basis defence.

Where the consequences of failure are demonstrated to be tolerable, this is considered within the design basis and safety classifications are applied as Safety Class1/2/3/Non-nuclear Classified (NC), dependant on the consequences of postulated failures and the level of protection considered necessary for the SSC.

* Components are designed to appropriate standards and requirements which provide suitable reliability. The RP has selected the 2021 edition of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III (ref. [32]) as the design basis for all nuclear Safety Class 1 and 2 pressure boundary components, and Safety Class 3 pressure boundary components, where the failure directly leads to the release of radioactivity above a defined threshold. ASME BPVC Section III (hereon ASME III) provides rules for materials, design, structural analysis, manufacturing, examination, over-pressure protection and quality assurance for nuclear components. For lower classification applications (safety class 3 pressure boundary components where the failure does not directly lead to the release of radioactivity above a defined threshold), conventional (non-nuclear) design codes are proposed. These are selected based on the technical provisions, the provenance and experience of the code, and its status in relevant legislation. Pressure Equipment Safety Regulations (PE(S)R) (ref. [37]) requirements will be met where either conventional standards are adopted or failure of the component would not lead to an emission of radioactivity.
* Components are designed for structural integrity, including a review of improvements in material forming, welding and examination and incorporating lessons learned from relevant OPEX. Systems engineering approaches ensure design is carried out in a systematic and structured way. This RP is applying a concept of “Design for X” (DfX), whereby working practices use integrated project teams (IPT), made up from a range of functions such as design, structural analysis, procurement, manufacturing, examination and verification. This approach is informed by a series of structural integrity specific design principles, which have been produced based on sources of RGP and OPEX compiled by the RP. The RP claims that using IPTs promotes cross-functional engagement through the design process, including transverse requirements from different functional areas.
* Structural assessments are carried out to show that the design code limitations are satisfied for the applicable loading conditions. Assessments are carried out against the design limits in the relevant design code using conservative inputs while accounting for relevant through-life degradation mechanisms.
* Materials are selected and specified to ensure they are well understood and characterised, based on OPEX and RGP. Materials selection is based on those allowed under ASME Sections II and III and a down-selection optioneering process has been followed for major decisions/components, considering legislative requirements, commercial factors and OPEX. Ageing management is also cited as a key consideration in the materials selection process.
* Components are fabricated and installed to appropriate standards using qualified and controlled processes. Welding will be carried out in accordance with written, approved and qualified procedures by appropriately qualified welders. Design for manufacture is also employed to ensure designs are optimised for manufacture. Independent inspections will be employed for process controls and a supplier selection process will be used to assess capability of suppliers to meet all expected requirements.
* Code-based examinations will be employed to provide adequate demonstration that manufacturing processes have been performed to an acceptable standard and within expected parameters set by ASME, which provides minimum requirements for examination. The RP is also considering an approach of using Ultrasonic Testing (UT) in lieu of Radiographic Testing (RT), using an ASME code case (N659-3) using ASME Code Case N-659-3 to provide an alternative to the use of radiography which is argued to have an appropriate capability for the code examination.
* Component pressure testing will be undertaken to provide assurance on the integrity of the pressure boundary and highlight flaws before the vessel enters service. This provides evidence of pressure vessel failures under hydrotest.
* Appropriate controls during design and manufacture are implemented to provide assurance of compliance with the relevant requirements. This approach will be taken to ensure appropriate quality assurance, which is largely based on the consequences of the activity being incorrectly performed, which is claimed to be consistent with IAEA guidance. For HR/VHR components, an Independent Third Party Inspection Agent (ITPIA) will be used to provide assurance that code (ASME III) and beyond code (VHR/HR) requirements are satisfied.

### SC 23.2 Demonstration of reliability

1. This subclaim details the RP’s approach for a demonstration of fracture avoidance, applied specifically for VHR/HR classified components, where an appropriate conceptual defence in depth argument is required. This requires additional material fracture toughness testing and assurance of examination capability, linked by conservative fracture analysis to provide a balanced Avoidance of Fracture Demonstration (AoFD). These are further decomposed into three key areas, as described below.

Objective-based manufacturing examinations are capable of reliably detecting defects of structural concern

1. Examinations will be designed to meet the specific objectives based on credible defects identified for the manufacturing process and their depth obtained from the Defect Tolerance Assessment (DTA). Robust demonstration of inspection assurance is provided by the European Network of Inspection Qualification (ENIQ)-based inspection qualification for VHR welds and capability statements for HR welds and HR/VHR forgings.
2. The RP claims that minimal changes will be required to the code examinations on forgings to demonstrate that objectives can be met.

Use of suitably conservative, lower bound fracture toughness values in the limiting defect size calculations

1. Sufficient Upper Shelf Fracture Toughness Testing (USFTT) is to be carried out on forgings and representative welds to provide confidence that the toughness values used in the DTA are suitably conservative lower bound. The USFTT requirements will be specified on prolongations of production forgings and representative welds, such as those in the welding qualification. Supplementary Charpy testing will be specified to better characterise the transition and upper shelf regime.

Conservative fracture mechanics analysis

1. The DTAs will be based on the R6 procedure which has substantial provenance and precedent for its application in the UK. A further target Reserve Factor (RF) will be applied on defect size to demonstrate margin, consistent with OPEX and RGP.
2. R6 is the accepted basis for analysis of safety critical structures and components in the UK, representing industry best practice. R6 has undergone extensive validation during its development and is actively managed and maintained with any substantial changes subject to expert peer review.
3. The target RF for VHR and HR components is 2. Sensitivity studies are carried out for inputs with variation or uncertainty where bounding values have not been used. This demonstrates tolerance to credible variations in those inputs and an absence of cliff-edge effects.

### SC 23.3 Maintenance of reliability

1. The RP claims that reliability can be maintained with the provision of effective in-service systems and procedures. It argues that reliability is maintained through pro-active planning in the design phase and through monitoring of information through-life. Ageing management of materials is managed through identification and consideration in the design phase, with a programme in place for irradiation surveillance of RPV materials.

In-Service Inspection (ISI) in accordance with ASME Section XI provides a forewarning of failure, through examinations with demonstrated capability. Plant monitoring systems are provided to detect the presence of loose parts, leaks, pressure and temperature transients, deviations from the specified chemistry regime and seismic activity, with digital twins employed to maximise the use of available data. The case presents three further subclaims to help illustrate how essentially forewarning of failure is to be demonstrated through life, which are described below.

Ageing and degradation of materials is actively managed and monitored where appropriate

1. This approach details how ageing degradation mechanisms will be systematically identified during the design stage, based on OPEX and RGP, and individually justified. The RP’s approach uses an Ageing Management Plan (AMP) which outlines the proactive approach taken to identifying potential threats. It identifies these based on industry best-practice such as information from Electric Power Research Institute (EPRI) and guidance in ASME III (Appendix W). An irradiation surveillance programme will be implemented to measure changes to fracture toughness and mechanical properties in the beltline region of the RPV throughout the plant life, and validate the irradiation embrittlement model and analysis employed.

In-service inspection will detect defects that may propagate to a critical size

1. The RP claims that application of ASME XI requirements will be suitable and effective for ISI of nuclear plant components and that additional assurance may be provided for higher reliability components.
2. The E3S case places a commitment on the future duty holder/licensee/permit holder to this extent, such that in-service inspections should be carried out in accordance with ASME XI, Division 1. Pre-service Inspection (PSI) will be carried out using equivalent procedures and equipment as for the ISI, to baseline the inspections. The RP notes that Risk-Informed ISI may be adopted, with further substantiation to be provided in the future.
3. The RP states that the components are designed to provide adequate provision for any other alternative assurance (for example, ENIQ inspection qualification).

Plant monitoring will provide forewarning of failure

1. The RP’s E3S case details how monitoring methods will confirm the absence of unacceptable degradation mechanisms and provide sufficient forewarning of potential deviations from design intent. Sufficient monitoring should be in place to enable the use of a digital twin approach. It will include monitoring of; transient temperature and pressure, reactor coolant chemistry, leakage, loose parts and vibration.

### Demonstration of reducing risk to ALARP

1. The E3S case structure used by the RP consolidates its approach to demonstrating risk is reduced ALARP under Chapter 24. A brief note is included in Chapter 23 that particular focus is placed on the structural integrity requirements for HR and VHR components to ensure that overall risks are reduced to ALARP.

## Basis of assessment: Requesting Party’s documentation

1. The principal documents that have formed the basis of my structural integrity assessment of the E3S case are:

* Chapter 3: Objectives and Design Rules for SSCs (ref. [2])
* Chapter 5: Reactor Coolant System & Associated Systems (ref. [3])
* Chapter 6: Engineered Safety Features (ref. [4])
* Chapter 9B: Civil Engineering Works and Structures (ref. [5])
* Chapter 15: Safety Analysis (ref. [6])
* Chapter 23: Structural Integrity (ref. [8])
* Environment, Safety, Security and Safeguards Design Principles (ref. [38])
* E3S and Safeguards Categorisation and Classification Method (ref. [36])
* Key Structural Integrity Processes for GDA Submission (ref. [39])
* Structural Integrity Requirements (SIR) (ref. [35])
* Defect Tolerance Assessment Guide (DTAG) (ref. [40])
* Defect Tolerance Assessment Guide (DTAG) Technical Basis (TB) (ref. [41])
* ASME III Design by Stress Analysis Guide (DBSAG) (ref. [42])
* Non-Destructive Examination Framework (ref. [43])
* Rolls-Royce SMR Ageing Management Plan (ref. [44])
* RPV Body Component Substantiation Report (CSR) (ref. [45])
* Safety Measure Design Description for the Containment [JM01] Safety Measure (ref. [46])
* Containment Structure [JMA] Design Definition (ref. [47])

# ONR assessment

## Assessment strategy

1. In GDA Step 1, I drafted my Step 2 Assessment Plan (ref. [15]), identifying a number of key technical aspects for structural integrity, as described in ONR guidance (ref. [11] [12]). These technical topics included:

* The RP’s structural integrity safety case strategy and development of claims for SSCs, to show traceability from safety functional requirements through to structural integrity controls.
* Consideration of structural integrity to reduce risk ALARP; including decision making for classification of SSCs, selection of design codes and standards and avoidance of fracture demonstration for highest reliability claims.
* Provisions to account for structural integrity during design, manufacture, construction and through life; including multidiscipline engagement for SSC design, plant layout considerations, component access for inspectability, monitoring and surveillance and consequence of failure analysis.
* Materials selection methodology to inform design limits, operating conditions and management of through life ageing and degradation mechanisms.

1. In addition to the fundamental structural integrity technical aspects, I also identified several aspects that I considered likely to require further understanding at a principles level. These were related to a number of specific Rolls-Royce SMR design features, for which I have engaged with ONR specialists other than structural integrity throughout Step 2. In summary, these were identified as:

* Potential impact of having a compact design on the capacity to install in-core irradiation embrittlement surveillance specimen capsules.
* Understanding and demonstration of reducing risk ALARP through materials performance and component design within a non-boron potassium hydroxide primary circuit chemistry.
* Design considerations for demonstrating structural integrity reliability of key nuclear safety significant SSCs, with respect to the seismic isolation system.

1. At the time of my Step 2 assessment, the maturity of Rolls-Royce SMR design is specified by the RP as being at DRP1, which is defined within the ‘GDA Design Reference Report’ (ref. [34]). I have reviewed this report and am satisfied that the documents of relevance to the scope of my structural integrity assessment that are referenced within (ref. [34]) align with the key E3S case documents I have sampled.
2. My Step 2 assessment has involved regular engagements with the RP’s structural integrity specialists. During these technical exchanges, I have gained a better understanding of the RP’s E3S case structural integrity claim and the broader Rolls-Royce SMR design. I have raised several Regulatory Questions (RQs) to inform my assessment, which are referenced accordingly.
3. In accordance with ONR guidance on assessment of safety cases (ref. [25]), I have used a sampling approach to identify those areas that I consider nuclear risk to be most significant. Within Step 2, my sampling has been broad, focusing on the RP’s high level methodologies, approaches and strategies for ensuring structural integrity risk to SSCs is reduced ALARP.

## Assessment

### Structural integrity safety case strategy and development of claims for SSCs

1. ONR expectations contained within the EMC series of the SAPs define what should be within scope of a structural integrity safety claim. This includes guidance on the engineering assessment of the integrity of metallic components and structures, which includes pressure vessels, boilers, pressure parts, coolant circuits, pipework, core support, pumps, valves, storage tanks and the freestanding metal shell of pressure retaining containment structures. The RP’s consideration of structural integrity for the Rolls-Royce SMR design is contained in the E3S case under Chapter 23. My assessment compared the scope of Chapter 23 against ONR expectations.

#### Scope and development of structural integrity claims and subclaims

1. Chapter 23 is defined by the RP as addressing claims associated with the substantiation of the structural integrity of safety-classified metallic pressure boundary components and associated support functions. This includes lower pressure components (such as atmospheric tanks) and the core support function and control rod alignment provided by the reactor pressure vessel (RPV) internals. The scope explicitly states that concrete structures and steel building structures are excluded from Chapter 23. The RP confirmed through RQ-01084 (ref. [48]) that the integrity of the containment vessel structure is in-scope of Chapter 23, as a pressure boundary component and is not excluded as a steel building structure. Based on this response, I am satisfied that the scope of Chapter 23 SSCs broadly aligns with the scope of ONR expectations for integrity of metal components and structures.
2. As described in Section 3 above, Claim 23 is decomposed into three SCs to justify how SSC reliability is; achieved (SC 23.1); demonstrated (SC 23.2); and maintained (SC 23.3). These SCs summarise and consolidate a set of underpinning requirements presented in an accompanying document ‘Structural Integrity Requirements’ (SIR) (ref. [35]). The SIR captures the relevant technical requirements aligned to Chapter 23, decomposing them to a greater level of detail. These set the technical requirements for each component that are embedded in the component design specification, which applies internally (within the RP’s organisation) and externally (supply chain).
3. In my opinion, the high level scope and content of structural integrity aspects considered within the SIR are well aligned with ONR expectations, as set out under EMC.1 to EMC.34. I consider the RP’s approach of adopting IAEA SSG-61 (ref. [29]) as a baseline position, with an additional chapter to capture structural integrity provisions, is reasonable. It is important to understand how the SIR, in their broadest terms, have been derived for the Rolls-Royce SMR design and used to influence the design of SSC important for safety. I therefore targeted SC 23.1.3 ‘Components are designed for structural integrity, including a review of OPEX and RGP’, to sample further.
4. This SC makes reference to a series of fundamental design principles established for the Rolls-Royce SMR design. These design principles are referenced from Chapter 3 of the E3S case, which are contained within the Tier 2 document ‘Rolls-Royce SMR Environment, Safety, Security and Safeguards Design Principles’ (ref. [38]). I sampled this document, from which I was able to identify ten design principles explicitly linked to structural integrity. The principles closely align with ONR guidance provided in the SAPs EMC series, with the source of OPEX referenced clearly as being ONR and IAEA guidance. I am satisfied that the RP’s approach broadly considers ONR expectations for a structural integrity safety case as a key input to SSC design, with capacity to inform the decision making process.
5. Whilst I consider this level of detail to be reasonable for a Step 2 assessment, I expect further evidence to be presented within the E3S case for Step 3 to demonstrate how the structural integrity design principles established from RGP and OPEX have been used to inform the design of different component classifications, such that risk of structural integrity failure is reduced ALARP. I consider this to be a residual matter.

#### Derivation of structural integrity claims from safety functional requirements

1. ONR expects safety case claims to be complete, coherent and traceable. One way to achieve this is with a clear trail from claims through arguments to evidence (ref. [11]). The rigour of the case presented should be proportionate to the importance of the SSC to nuclear safety and written for the user(s) (NS-TAST-GD-051; The Purpose, Scope, and Content of Safety Cases; ref. [25]). ONR guidance (see SAPs paragraphs 407 and 643, ref. [11] and [24]) also identifies that the safety functional requirements of SSCs should be identified from the fault schedule and an appropriate safety classification determined in accordance with principles ECS.1, ECS.2 (and associated paragraphs).
2. With this guidance in mind, I sampled the categorisation of functions (ECS.1) and the safety classification of SSCs (ECS.2) to understand how it is used to inform the design of plant, from a structural integrity perspective. Chapter 23, Subclaim 23.1.1 explicitly defines safety classification of components as an input for determining the structural integrity controls for SSCs, noting that the RP’s approach accounts for components of the highest reliability, which is a key consideration for structural integrity. The RP’s approach for safety categorisation and classification of components (ref. [36]) is presented in Chapter 3: ‘Objectives and Design Rules’ (ref. [2]).
3. From here, I was able to identify the Fundamental Safety Functions (FSF) which are further decomposed into High Level Safety Functions (HLSF) and presented in the Fault Schedule (E3S Chapter 15: Safety Analysis (ref. [6]), along with required safety measures assigned to deliver them. HLSFs are assigned a safety category, which is then used to classify the SSC that deliver the HLSFs. From my sample, I was satisfied that an E3S case user could clearly establish the traceability of structural integrity controls to safety functional requirements.
4. I assessed where the substantiation and evidence to underpin structural integrity claims were presented for specific SSCs within the E3S case. Whilst Chapter 23 provides the claims and arguments for the generic Rolls-Royce SMR design structural integrity approach, the evidence for specific SSCs is provided in the broader E3S case safety demonstration. This is discussed within Chapter 23 as being contained in ‘Component Substantiation Reports’ (CSRs), the purpose of which is to compile the component-specific structural integrity controls, as per the three Chapter 23 subclaims. This is reiterated in accompanying SIRs, where CSRs are described as “consolidating the evidence to underpin Claim 23 for individual components or groups of components”.
5. Within Step 2, the E3S case is not fully developed across all three tiers to a state of maturity where detailed, component-specific evidence is available and embedded to substantiate the specific SSCs structural integrity controls. It was, therefore, not immediately clear where in the E3S case structure the CSRs will be provided. I raised RQ-01084 (ref. [48]) to seek further information.
6. The RP’s response identified that the CSRs link the engineering chapters (defined by the RP as Chapters 4 – 11, inclusive) and Chapter 23, with direct referencing to evidence at a component-specific level. By way of example, the Reactor Pressure Vessel (RPV) CSR was submitted by the RP (ref. [45]), to show typical CSR content and scope. I considered this CSR to be relevant and useful to support my Step 2 assessment with respect to the scope and content of the structural integrity E3S case.
7. From the RPV CSR, I was able to identify clear alignment with the Chapter 23 subclaims. I considered the content was comprehensive, capturing the key aspects for structural integrity substantiation, enabling the reader to highlight key structural integrity controls with links to more detailed technical evidence contained within subordinate documents.
8. From my sample, I am satisfied that the RP’s E3S case structure, through the use of CSRs, enables users to identify how structural integrity controls have been informed by the Rolls-Royce SMR design FSFs, following an embedded process for categorisation and classification of components. I expect evidence of how this process is implemented to be presented within the E3S case for several SSCs with varying importance for nuclear safety, which I will review when the relevant documents are available in Step 3.

### Structural integrity considerations to reduce risk ALARP

1. The starting point of a structural integrity assessment is the categorisation of functions and safety classification of SSCs. ONR SAPs ECS.1 to ECS.3 (ref. [11]) identify that categorisation and classification of SSCs important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate codes and standards. I sampled the RP’s approach to determine how structural integrity provisions inform/are informed by the overall categorisation and classification of Rolls-Royce SMR design SSCs. I have targeted how the classification method addresses circumstances where the consequences of gross failure are intolerable (highest reliability).

#### Classification of SSCs to inform structural integrity requirements

1. Subclaim 23.1.1 identifies that “structural integrity measures are invoked based on the component’s role in protecting people and the environment”. This involves taking a graded, top down approach, with further consideration for refinement of classification for sub-components.
2. Four criteria are considered by the RP to establish the structural integrity controls applied for SSCs. These criteria include derived safety classification, radiological consequences of failure (for lower class components), design code classification and seismic performance classification. Of these four criteria, I considered that safety classification and design code classification to be important areas for further sampling, to determine how highest reliability claims are derived.

Safety classification

1. In the E3S case, the RP’s approach to safety classification is referenced primarily within Chapter 3, which claims that “suitable design principles and associated methods, approaches, and requirements are established for the Rolls-Royce SMR design to achieve the E3S fundamental objective”. Further details are referenced in a supporting Tier 2 document ‘SMR0004589 Issue 1 Categorisation and Classification Method’ (ref. [36]).
2. According to SC 23.1.1, following initial safety classification, a further process of Failure Modes and Effects Analysis (FMEA) is undertaken to understand the complete consequences of gross failure of sub-components and regions, including accounting for secondary consequences (internal hazards generated as a result of failure). I am satisfied at a high level the RP’s approach is aligned with ONR expectations by considering the more conservative position of assessing the consequences of gross failure, including indirect consequences, and does not rely on a leak before break approach as a primary safety claim.
3. Higher reliability classifications are defined in SC 23.1.1 and (ref. [36]) for instances where the consequences of failure are unacceptable and beyond-code structural integrity measures are required. The E3S case states that in exceptional cases, it is not reasonably practicable to provide control of the conditions resulting from structural failure of a component within the design basis (the highest classification being Safety Class 1). In such cases, conceptual defence in depth is provided through assignment of a safety classification that goes beyond the normal level of reliability demonstration for Safety Class 1. As described in Section 3 above, these are defined as VHR and HR, with the structural integrity controls applied for these classifications being proportionate to the predicted consequences and subsequent radiological release as a result of gross failure. The following consequence criteria are used to define the classifications:

* VHR is applied where structural failure could lead to either an off-site release of dose exceeding 100 mSv or no physical barrier intact to confine any substantial relocation of radioactive material. The RP considers it is not reasonably practicable to provide control of the resulting conditions either within or beyond the design basis.
* HR is applied where structural failure could lead to relocation of radioactive material, but with off-site doses limited to less than 100 mSv. The RP considers it is not reasonably practicable to provide control of the resulting conditions within the design basis; however, it is reasonably practicable to provide beyond design basis defence. In the late stages of Step 2, the RP revised the definition of HR provided in (ref. [8]), removing a radiological consequence threshold of ‘exceeding 500 mSv on-site release’. Whilst I do not consider this change affects the structural integrity substantiation or provisions that I have assessed for the E3S case so far, this late change in the HR definition will require further review in Step 3 to confirm it is consistent with other aspects of the E3S case.

1. It is clear from the above that the differentiation between VHR and HR is determined by consequence analyses to understand the significance of the fault identified. Failure consequence analyses and the consideration of measures to protect, prevent or mitigate the consequences of failure are considered by the ONR fault studies assessment and internal hazards disciplines, whereby the direct and indirect consequences of failure (respectively) are needed to inform the safety classification of SSCs. This will be dependant on the decision making process that has resulted in the designation of VHR/HR, which should demonstrate that risk is reduced ALARP.
2. For VHR/HR claims, the RP applies the principle of conceptual Defence in Depth, through assignment of a more rigorous safety case, in terms of engineering substantiation, manufacturing controls, inspection, testing, quality assurance and through-life management.
3. I generally consider the RP’s classification of VHR to be akin to ONR expectations for highest reliability (SAP EMC.1 to EMC.3, ref. [11]), for which measures over and above the provisions of a recognised nuclear design and construction code should be demonstrated. Based on the importance for nuclear safety, I sampled the approach to highest reliability classifications, both the determination of classification and the structural integrity controls proposed to substantiate it. The SIR provides further guidance on the additional measures that are required for VHR/HR components. The technical aspects of structural integrity controls proposed for VHR/HR components are assessed further in Section 4.2.3 below, therefore this part of my assessment is based on the RP’s initial identification of VHR/HR components.
4. At this stage of GDA, the RP is still refining the design of SSCs, meaning some key consequence analyses are not yet complete. The RP has however completed preliminary component analyses, and under SC 23.1.1, has provided a list of potential components containing VHR/HR classifications. These include:

* RPV body and closure head
* Pressuriser shell
* SG primary head, tubesheet and secondary shell
* RCP casing
* Reactor coolant loop pipework
* Main steam line pipework
* Main steam isolation valve body

1. In my opinion, Rolls-Royce SMR design list of preliminary VHR/HR components is reasonable and aligned with precedent from previous GDAs and the UK’s only operating civil PWR at Sizewell B. The final classification of Rolls-Royce SMR design components will depend on the finalised consequence analyses, therefore will require further structural integrity review when more detailed evidence is available later in Step 3 of GDA.
2. I note from SC 23.1.1 that sections of the Main Steam Line pipework and main steam isolation valves are classified as VHR/HR on a preliminary basis, based on the RP’s early consequence analyses. It is currently unclear as to whether the scope of these consequence analyses include other in-line valves (bodies and their potential missiles) of potentially lower classifications, but that may still form part of the main steam line.
3. Specific details on the manufacture of major safety significant valves for the Rolls-Royce SMR plant have not been presented within Step 2. It should be noted that if cast valve bodies are proposed, previous GDA experience has shown that demonstration of the structural integrity of any valve bodies manufactured by casting may prove problematic for the AoFD. Potentially low fracture toughness values may lead to small but structurally significant defect sizes, which are difficult to detect and reject with qualified inspection. Cast components also often require repair welds which, if of significant size, can make the defect tolerance argument difficult. I expect substantiation of the component classification to be presented as evidence within the E3S case when it is further developed within Step 3 of GDA.
4. The RP’s preliminary list of VHR and HR components is important to ONR’s fault studies assessment, since VHR and HR components are discounted from deterministic analysis. Similarly, the need to consider the indirect consequences within the internal hazards discipline is crucial to establishing the plant layout and structural integrity classifications. I have engaged with the ONR fault studies and internal hazards specialists throughout my Step 2 assessment, which has confirmed the RP’s position that important consequences analyses regarding plant layout, fault sequences and severe accidents needs to be undertaken to support SSC classification.
5. At a high level, I am satisfied that the RP’s approach to inform structural integrity controls based on safety classification aligns with the ONR expectation (ref. [24]) that good design practice is informed by a rigorous consideration of the consequences (direct and indirect) of postulated failure, with gross failure used as the limiting case. However, at this stage of the GDA, the necessary level of consequence analyses has not yet been undertaken to substantiate and confirm component-level classification.
6. I expect further evidence to be presented within the E3S case, to substantiate the RP’s safety classification approach as an important input to informing structural integrity controls. To achieve this, I expect evidence to be provided within the E3S case to demonstrate how safety functional requirements relate to structural integrity controls, including how VHR and HR classifications have been applied and whether all reasonably practicable options to reduce reliance on the structural integrity case have been exhausted, such that the VHR/HR claim is truly warranted. I will review this in Step 3 when more detailed, component-specific evidence is available.

Determination of code classification to inform structural integrity

1. With safety classification of components assigned, it is necessary to establish the corresponding structural integrity controls and requirements.
2. Under SC 23.1.1, the RP claims that a nuclear pressure vessel code is used to establish the rules for design, structural analysis, materials, manufacturing and quality assurance for nuclear components. This is initially determined by the assignment of a code classification, which for the Rolls-Royce SMR design is the American Society for Mechanical Engineers Section III (ASME III) Boiler and Pressure Vessel Code (BPVC) 2021. The adequacy of this code for meeting ONR expectations is considered in Section 4.2.2.2 below.
3. The RP’s process for assignment of ASME III code class to components follows the United States Nuclear Regulatory Commission (USNRC) Regulatory Guidance 1.26, Revision 5 (ref. [49]). This approach assigns a quality group to components that deliver a predefined and prescriptive set of safety functions. These quality groups range from A – D, with Group A being for the most safety significant components and Group D being the least. These groups define directly what ASME III code class should be used, such that Group A components are designated as ASME III Code Class 1, Group B, Class 2 and so on.
4. The RP has developed a process (C3.2.2-5 ‘Establish ASME III Code Class’, (ref. [39]), whereby the USNRC Regulatory Guide 1.26 is used to initially determine SSC code classification. Additional rules are then applied to ensure that the consequence-led safety classification approach is accounted for in determining the final assigned ASME III code classification. These rules include:

* The ASME III code class is not higher than the safety class, so that the process does not escalate the design rules.
* VHR/HR components are always designed to ASME III Code Class 1 rules (noting this is a start position and that additional provisions are applied as per SC 23.2).
* ASME III code class does not reduce the classification where it has been elevated due to secondary consequences.
* Where a quality grade is not provided by USNRC Regulatory Guide 1.26, the safety class is used as a default position.

1. In my opinion, the RP’s use of the USNRC Regulatory Guide 1.26 as a starting point to determine component design code classification is reasonable, through adoption of internationally accepted approaches. The RP’s approach for safety classification to set the ultimate design and operation requirements enables highest reliability claims to be implemented, based on the consequences of failure specific for the Rolls-Royce SMR design. I am therefore satisfied that this meets ONR expectation that structural integrity design requirements are defined based on SSC safety classification (ref. [24]). In my opinion, this approach generally aligns with Requirement 22 of the IAEA SSR2/1 (Ref. [28]) for safety classification and SSG-56 that the safety classification is used to inform the engineering design, with manufacturing rules applicable to each individual component selected with due account taken of the consequences of failure, in terms of both the fulfilment of the safety function and the prevention of a radioactive release (Ref. [30]). I am also satisfied that the RP’s approach to classification satisfies the structural integrity aspects of Issue G of the WENRA reference safety levels and safety objectives for new plants (ref. [21] and [22]) on safety classification of SSCs, that SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification.
2. From the information I have sampled within Step 2, I consider the RP’s approach to invoke structural integrity measures based on the component’s role for nuclear safety is reasonable, at a high level. The RP’s approach combines international guidance, OPEX and ONR expectations for safety classification informed by direct and indirect consequences of gross failure. In my opinion, this is satisfactory for the purposes of Step 2 of GDA.
3. Whilst I consider the RP’s approach to be reasonable at a high level, I consider it is important to confirm that cross discipline engagement has been maintained and that the safety case claims are substantiated, with the outcomes of SSC classification decisions recorded, visible and fully implemented across the E3S case.
4. I expect the RP to provide further evidence to support the RP’s classification approach at a component/subcomponent level. This is to demonstrate how the approach has been applied for a variety of relevant Rolls-Royce SMR design components and subcomponents, and how it meets ONR structural integrity expectations. I consider this to be a residual matter.

#### Application of relevant codes and standards

1. ONR guidance on ‘Safety Standards’ (SAP ECS.3) identifies that components and structures important to safety should be designed, manufactured, installed, examined and inspected using codes, specifications and standards commensurate with their safety classification. As such, the starting point for design is compliance with relevant national and international codes and standards. In addition, depending on the nuclear safety significance, safety case claims for the structural integrity of SSCs may require further substantiation (ref. [24]). Selection and implementation of appropriate design, manufacturing standards and inspection provisions for SSCs is key to demonstrating the risks of failure are reduced to ALARP.
2. Information provided in SC 23.1.2 states that “components are designed to appropriate standards and requirements which provide suitable reliability”. Components assigned Safety Class 2 and above adopt nuclear-specific design codes, and for Safety Class 3, either nuclear-specific or conventional design codes may be used. The determination between nuclear or non-nuclear codes for Class 3 is based on the radiological consequences of component failure, which is a criteria defined under the RP’s classification process (See Section 4.2.2.1 above).
3. The design requirements set by the ASME III (selected for the Rolls-Royce SMR design – see para. 97) have been evaluated by ONR in previous GDAs, namely the AP1000 (ref. [50]), the UK ABWR (ref. [51]) and the UK HPR1000 (ref. [52]). ASME III is also the design basis for the UK’s only operating PWR at Sizewell B. These were judged as an acceptable baseline position to demonstrate structural integrity of components important for safety. With this precedent, I am satisfied with the RP’s proposed use of ASME III as a baseline for the structural integrity demonstration of the Rolls-Royce SMR design SSCs. In my opinion, this approach broadly aligns with Requirement 9 of the IAEA SSR2/1 (Ref. [28]) for the demonstration of proven engineering practices and SSG-56 for the use of codes and standards as part of the design basis of the reactor coolant system and associated systems (Ref. [30])
4. In some cases, the RP has proactively identified that ASME III specifies requirements relating to its implementation, operation or assurance with institutions, federal laws and practices which are applicable or common in North America. As a result, there is a restricted supply chain of ASME III certified organisations and personnel with the required qualification outside of North America, meaning it may restrict its application outside of this region. The RP’s approach permits adaptations to ASME III requirements, where it can be demonstrated as not reasonably practicable or unnecessarily prohibitive to apply, in light of alternative arrangements that can be demonstrated as being an equal or higher level of quality demonstration.
5. ONR guidance states that “any deviation from the code should be justified since design codes are developed with implicit safety factors and assume minimum material properties and quality of fabrication”(ref. [24]). The details of the RP’s process, as well as a list of circumstances where this approach is being invoked, is contained in the E3S case under document ‘Rolls-Royce SMR Permitted Adaptations to ASME Boiler and Pressure Vessel Code Section III’ (ref. [53]). At a high level, I consider the RP’s approach aligns with ONR expectations for Step 2 of GDA, on the basis that a component specific substantiation will be produced to demonstrate risks are reduced ALARP. At this stage of GDA, component specific level of evidence has not been submitted for assessment, therefore I expect evidence of instances where ASME III adaptations are being considered by the RP to be included within the E3S case, to ensure it aligns with ONR expectations for substantiation of why this approach reduces risk ALARP. I will follow this up in Step 3 when component-specific evidence is available for assessment.

#### Approach to structural assessment

1. ONR guidance sets the expectation that the schedule of design loadings (including combinations of loadings) for components and structures, together with conservative estimates of their frequency of occurrence should be used as the basis for design against normal operation, fault and accident conditions (EMC.7, ref. [11]). All loadings for operation, credible faults, accident conditions and tests should be identified and the magnitudes specified (ref. [24]). I sampled the RP’s approach to understand how it has considered ONR expectations.
2. The E3S case has considered structural analyses within SC 23.1.4 “structural assessments are carried out to show that the design code limitations are satisfied for the applicable loading conditions”. The RP’s approach describes the factors considered within its structural assessments, including design load cases defined in Chapter 15: Safety Analysis (ref. [6]), and loading specifications and limits established from the selected design code (ASME III, ref. [32]). The RP’s design by analysis requirements (ref. [42]) includes a high level overview of the fatigue analysis approach, which is based on ASME III.
3. Information to substantiate these approaches are provided in the SIR (ref. [35]), with more detailed information contained within the ‘Fatigue Assessment Method and Technical Basis’ document (ref. [54]) which provides evidence related to fatigue analysis and consideration of Environmentally-Assisted Fatigue (EAF). This is a key ageing and degradation mechanism identified under the RP’s ageing management plan (ref. [44], see section 4.2.5.2 below). I note that the RP’s proposed primary circuit coolant chemistry regime is likely to influence the level of demonstration required to substantiate the impact of EAF in the design (discussed in section 4.2.5.2, below), which I expect to be reflected in the E3S case.
4. At the fundamental level of assessment required for Step 2, I am satisfied that the RP’s approach for achievement of reliability under SC 23.1 is broadly aligned with ONR expectations for structural assessment of design loadings (EMC.7). Given the importance of robust analysis to underpin claims of structural integrity, I expect further evidence such as detailed stress analysis, fatigue analysis and fundamental code compliance, to be presented as part of the E3S case to support component-specific claims. I will follow this up in Step 3 when component specific evidence is available for assessment.
5. I note that at this early stage in GDA, the RP’s design approach has accounted for aspects that may challenge structural assessments, such as the consideration of EAF as an influence on component integrity, which is based on regulatory guidance from the USNRC. In my opinion, there are a number of aspects in the RP’s approach that require further consideration. These are associated with the RP’s proposal to apply adaptations to USNRC regulatory guidance to reduce over-conservatism, as well as the significance of any gaps in the use of this OPEX related to the specific Rolls-Royce SMR design primary circuit chemistry regime. These aspects will require further scrutiny to substantiate how this approach reduces risk ALARP, which I consider to be a residual matter.

#### Consideration of structural integrity for non-nuclear pressure equipment legislation for the design of SSCs

1. The scope of an ONR structural integrity assessment includes pressure vessels, boilers and pressure parts, for which within the UK there is specific legislation to regulate the safety of pressure systems, notably the Pressure Equipment Safety Regulations (PE(S)R) (ref. [37]) and Pressure System Safety Regulations (PSSR) (ref. [55]).
2. ONR guidance relevant to the assessment of pressure systems safety on nuclear sites is provided in (ref. [26]) and (ref. [24]). Whilst consideration of non-nuclear health and safety aspects of the E3S case are typically managed by the ONR’s conventional health and safety (CHS) topic, there is specific guidance in NS-TAST-GD-067 (ref. [26]) related to circumstances where pressure equipment for nuclear use may require additional safety measures. In short, the PE(S)R excludes items “specifically designed for nuclear use, failure of which may cause an emission of radioactivity”. The exclusion recognises that the regulations do not cover the hazards associated with radioactivity and applying PE(S)R to nuclear pressure systems could result in equipment that is not adequate for nuclear use, albeit that it is adequate for non-nuclear applications.
3. This exclusion has been identified in previous GDA projects as an area for consideration within structural integrity assessment for nuclear pressure vessels, to ensure RP’s are fully aware of ONR expectations and that the RP’s approach reduces risk ALARP. I therefore considered it pertinent to sample this further, to understand whether it has been considered in the design and safety demonstration of the Rolls-Royce SMR design.
4. Subclaim 23.1.2 claims that components are designed to appropriate standards and requirements which provide suitable reliability. To demonstrate this, the RP states “for lower classification applications, conventional design codes are selected based on the technical provisions, the provenance and experience of the code, and its status in relevant legislation. The (PE(S)R) are applied where either conventional standards are adopted or failure of the component would not lead to an emission of radioactivity.”
5. The RP’s approach to design code selection for pressure retaining components identifies that conventional design codes can be applied to safety class 3 components, where failure does not directly lead to a release of dose < 0.1mSv on-site, and < 0.01mSv off-site. Where the consequences of failure result in a direct release of dose above these limits, the RP claims that either ASME III, or conventional design codes supplemented to give equivalent technical requirements to ASME III, will be used.
6. For the purposes of Step 2, I note that the RP has recognised the need to meet PE(S)R and has provided information in the E3S case as to when this will be applied. In principle, this approach indicates that pressure equipment associated with the Rolls-Royce SMR design will either meet PE(S)R or ASME III requirements (or equivalent). This goes some way to providing confidence that the Rolls-Royce SMR pressure vessels will be subject to a reasonable level of design and manufacturing controls as a result of being located on a nuclear site or for nuclear use.
7. I do however consider that the high level information supplied is ambiguous for circumstances of when the PE(S)R is applied, which does not clearly address the complexity of how the RP has interpreted PE(S)R and potential implementation of nuclear exclusion.
8. For example, the RP states that PE(S)R will be applied where failure of the component would “not lead to an emission of radioactivity”. This suggests that the nuclear exclusion is applied for instances where component failure may result in an emission of radioactivity. An emission of radioactivity can be interpreted as either the failure of the item directly, or by a chain of reasonably foreseeable events.
9. The use of "may cause" in this instance enables a wider interpretation to be applied. Hence, if failure of a pressure system can foreseeably lead to a release of radioactivity (directly or indirectly), then it may come within the exclusion. The RP’s design code selection criteria for lower classification (safety class 3) components is defined by whether failure directly leads to release of dose, whereas the PE(S)R exclusion applies to components that may cause an emission of radioactivity. It is therefore unclear how the RP’s approach will determine code selection for a component that may indirectly lead to an emission of radioactivity.
10. I expect sufficient evidence to be presented within the E3S case to demonstrate how the RP’s approach has considered key aspects of the PE(S)R nuclear exclusion, to ensure appropriate design codes are selected based on the full radiological consequences of failure. In my opinion, this evidence should include:

* how the RP has interpreted the use of PE(S)R and nuclear exclusion, particularly with deriving the approach of applying PE(S)R to components for which failure would not lead to an emission of radioactivity;
* how the design codes are selected for Rolls-Royce SMR design components that may indirectly lead to an emission of radioactivity;
* how the dose release thresholds of 0.1mSv on-site, and 0.01mSv off-site have been derived, including how they relate to the broader PE(S)R description of “emission of radioactivity”; and
* the RPs understanding of how ASME III NCD (Class 3) design requirements compare with PE(S)R requirements, with respect to identification and resolution of any gaps or significant variations.

1. I will follow this up in Step 3 of GDA to understand how the RP’s approach meets ONR expectations.

### Avoidance of fracture demonstration for highest reliability claims

1. ONR’s expectation for highest reliability components is that the component or structure should be as defect free as possible and is demonstrated to be tolerant of defects (ONR SAP EMC.1). In particular, the limiting defect size needs to be shown to be larger than the defect size that can be reliably detected by the applied examination techniques. This can be provided through an avoidance of fracture demonstration (AoFD) and Reserve Factor (RF) indicating the level of safety margin that should be achieved.
2. The AoFD involves the integration of detailed fracture mechanics-based Defect Tolerance Assessments (DTAs), using verifiable bounding material properties, to determine the limiting defect sizes for these components at the start of life, taking into account any potential for through-life crack growth. The non-destructive examinations being proposed for the components should be able to reliably detect such start of life defects by a suitable margin (SAP EMC.28 and EMC.3, ref. [11]). This demonstration is referred to as “beyond design code” compliance, for which some key examples are presented within SC 23.2. It is noted that other beyond code requirements are detailed elsewhere within Claim 23, for example, SC 23.1.5 requires that “for VHR/HR SA-508M Grade 3 forging specifications, additional limits are imposed on the chemical composition against a number of trace impurity elements to improve weldability, toughness and material degradation”.
3. Subclaim 23.2 states that reliability is demonstrated for VHR/HR components through a robust avoidance of fracture case, with VHR and HR components defined under SC 23.1.1 (see Section 4.2.2.1, above). SC 23.2 defines the RP’s approach to AoFD, based on the following three main inputs:

* SC 23.2.1: Objective-based manufacturing examinations are capable of reliably detecting defects of structural concern
* SC 23.2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound
* SC 23.2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern

1. The E3S case provides a high level overview of how each aspect will be applied to VHR/HR components, with further guidance provided in the SIR. I have sampled these accordingly with my findings presented below.

#### Subclaim 23.2.1: Objective-based manufacturing examinations are capable of reliably detecting defects of structural concern

1. This subclaim seeks to demonstrate that the manufacturing inspections proposed are designed to detect manufacturing defects of concern and that assurance of these inspections is provided by ENIQ-based qualification for VHR welds and capability statements for HR welds and VHR/HR forgings. Although this claim focuses on objective-based inspections, which I deem appropriate for meeting beyond code expectations for highest reliability components, I have focussed my assessment on high reliability inspection in general (which includes code based inspections) but excludes design for inspectability which is covered in Section 4.2.4.5, below.
2. The RP’s approach involves an Expert Elicitation Panel (EEP) to determine credible defects and a DTA to determine a defect size. An important factor here is the importance of the balance of activities in calculating a limiting defect size from a DTA, which is recognised in (ref. [35]). Another key aspect to the approach is that the inspections will be objective-based (such as following ENIQ methodology) and these will supplement code based inspections.
3. In order to demonstrate how it will implement objective-based inspections (and non-destructive examination/testing generally), the RP has presented its overarching Non-Destructive Examination (NDE) framework (ref. [56]), along with key supporting references. The RP claims that this general framework document provides the basis for requirements and guidance and links back to the structural integrity case claims, covering both code based inspections and objective-based inspections for highest reliability components. The framework is made up from seven annexes, with annexes 1-4, 6 and 7 provided within Step 2 of GDA for assessment. I considered these submissions to be sufficient to inform my Step 2 assessment, which included:

* Annex 1: UT in lieu of RT (ref. [43]). An important document, detailing the technical and ALARP basis for choosing UT over RT.
* Annex 2: VHR/HR Forgings (ref. [57]). Documents the general requirements and applicable techniques for the large forgings.
* Annex 3: VHR/HR Weld Objective-based inspections (ref. [58]). The document that essentially forms the evidence behind this claim (2.1).
* Annex 4: UT Defect detection mechanisms (ref. [59]) A standard document detailing the physical principles behind the UT technique and defects it can detect.
* Annex 5: Inspection assurance (Not yet submitted to ONR).
* Annex 6: Flaw Expert Elicitation Guidance (ref. [60]) A document describing the process of setting up and holding EEPs
* Annex 7: ENIQ-based Inspection Qualification (ref. [61]) This document describes how the RP intends to implement Inspection Qualification for highest reliability components.

1. Within Step 2, it is my expectation that the RP should be able to present the approaches and methodologies for inspection, rather than detailed demonstration of capability.
2. The RP’s approach of using UT in lieu of RT (ref. [43]) is proposed for ASME III components. The RP provided three key reasons, which included compliance with ASME III code case (N659-3 – ref. [43]), reduction of risk associated with radiation exposure to employees and the ability to use phased arrays for improved defect detection and sizing. For stainless steel components and other components where ultrasonics capability may be lower, it is my expectation that the use of RT will be considered. I was satisfied that the RP’s approach and substantiation is reasonable, however further review of the technical substantiation on a component-specific basis, particularly for those within the highest reliability bracket, should be undertaken in Step 3. I consider this to be a residual matter.
3. Within Step 2, the RP highlighted it is considering the use of digital radiography or other alternatives to film, which are now in common use in the UK and internationally. The RP is also considering the use of eddy current arrays to replace surface inspections (conventionally done using dye penetrant or magnetic particle inspection), which is currently in the development phase. It is my expectation that any such application proposed under the NDE framework within GDA will be subject to future regulatory scrutiny, to ensure that risk is reduced ALARP.
4. With respect to inspection qualification for highest reliability applications, the RP’s approach claims to align with the ENIQ methodology. From the high level information I have sampled in Step 2, I consider this is reflected clearly in the E3S case. I consider ENIQ approaches to be well founded, and capable of meeting ONR’s expectations for reliable detection of defects, as is described in ONR guidance under Appendix A4 of (ref. [24]).
5. The main elements of the ENIQ methodology are to develop an inspection specification to define defect types and performance requirements, develop inspection techniques to meet the requirements of that specification, and then qualification of the inspection procedures and personnel through a combination of technical justifications and practical trials.
6. Throughout Step 2, the RP has identified that it is considering the use of partial qualification in relation to transverse defects, the potential use of the Performance Demonstration Initiative (PDI), combination of inspection for manufacturing and PSI, use of Artificial Intelligence for data interpretation and risk informed ISI. Whilst these approaches are not currently presented within the Chapter 23 SCs, the RP has expressed an interest in exploring the viability further. In my opinion, any such proposals would need further scrutiny to understand the intent for application as part of the E3S case, including demonstration of the level of maturity, precedent and the strength of evidence needed to ensure safety demonstration is achieved for the component specific claims.
7. From the information I have sampled, I am broadly satisfied that the level of detail provided in the E3S case and associated NDE framework documents submitted is reasonable for Step 2 of GDA. In my opinion, the approach taken by the RP for the inspection of highest reliability components (of which objective-based inspections are a major aspect) aligns with ONR expectations.
8. During my assessment, I noted that the RP is proposing a graded approach to inspection assurance for the differing levels of “beyond code” classification components, with VHR/HR welds an example (see Section 4.2.3.5 below). The RP has not yet finalised this approach, for which the detailed evidence will be contained in Annex 5 of the NDE Framework (see paragraph 131 above). At the high level presented elsewhere in the E3S case, it is plausible that a proportionate and graded approach to inspection can be developed in line with differing levels of predicted consequences, an approach which has been adopted by other GDA RPs/licensees. The RP is proposing an objective-based approach “in general” for VHR/HR components. In my opinion, the RP’s proposed capability assessment approach may provide confidence in an application, particularly if it follows a partial ENIQ approach and is demonstrably objective-based in nature. However, it should be noted that such a claim of reliability will not be similar to that of the VHR welds subject to inspection qualification, which provides the highest level of reliability. This will require further evidence to be provided on how the RP’s approach ensures risk reduction is proportionate to the level of reliability placed on NDE. I consider this to be a residual matter, which should be taken forward for further consideration in Step 3, as is discussed further in Section 4.2.3.5, below.

#### Subclaim 23.2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound

1. ONR guidance highlights the expectation that for a safety case where a highest reliability claim is made, the design code assessment is supported by a DTA that uses fracture toughness data for the specific metals and welds of concern.
2. It is ONR’s expectation that lower bound fracture toughness data are used for the assessment, based on proven material properties. These properties should be underpinned by direct fracture toughness testing of representative materials in manufacture and through life via a material surveillance strategy for SSCs that may be affected by the environment via mechanisms such as irradiation embrittlement. The values obtained from representative testing should therefore demonstrate a conservative margin between the values used in the DTA, which is bounding.
3. The RP’s approach to underpinning fracture toughness values used in the DTA is consolidated under SC 23.2.2, with further requirements presented within the SIR (ref. [35]). The RP states that upper shelf fracture toughness testing will be carried out on forgings and representative welds. Testing will also be carried out to ensure the transition region for materials is appropriately characterised, with the ASME III KIC fracture toughness curve providing a suitable lower bound value for material behaviour. Additional requirements are placed upon the upper shelf fracture toughness testing to include both ferritic and austenitic forgings and welds.
4. The upper shelf fracture toughness testing method is not currently presented within the E3S case, with the RP stating that an acceptance criterion is the responsibility of the component designer to identify. Whilst the test method for each component is yet to be derived, a number of requirements for such a test programme are listed, including the use of applicable test standards (such as the American Society for Testing and Materials - ASTM), representativeness of test specimens, orientation of specimens, test temperature and the consideration of OPEX. The RP also makes provisions in the SIR (ref. [35]) for the requirement of representative testing, such as “HR/VHR weld upper shelf fracture toughness testing shall use test coupons with a thickness which simulates the geometry and degree of restraint in the production weld”. This is on the basis that “the use of a full thickness specimen provides a high level of confidence that the production welds will be consistent with that demonstrated in the welding procedure qualification”.
5. At a high level, I consider the level of detail provided in Step 2 on fracture toughness testing to be sufficient. I am satisfied that the RP’s approach is reasonable and in line with ONR expectations (Refs. [11] and [24]), citing the RP’s proposal to undertake bounding and representative testing of HR and VHR forgings and welds. I note the RP’s requirement that “the acceptance criterion for upper shelf fracture toughness is not prescribed, and it is the responsibility of the designer to identify an appropriate value considering the context in the safety case.” In my opinion, further evidence is needed on a component/sub-component specific basis, to understand how this approach is derived in context. I consider this to be a residual matter.

#### Subclaim 23.2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern

1. This subclaim describes the RP’s approach to fracture analysis through the production of a DTA. Within GDA, it is ONR’s expectation that DTAs for a sample of the limiting locations in highest reliability SSCs are provided. These should provide evidence of a conservative demonstration using lower bound material properties, with plans for fully representative fracture toughness testing during manufacture to underpin the values used in DTA (as per Section 4.2.3.2 above). The focus of my Step 2 assessment is to identify any fundamental shortfalls in the guidance, methodologies and approaches proposed by the RP against ONR expectations.
2. Subclaim 23.2.3 identifies that the RP’s DTA approach follows the R6 defect assessment procedure (ref. [8]). This procedure is a well-established and validated procedure for assessing the integrity of structures containing defects, or postulated defects, and is routinely used by nuclear site licensees in Great Britain to support structural integrity aspects of nuclear safety cases. The R6 procedure has been used previously by several RPs to support structural integrity claims within GDA. I am therefore satisfied with the RP’s choice of the R6 procedure for DTA.
3. Within Step 2, the RP has submitted two E3S case Tier 2 submissions to further detail the approach described in Subclaim 23.2.3, namely the ‘Defect Tolerance Assessment Guidance’ (DTAG) (ref. [40]) and an accompanying ‘DTAG Technical Basis’ document (ref. [41]). These submissions provide the detailed approach for the application of the R6 method and the selection of key inputs including; material property determination, load combinations, fatigue crack growth, stress analysis considerations, flaw characterisation, failure assessment diagrams and the determination of limiting and safety significant defects. I have conducted a high level review of these documents and have not identified any fundamental concerns with the approach at this stage of GDA.
4. The RP’s acceptance criterion requires a target RF between the end of life defect size, calculated from the Inspection Target Defect Size (determined under Subclaim 23.2.1) with Fatigue Crack Growth (FCG) through life (accounting for environmental effects), and the Limiting Defect Size (LDS) corresponding to the position on the R6 failure assessment curve, calculated using lower bound material properties (determined under SC 2.2).
5. The target RF for proposed HR and VHR components is 2 (see Section 4.2.3.5), which is consistent with the approaches established in previous GDAs and aligns with ONR expectations in NS-TAST-GD-016 (Sec.5.92 of ref. [24]).
6. In my opinion, the RP has provided a comprehensive package of information on its DTA approach at this early stage of GDA, with consideration of detailed technical inputs included. From the information I have sampled, I am broadly satisfied that the RP’s submissions in Step 2 demonstrate that the approach to DTA is in line with ONR expectations and provides confidence that the RP is capable of producing a robust and conservative DTA.
7. Whilst at a fundamental level I am satisfied with the information provided on the RP’s approach to DTA, component specific examples are yet to be finalised by the RP, while the Rolls-Royce SMR design is being developed. I do not consider this to be a fundamental shortfall at this stage of GDA, as sampling of component specific DTA calculations is an evidential level of assessment, which I do not consider to be necessary during Step 2.
8. During engagements within Step 2, the RP has indicated that DTAs will be provided within the scope of GDA for all highest reliability components (VHR/HR). From my assessment of the E3S case structure and structural integrity demonstration under Chapter 23, I am aware that the CSRs will summarise the RFs for each component, enabling the user to identify where RFs are most challenging. On the basis that the DTA approach and application to components is still being developed and refined, in my opinion further assessment of the technical details of the RP’s approach, as well as how it is implemented for several components where the margins are most bounding, should be undertaken during Step 3 when the RP’s E3S case submissions are updated.
9. This is a residual matter to be progressed when further information is available on the DTA calculations and evidence presented for components, sub components and welds, where the consequence of failure is highest or the demonstration of safety is most challenging. This will provide assurance that the RP’s approach, as detailed in the guidance documents and claimed in the E3S case, is appropriately undertaken in practice, based on conservative calculations and input parameters.

#### Reconciliation of the avoidance of fracture demonstration

1. One important aspect related to a robust AoFD is the integration and reconciliation of inputs, which have been clearly defined by the RP as being related to qualified inspection, conservative material properties and a robust DTA (SC 23.2.1, 23.2.2 and 23.2.3 respectively). Determination of the component-specific RFs should not be overly reliant on any one input of the AoFD, ensuring the RF is credible, reasonable and achievable.
2. Within SC 23.2, the RP has reflected this expectation, arguing that a balance is sought between activities undertaken to derive the key inputs of an AoFD, so that there is not too great an onus placed on any one aspect in the safety case. I am satisfied that this high level expectation is contained within the SC but noted that, unlike other instances where the RP has provided comprehensive information and guidance in E3S case supporting documents, there was no similar guidance or requirements regarding how the RFs are reconciled to ensure a balanced and achievable AoFD.
3. I sought further information as to how this is achieved. The RP noted that the process of determining RFs is iterative and necessary to inform and refine the design of Rolls-Royce SMR components and that the decisions were captured in the component design under the RP’s requirements management process. The RP acknowledged that additional guidance on the reconciliation of the AoFD process and determining RFs could be captured within the E3S case.
4. I consider the detail and scope of the reconciliation process guidance to be important to the E3S case, to demonstrate how a balanced AoFD for the design of VHR/HR components is substantiated. Evidence of how multiple technical inputs to underpin the beyond code provisions for VHR/HR components have been considered is an ONR expectation under SAP EMC. 3 (ref. [11]). Input considerations to support such claims may include the material testing strategy to ensure robust demonstration of material properties, refinement of material specification to improve mechanical properties, component optimisation to improve inspectability (surface finish, grain size consideration) and conservative structural analysis. I consider the substantiation of AoFD reconciliation within the E3S case to be a residual matter that I will follow up in Step 3.

#### Variation between HR and VHR for the demonstration of reliability.

1. Section 4.2.2.1 discusses how the RP has differentiated between VHR and HR classification, whereby it seeks to demonstrate that HR components have less severe consequences of failure compared with VHR, due to safety measures provided by In-vessel Retention (IVR) and the containment vessel to prevent significant off-site release.
2. The RP claims that the tolerable failure frequency of HR components is greater than for VHR components, therefore warranting a reduced burden of evidence to support the safety case structural integrity claims. It should be noted that measures for HR classification components still go beyond those required for Safety Class 1 under ASME III requirements. These are discussed in turn for each of the parameters affecting AoFD.
3. The RP states in (ref. [8]) that “both HR and VHR classifications are considered to be higher reliability classifications which warrant beyond code activities. No distinction is made between the requirements for these classifications, with the exception of inspection assurance (subclaim 23.2.1, section 23.2.1)”. I sampled the substantiation of these requirements further, based on the SIR and other supporting documents.
4. For Subclaim 23.2.1 (manufacturing examination), the RP claims that for HR welds, the additional end of manufacture examination shall be subject to an Inspection Capability Assessment, as opposed to Inspection Qualification, which is specific for VHR welds. At a high level, the RP’s proposed use of a graded inspection approach is reasonable, however any such strategy should be considered and substantiated in the context of reducing risks ALARP and the impact on the overall AoFD for the component in question.
5. It is challenging to quantify a reduction in safety justification based on any one element of an AoFD, when considering above code aspects whereby conservatism is necessary to manage uncertainty. It is therefore my opinion that this aspect of the RP’s approach and E3S case requires further assessment, to fully understand the benefits and disbenefits that have been considered by the RP in terms of time, cost and effort compared to the reduction in risk. I expect further evidence to be presented within the E3S case of where HR classification is being claimed, to demonstrate how the inspection approach varies compared to VHR examples and what overall effect this has on the AoFD to support the conceptual defence in depth argument. I consider this to be a residual mater.
6. For SC 23.2.2 (fracture toughness), the fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound. To underpin this, a range of additional testing is proposed, including supplementary Charpy testing in the transition region and upper shelf fracture toughness testing for all VHR/HR ferritic forgings and representative welds (including the weld heat affected zone).
7. The RP’s SIR (specifically SIR no. 831, ref. [35]) states that “the acceptance criterion for upper shelf fracture toughness is not prescribed, and it is the responsibility of the designer to identify an appropriate value considering the context in the safety case”. The SIR (Ref. [35], SIR no. 299) states that “it is likely to be proportionate to still test those representative welds as the cost and effort is relatively low”. This is confirmed under SIR no. 270 (ref. [35]) that “for HR/VHR welds, representative welds (for example, welding procedure qualification specimens) shall be subject to upper shelf fracture toughness testing of both weld and heat affected zone locations”. In my opinion, the RP’s consistent approach for fracture toughness testing of HR and VHR welds is reasonable. The identification of appropriate fracture toughness values for HR and VHR classified components, sub components and welds specific to the context of the E3S case (as per SIR no. 831) will require further assessment, when the necessary evidence is available on a component specific basis in Step 3.
8. Subclaim 23.2.3 (Defect Tolerance Assessment) initially stated that the target RF for HR components was 1.4, however this approach has been revised by the RP during Step 2 to match that for VHR, meaning the target RFs for both VHR and HR components are now 2. The RP has confirmed that future updates of Chapter 23 (ref. [8]) and associated documents (SIR, DTAG, DTAG TB, CSRs etc) will be amended to reflect this position.
9. This change is expected to have an impact on the design of the components and subcomponents of this classification, as well as the level and scope of evidence expected to underpin such claims. I have already identified the designation, scrutiny and demonstration of HR classification to be a residual matter (Section 4.2.2.1 above). Therefore, I expect the updated E3S Tier 1 and 2 documents to ensure the target RF of 1.4 has been amended and that the requirement for a target RF of 2 is reflected in subsequent evidence submitted for HR classified components and subcomponents.
10. Overall, based on the information provided so far, it is my opinion that the E3S case does not contain sufficient evidence to determine whether the reduced structural integrity controls proposed for HR claims are reasonable and proportionate to the claimed lower failure consequences. I do not consider that this constitutes a shortfall for the purposes of the GDA structural integrity Step 2 assessment, on the basis that a graded approach to demonstrating multiple highest reliability safety classifications has been accepted in previous GDAs, when the provision of more underpinning evidence has been sampled.
11. I note that further component-specific evidence will be provided in the CSRs, for instances where the HR claim is being applied to components, subcomponents and welds. I expect this to demonstrate how the RP (through the component design process) has identified, specified and justified inspection requirements, commensurate with safety significance.
12. I consider the overall demonstration of differences between the proposed AoFD for HR and VHR components to be captured within the extant residual matters identified above within section 4.2.3 of my assessment report.

### Structural integrity design provisions for the Rolls-Royce SMR design

1. ONR guidance states that to demonstrate structures meet their safety functional requirements, it is necessary to establish that sound design concepts, rules, standards, methodologies and proven design features have been used, and that the design is robust (ref. [24]).
2. Whilst the fundamental structural integrity aspects of the E3S case are considered in the previous sections of my report, this section focuses on how sound design concepts and design features have informed or been informed by the RP’s approach to structural integrity demonstration.

#### Structural integrity aspects of component design related to the seismic isolation system

1. The RP is applying the concept of seismic isolation for the Rolls-Royce SMR design, through the use of a seismic isolation system to provide attenuation of horizontal accelerations during an earthquake (described further in E3S Case Chapter 9B: Civil Engineering Works & Structure, ref. [5]). In general, this should reduce the impact of earthquakes and the associated dynamic loading on components. The seismic isolation system will support the entire containment vessel structure and the majority of SSCs important for safety, including the reactor coolant system and emergency safety systems.
2. The containment structure lower dome is supported by a thick concrete base mat, which in turn is supported by the seismic isolation system, which is mounted on the Reactor Island foundation slab. The seismic isolation system offers a base isolation system for the hazard shield (and SSCs housed within it) against horizontal seismic loading.
3. ONR guidance in SAP ERL.1 (ref. [11]) states that “the reliability claimed for any structure, system or component should take into account its novelty, experience relevant to its proposed environment, and uncertainties in operating and fault conditions, physical data and design methods”. ONR guidance provide under SAP EMC. 7 states that the “schedule of design loadings (including combinations of loadings) for components and structures, together with conservative estimates of their frequency of occurrence should be used as the basis for design against normal operation, fault and accident conditions”. This should include plant transients and tests together with internal and external hazards. This is a further supported by the EHA series of SAPs, that external and internal hazard loads should also be considered as part of the design (ref. [11]).
4. I sampled the E3S case to understand how the function and performance of the seismic isolation system has influenced the structural integrity controls for component design. I considered how the seismic isolation system has influenced the derivation of design loadings and hazard withstand of safety classified components and how the structural integrity aspects of pressure retaining SSCs that traverse the seismic isolation boundary address the challenges of same-system differential movement and dynamic loading.
5. My sample identified that the SIR (ref. [35] SIR no. 147) requires safety class 2/1/HR/VHR mechanical components and their supports (including the containment vessel) to comply with ASME III. This code requires component design loadings to be established, which define all design basis loadings from both internal and external sources, including earthquake loads. In short, components must be designed and substantiated to account for design basis seismic loading events.
6. Subclaim 23.1.4 refers that “conservative analyses are carried out in accordance with the design code requirements to demonstrate tolerance to all design basis loadings”. Loading combinations that need to be considered in the design of components are discussed in the E3S under ASME III Design by Stress Analysis Guide (DBSAG) (referenced from SIR no. 882 as ref. [42]) and Defect Tolerance Assessment Guide (referenced from SIR no. 884 as ref. [40]). The DBSAG states that the overall Rolls-Royce SMR design seismic design, including seismic isolation provided by the seismic isolation system, and assessment methodology will be managed in the seismic methodology and seismic assessment strategy documents, which are currently being developed.
7. As explained in Section 4.2.1 above, the evidence to underpin the subclaims (in this case, loading values used in the stress analysis and sizing of components for code compliance) is presented within the relevant sections of the CSRs. At the time of my assessment, the E3S case did not contain this level of detailed evidence, however the RP provided the RPV CSR for another line of ONR sampling as an example of CSR content and structure. I noted that within the RPV CSR, the RP indicated that the detailed evidence of design loadings considered will be provided in the ‘RPV Body ASME III Design Report’, which consolidates the structural analysis, seismic assessment, and drawing definitions to substantiate the requirements of the ASME III design specification.
8. The Rolls-Royce SMR design seismic assessments are being undertaken by the RP’s Civil Structural team. The overall seismic aspects of the Rolls-Royce SMR design, including the protection from the seismic isolation system and assessment methodology, will be managed in the seismic methodology and seismic assessment strategy documents. The adequacy of these approaches, design methodologies and strategies for the E3S claims are being considered within the ONR GDA civil engineering (ref. [62]) and external hazards (ref. [63]) assessment and are not within scope of my structural integrity assessment.
9. In my opinion, the information sampled so far within Step 2 confirms that the achievement and demonstration of reliability for SSCs located within the hazard shield are based on bounding seismic loads, derived from performance of the seismic isolation system. I therefore consider that the performance of the seismic isolation system is an important factor in underpinning the design of all SSCs, for which Claim 23 on structural integrity is applicable.
10. For this reason, I consider that any structural integrity judgement related to the design of SSCs within the scope of Chapter 23 that are located on the seismic isolation system are contingent on the outcome of the ONR civil engineering and external hazards GDA reports (refs. [62] [63]). I expect the E3S case to demonstrate how any reduction in seismic loads have been used to establish the design parameters of higher classification pressure boundary components, to understand how safety margins have substantiated. I will assess this further in Step 3 when component-specific evidence is available.
11. For my second sample of the seismic isolation design feature, I sought to understand how the RP has considered and addressed structural integrity demonstration of pressure retaining SSCs that may traverse the boundary between seismic isolated and non-isolated structures. In particular, how the challenges of differential movement and dynamic loading of components during a seismic event is substantiated.
12. Section 6.4 of the RP’s E3S submission ‘Design Description for Reactor Island – Hazard (ref. [47]), confirms that several SSCs important for safety traverse the hazard shield (and thus the seismic isolation) boundary. These are identified as the Main Steam Piping, Feedwater Piping, Auxiliary Feedwater Piping and Essential Services Water System.
13. At the time of my Step 2 assessment, there was no system-specific detailed information provided in the E3S on the consequences of failure of the systems identified above, especially in the locality of the hazard shield building penetrations. There is work being undertaken by the RP regarding layout of these systems, therefore it is understandable that the full consequences of failure (particularly from an indirect perspective) for pipework traversing the seismic isolation boundary is not yet complete. As this information is outstanding, a robust classification to define structural integrity controls cannot be established or provided for sampling.
14. To ensure these measures are proportionate to safety significance and to understand how the structural integrity of such components is accounted for within the design, further evidence should be provided within the E3S case for relevant design aspects and structural integrity substantiation of the main steam line and feedwater line pipes when the necessary information is available. I expect this to include relevant consequence analyses, design considerations and the classification of subcomponents and welds. I consider this to be a residual matter.

#### Structural integrity design provisions for response of the reactor pressure vessel during severe accidents

1. The Rolls-Royce SMR design uses an IVR safety feature for severe accidents. This is briefly described in Chapter 6 ‘Engineered Safety Features’ under Section 6.3 “safety features for stabilisation of the molten core”. The RP has acknowledged that at this stage of the Rolls-Royce SMR design GDA, this engineered safety feature is in the early stages of development and that further details will be presented in future revisions of the E3S Case as the design and associated evidence matures.
2. From the limited information provided, it is apparent that the RPV provides a key safety role for the retention function of the core melt, and as such is likely to have an associated integrity claim for performance during a design extension condition B (DEC-B) scenario. ONR guidance states that a safety case should present a list of all initiating faults which are included within the design basis of the plant. All loadings for operation, credible faults, accident conditions and tests should be identified and the magnitudes specified (ref. [11], EMC.7, EMC.11). Load definitions should be conservative, and remain appropriate for the future operation of the structure.
3. Evidence to support the Chapter 23 SCs (achievement, demonstration and maintenance of reliability) is presented within CSRs. The RP claims that the approach to presenting a robust fracture case for the IVR transients will be provided in the RPV CSR, which will take account of RGP, including the Rolls-Royce SMR design DTAG (ref. [40]).
4. At this early stage of GDA, it is not my expectation for detailed accident condition loadings to be presented in the E3S case, however I do consider it important for the function of IVR to be recognised as an important case for the RPV integrity demonstration.
5. The RP provided the RPV CSR, from which I identified that the demonstration of RPV integrity across all claimed operating conditions is intended, including during a severe accident, which is necessary to underpin IVR performance. The CSR has a dedicated chapter to provide evidence that during a core melt scenario, gross failure of the RPV does not occur on the basis that successful IVR operation provides sufficient cooling to prevent loss of RPV via creep or internal wall ablation.
6. At a high level, I am broadly satisfied that the RP’s approach to structural integrity demonstration considers this onerous severe accident condition in the design of the RPV, which is essential to underpin IVR functionality and in turn, any claimed HR classification (see Section 4.2.2.1). I expect the E3S case to contain evidence to ensure that a reasonable margin of demonstration is achieved, which is proportionately conservative to the claim being made.
7. I expect further evidence to be presented within the E3S case on the tolerability of the RPV to spurious or malicious operation of the IVR during normal plant operation, to understand whether the RPV is tolerant to such transients or whether claims are made and substantiated on the reliability (valve failure, control and instrumentation failure or security measures in place for the system, that likelihood of an such an event occurring is reduced to ALARP). I consider these aspects of the structural integrity provisions for the RPV design to support IVR functionality to be a residual matter.

#### Structural integrity design provisions to support multidiscipline engagement

1. Structural integrity demonstration of SSCs important for nuclear safety are often complex and can require engagement between a range of specialist and topic areas, to ensure the outcome meets the overall safety and performance design requirements, while reducing risk to ALARP. Within Step 2, I considered it important to understand how inter-discipline interactions have informed the structural integrity aspects of the E3S case, reflected through the SSC design decisions of the Rolls-Royce SMR design.
2. Subclaim 23.1.3 states that components are designed for structural integrity, including a review of OPEX and RGP. To achieve this, the RP has adopted an approach called ‘Design for X (DfX)’, which is embedded in working practices through the use of IPTs, made up from a range of functions such as design, structural analysis, procurement, manufacturing, examination and verification. This is defined as a specific requirement in the SIR (ref. [35] SIR no. 854) to ensure cross-functional engagement throughout the design process.
3. The RP provides several examples, showing how structural integrity design principles derived from RGP and OPEX have been used to inform the Rolls-Royce SMR design. These include access for inspectability to aid UT during manufacture, consideration of material forming to minimise the number and length of welds where reasonably practicable (one example being the RPV upper shell will use integrated nozzles) and the consideration of through-life ageing and degradation mechanisms to inform component design features (minimisation of crevices, materials selection).
4. The subclaim states that DfX follows a “structured design optioneering process”, referenced as C3.2.2-2 ‘Design Optioneering’. Within GDA, this process is defined in (ref. [39]), which is populated by information exported from the RP’s Engineering Portal, which is part of its Integrated Management System.
5. At a high level within Step 2, I am satisfied that the RP’s design approach referenced in the E3S case has identified the need for key interactions with other technical disciplines. However, I consider further assessment is necessary in Step 3 to understand how this process is managed, such that relevant stakeholders (in this instance, those with technical knowledge of the SIR rationale) are identified to inform the component design decision making process.
6. To understand how this design approach identifies and manages key interactions with other technical disciplines, I sampled the RP’s submission summarising C3.2.2.-2, focusing on how the IPTs are formed, such that all necessary stakeholders related to SIR are involved in the decision making process. From my review, it was apparent that the C3.2.2-2 approach is a generic format that is used for all key design decisions, not solely for structural integrity aspects. As such, the guidance provided in (ref [39]) only defines the steps undertaken as part of the process. Step 1 of the C3.2.2-2 approach contains the rule that “all relevant stakeholders shall be identified”. The process states that the design engineer is responsible for this step and that the lead engineer/design manager is accountable.
7. I raised RQ-01198 (ref. [48]) for the RP to explain how the members in the IPT are selected. The response explained that IPT members are invited from relevant areas and that the specific make-up will vary dependent on the component - for example specific attendees from the systems design or layout teams may be required. IPTs are broad ranging so anyone can be invited where they require awareness or input. The RP claims that its Gated Review (GR) process will ensure appropriate stakeholder engagement. No further details were provided on how the GR process ensures appropriate stakeholder engagement.
8. At the high level presented in Chapter 23 and the SIR, the principles of DfX seem reasonable for mandating interaction with multiple technical disciplines, however from my review of the information provided, it is not immediately clear how relevant stakeholders are identified, especially in the context of ensuring that structural integrity requirements are covered. I do not consider this to be a significant shortfall at present, based on the maturity of the case and the example of design decisions made so far. I expect suitable evidence to be available to demonstrate how structural integrity expertise has been used to inform the design of components to support E3S case claims, and how decisions that may impact other claims in the E3S case have been communicated to other disciplines. This is a residual matter.

#### Structural integrity design provisions for modular construction

1. The Rolls-Royce SMR design is proposing to use a modular construction approach, whereby the modules conform to a standardised grid, allowing for the creation of a tessellating system that allows flexibility of layout to suit system requirements as well as allowing alignment of interfaces to civil elements. The RP’s E3S case submission ‘SMR Mechanical, Electrical, Pipework Module Kit of Parts System Concept Definition (ref. [64]) provides an overview of the generic proposed module design, including the differentiation between ‘Primary Structures’ (consisting of all module frame types and variants) and ‘Secondary Structures’ (ancillary items of standardised commodities, used for safety and hazard protection elements).
2. ONR expectations for plant layout of a nuclear facility are described under the ELO series of the ONR SAPs (ref. [11]). In general, it is ONR’s expectation that the design and layout should enable access for necessary activities and minimise adverse interactions while not compromising security aspects (ELO.1). The design and layout of the site should also be such that the effects of faults and accidents are minimised.
3. I raised RQ-01082 (ref. [48]) to clarify where the RP considered the jurisdiction between civil engineering and structural integrity (and supporting design codes) topics lie, given the close relationship between the safety demonstration for both topics. The RP’s response clearly defined the difference between SSC supports (according to ASME III description) and the steel framed modules, the latter being part of the civil structure. I was satisfied with the response, which has resolved ambiguity between the assessment scope of the structural integrity and civil engineering assessments and consider that the structural claims on modular unit framework are out of scope of the structural integrity assessment.

#### Structural integrity design provisions for plant layout and access for inspectability

1. ONR SAP EMC.8 describes ONR expectations specific for structural integrity and provision for examination, such that “geometry and access arrangements should have regard to the need for examination”. I considered it necessary to understand how ONR expectations for access for inspectability have been considered, given the compact nature of the Rolls-Royce SMR design.
2. Section 4.2.4.3 above discusses the RP’s design approach of DfX. One example given by the RP is “Examination Access”, highlighting that the Rolls-Royce SMR design E3S principles (ref. [38]) require designs to enable access for inspections. This is captured in the SIR (ref. [35] SIR no. 670) and E3S Principle 2.19.3 (ref. [38]): "the design of metallic components should facilitate necessary examinations to identify the existence of defects where reasonably practicable, both in manufacture and throughout the lifetime of the component. This includes methods of redundant and diverse examination where appropriate”. The E3S design principles (ref. [38]) also contain a section on Examination, Maintenance, Inspection and Testing, under which Principle 2.31.3 states “SSCs should be designed to facilitate Examination, Maintenance, Inspection and Testing, as far as reasonably practicable”.
3. An example was provided by the RP under the SIR whereby design requirements for pipework are in place which ensure a sufficient straight length for access for UT. The RP is therefore claiming that components have been designed for UT, on the expectation that UT will be required for in-service inspection (ISI) as well as manufacturing examination for highest reliability components, and for any potential deployment of UT in lieu of RT.
4. At a high level, I consider there is clear indication that the RP’s safety justification has provision for the consideration of access for inspectability and geometric optimisation in the Rolls-Royce SMR design components. Whilst this is indicative of meeting ONR expectations, the practicalities of implementing these requirements and principles has yet to be demonstrated at a component level. Until plant layout is finalised and component specific design details are available for assessment, it is not possible to confirm that the plant design has been optimised to provide the necessary access for inspection. There are also other aspects of component design optimisation for inspection that should be addressed, such as materials selection, surface finish and control of local environmental conditions, to name a few.
5. I am satisfied that for Step 2, the RP has demonstrated an appreciation of access for inspectability within its fundamental structural integrity requirements, which informs design approaches. In my opinion, this approach generally aligns with Requirement 32 of the IAEA SSR2/1 (Ref. [28]) to design for optimal operator performance and SSG-56 that the layout and location of piping and equipment should provide sufficient accessibility to allow periodic testing, maintenance and inspection to be conducted, including maintenance and inspection of welds and piping supports (Ref. [30]).
6. I expect further evidence to be presented within the E3S case to demonstrate how all aspects of the DfX process have been reconciled (including access for inspectability) to ensure that the principles embedded in the design at a component-specific level are proportionate to safety significance and reduce risk ALARP. I consider this to be a residual matter.

#### Structural integrity design provisions for the primary circuit chemistry regime

1. During initial familiarisation with the Rolls-Royce SMR design within Step 1, I identified the RP’s adoption of a potassium hydroxide (KOH) primary circuit chemistry, which varies from most typical PWR designs that use boron-dosed lithium hydroxide (LiOH), as an area of interest for sampling from a structural integrity perspective. The RP claims that the decision to operate without boron demonstrates risk is reduced ALARP for a number of reasons, citing elimination of a waste stream, simplification of chemistry equipment, a passive safe shutdown condition and the potential reduction in material degradation as being key hazard reducing factors considered. Conversely, the RP also recognises that the adoption of a boron-free, KOH chemistry regime presents a number of challenges to the design and incorporation of the Control Rod Drive Mechanism (CRDM) penetrations in the RPV closure head (CH).
2. I targeted this as an area for sampling within Step 2, considering two key aspects from a structural integrity perspective; firstly, the influence on materials selection and compatibility from operating a KOH coolant; and secondly, the design, manufacturing and inspection challenges associated with increasing the number of CRDM penetrations of the high safety significant RPV CH.
3. For the first aspect considered, the topic of the KOH primary circuit material compatibility has been undertaken collaboratively with the ONR Chemistry specialist, which is covered in more detail within Section 4.2.5 (ageing and degradation) of my report. I have therefore focused this part of my assessment on the structural integrity aspects considered in the design of the RPV CH, specifically related to the number and significance of penetrations required to deliver the FSF for control of reactivity (CoR) and confinement of radioactive material (CoRM).
4. The safety demonstration for the RPV CH is presented in the E3S case under Chapter 5 – ‘Reactor Coolant System & Associated Systems’. The RPV CH is classified as VHR, warranting the highest level of structural integrity demonstration (see Section 4.2.2.1).
5. According to SC 23.1.3, “components are designed for structural integrity, including a review of OPEX and RGP”. Underpinning this, the SIR (SIR no. 855, ref. [35]) identifies E3S Principle 2.19.6 (ref. [38]): “the design of metallic components should minimise the number and length of welds needed where reasonably practicable and should position welds away from high-stress locations and adverse environments where reasonably practicable”.
6. Similarly, SIR no. 854 also underpins SC 23.1.3, setting the requirement that “the use of IPTs ensures cross-functional engagement throughout the design process. Other transverse requirements from different functional areas are also used to set requirements on the component designs”. This requirement, derived from the RP’s SI design principles, is broadly aligned with ONR expectations presented in ONR guidance under SAPs EMC. 9 to minimise number and length of welds in the design.
7. I sought to understand how these specific SIR had been considered in the design of the RPV CH and how structural integrity aspects have informed the overall design decision to operate a boron free chemistry. Two aspects of assessment are considered here – firstly, how the RP’s SIR have been considered as part of a balanced ALARP judgement to operate with a non-boron KOH primary circuit, and secondly how the challenges associated with this decision have been addressed for the three claims in Chapter 23, such that the claimed RPV CH reliability is achieved, demonstrated and maintained.
8. For the Rolls-Royce SMR design, the RP has undertaken a review comparing PWR options that use boron for duty reactivity control and a boron-free PWR that does not rely on boron for duty reactivity control. The boron-free option is identified as the design that presents risks that are ALARP. I sought to understand how the RP’s SIR had been considered as part of the RP’s decision to operate the Rolls-Royce SMR design in a boron free KOH primary circuit coolant chemistry regime.
9. I sampled the RP’s decision record on this design feature (RI1 – PCD2 Boron Free Decision, ref. [65]) and noted several entries where the RPV CH design challenges are recorded as disbenefits of a boron-free design, including “reactor circuit complexity associated with additional RPV head penetrations that will affect the internal stresses and overall strength of the RPV head in comparison to the boronated option”. This was judged as acceptable by the RP, on the basis that the component could still be designed to meet the requirements of an ASME III Class 1 component.
10. In my opinion, the consideration of structural integrity at this stage of development for the RPV CH is relatively basic, limited to an understanding of meeting code compliance with no reference to the SIR identified elsewhere in the E3S case as design principles. I also note that the RP refers to the RPV CH as being a safety class 1 component, where as information presented in the component-specific and structural integrity specific Chapters (Chapters 5 and 23, respectively) refer to the RPV CH as VHR, based on preliminary consequence analyses.
11. I am cognisant that the RP is currently developing the reactor design, and indeed the accompanying E3S case, such that safety classification is an iterative process. However the importance of structural integrity requirements at the design stage vary significantly between VHR and Class 1 components. In the instance of the RPV CH design, the requirements laid down in SC 23.3.1 (and possibly SC 23.2 pending final component classification) do not appear to have been considered as an influencing factor in the decision to adopt a boron free KOH operating chemistry. From my sample, it is unclear exactly how a VHR claim may affect the decisions made, given the increased burden associated with a VHR classification on key design decisions such as minimising the number and length of welds, access for inspectability and the potential burden of elevated requirements for through-life inspection and integrity demonstration.
12. Notwithstanding my finding, I note that the conclusion of the decision record states that the boron-free KOH option is identified as the design that reduces risk ALARP, conditional upon development of the reactor assembly design and analysis of its performance to confirm safe operation. At this stage of my assessment and E3S case maturity, I accept that refinements to component classification and design features are iterative as key analyses are being undertaken. However I consider it necessary to conduct further samples of similar SSC design decisions, for which structural integrity considerations have been applied, to inform my judgment on the adequacy of the RP;s approach to selecting IPTs. This is included within the existing residual matter I have raised on IPTs and the RP’s decision making process.
13. From my assessment, it is apparent that the RPV CH design is optimised on the basis of a non-boron containing primary circuit chemistry, with the number of control penetrations already reduced from initial expectations, in response to the SIR (ref. [35]). The evidence to demonstrate how the RPV CH design meets Claim 23 requirements is not currently available for assessment. I expect evidence to be available within the E3S case, to demonstrate how the iterative design process is managed when circumstances considered in previous decisions are changed (in this case component classification from Class 1 to VHR) and to confirm that the RPV CH design reduces risk ALARP, meeting the requirements of a VHR component. I consider this to be a residual matter.

#### Structural integrity design provisions for consequences of failure analysis

1. A safety case for metal SSCs should be examined in the context of the overall safety justification for the plant, taking account of interactions with other safety features. ONR guidance expects that good design practice is followed to ensure defence in depth in the plant design (SAP EKP.3) and that the design and layout should be such that the effects of faults and accidents are minimised (SAP ELO. 4). This is of particular importance for a compact plant, where physical space between SSCs may be restricted, influencing the RP’s approach for use of engineering protection measures or discounting indirect consequences. I therefore considered it important to sample how the RP has considered consequences of gross failure (direct and indirect) in component safety classification to determine the structural integrity controls applied.
2. Subclaim 23.1.1 states that “components are classified based on gross consequences of failure, assuming they no longer deliver the associated safety function”. It also states that the “process adopts a FMEA approach, to understand the complete consequences of gross failure of sub-components and regions, including accounting for secondary consequences”. The term of “secondary consequences” is further explained in the SIR under SIR no.757, as being the “generation of further hazards following a failure”. In my opinion, this description is akin to the ONR description of indirect consequences and the generation of internal hazards.
3. I consider the explicit reference of secondary consequences in the E3S case to be an important factor in determining component classification and one that aligns broadly with ONR expectations. The scope and adequacy of consequence analysis undertaken for component failure to inform classification is a topic previously considered (Section 4.2.2.1), which will require further assessment when the RP’s E3S case is further developed and component specific examples can be sampled. I have already raised this as a residual matter.

#### Structural integrity provisions for design of containment

1. ONR assessment of structural integrity claims is concerned with the integrity of metallic components and structures, which include the freestanding metal shell of pressure retaining containment structures and associated supports. Such a component is often the largest safety significant pressure vessel on site, necessary for demonstrating defence in depth. For this reason, I considered it a key item to sample within Step 2.
2. Chapter 6 of the E3S case identifies the Rolls-Royce SMR design ‘Engineered Safety Features’, which includes safety demonstration of the Rolls-Royce SMR design Containment Vessel Structure (CVS). The CVS is a leak tight, free-standing steel pressure vessel, just over 40 metres high and just under 35 metres (approximate) in diameter that houses the primary circuit (RPV, SGs, RCPs, pressuriser, coolant lines), associated equipment and safety systems. It consists of a cylindrical shell and two semi-elliptical domes. The shell and domes are manufactured from SA738 Grade B curved plates approximately 60mm thick.
3. The scope of Claim 23 covers “pressure boundary components and their supports”. As per ONR SAPs, I would consider the CVS to be a pressure boundary component and therefore subject to a demonstration of integrity commensurate with the claims in Chapter 23. I raised RQ-01084 (ref. [48]) to clarify my understanding, which the RP confirmed by response.
4. E3S case submission ‘SMR0005089 - Containment System Design Description’ (ref. [46]) provides an overview of the components that form the metallic containment pressure boundary, highlighting that the demonstration of safety for different SSCs within the Containment System will be the responsibility of several engineering disciplines, including structural integrity, civil engineering and mechanical engineering. The RP also confirmed that the Containment System (not just the CVS) is covered under E3S Chapter 6, but the CVS support structure is considered under E3S Chapter 9B. This is because the RP considers it “better aligned” to the civil engineering topic area due to the materials and construction method (response to RQ-01084, ref. [48]). I reviewed both Chapter 6 and Chapter 9B to understand the scope of SSCs that make up the CVS and the associated support structure that are relevant for my structural integrity assessment.
5. Structural integrity assessment of an SSC which forms part of a containment system considers guidance provided in the SAPs on containment and ventilation under ECV.1 to ECV.10 (ref. [24]). This guidance identifies a range of considerations (Para. 525, ref. [11]), of which I considered the following to be most relevant to structural integrity:

* Definition of the requirements for the performance of the containment during severe accidents, including its structural integrity and stability;
* Demonstration of integrity for the metal components, welds and supports including codes and standards.
* Withstand to internal and external hazards

1. I sampled the RP’s E3S case against each of these expectations. I was able to identify the definition of safety functional requirements for the CVS, which within Chapter 6, specifies it as being “CoRM during normal and fault conditions”. The FSF is to confine radioactive material, which is a category A function, making the CVS a safety classification 1 component in accordance with the RP’s classification approach (ref. [36]). Integrity of the CVS is important for demonstrating the overall defence in depth of the Rolls-Royce SMR design, however it provides a vital part of the safety demonstration of components with a safety classification of HR or below, whereby the combination of CVS and RPV integrity (with HR being reliant on successful operation of the IVR function during a severe accident) are required to limit off site radiological release.
2. I am satisfied that at a high level, the E3S case identities the definition of functional requirements for the CVS, through which the functional categorisation and component safety classification has been derived.
3. With safety classification established, the demonstration of integrity will be in accordance with ASME III, which is confirmed under E3S Claim 6, highlighting subsection NE, Class MC Components as being used to establish rules for material, design, fabrication, examination, inspection, testing, and preparation of reports for metal containment vessels.
4. At a high level, I consider the RP’s use of ASME III Subsection NE to demonstrate integrity of the CVS as reasonable, based on the broad adequacy of code compliance being an acceptable basis to underpin class 1 component safety demonstration. I do however note GDA precedent on this approach, whereby a number of specific aspects of CVS design were challenged during GDA in the context of reducing risks ALARP (ref. [66]). I consider these may be applicable to this GDA, given the similarities in approach for achievement of CVS integrity claims.
5. One example I noted from my review was the materials selection and specification proposed for the CVS. The RP’s decision has been influenced by the proposed use of an ASME III code case (N-841), whereby an exemption to undertake mandatory Post Weld Heat Treatment (PWHT) can be invoked for this material, up to a certain thickness. This decision was subject to the RP’s decision making process as item RI-76, which according to SIR no.737 (ref. [35]) states that there is further work being undertaken by an external contractor to demonstrate either that high residual stress levels from not undertaking PWHT can be tolerated, or identify the route to providing that demonstration.
6. At this stage in GDA, I do not consider the RP’s proposal to adopt this approach for the Rolls-Royce SMR design CVS to be a fundamental concern. It should be noted that whilst there may be a precedent set by previous GDAs, these have been assessed in the context of ALARP arguments presented for those specific designs. In this instance, the RP is still undertaking analysis work to support its approach, suggesting there are many design-specific aspects that need to be considered before a balanced judgement can be made that such an approach reduces risk ALARP. I expect substantiation of the CVS design within the E3S case, including implementation of ASME III Code Case N-841 and consideration of relevant GDA precedent when the supporting evidence is complete. I consider this to be a residual matter.
7. The third aspect of my structural integrity assessment of the CVS is the need to demonstrate tolerance to all internal and external hazards (including combined hazards), such that if a hazard occurred the design is able to reach a safe state and the risk to nuclear safety is ALARP. To meet this requirement, the generic CSR document structure contains a dedicated section on the demonstration of withstand to internal and external hazards. I consider this to be important because early plant layout proposals indicate there are a number of key safety significant pressure vessels and tanks in close proximity to the CVS, including the safety injection accumulators and local ultimate heat sink reservoirs. Gross failure of these could generate internal hazards, such as significant localised impact loading to the CVS (missiles, pipe whip, steam jetting) or flooding.
8. At the time of my GDA Step 2 assessment, the specific CVS CSR was not available for review. Based on my understanding of the RP’s generic CSR document structure and content, I am satisfied that the CVS CSR has the capacity to address ONR expectations for the demonstration of the CVS to withstand internal and external hazards. The specific evidence should be sampled in more detail when the necessary supporting CVS information is available. I consider this is a residual matter.
9. One further aspect for consideration in my assessment is the CVS support structure. This component provides an important interface between the steel pressure vessel and the civil concrete building structure.
10. From my review of the relevant chapters in the E3S case (namely Chapters 6 and 9B), I noted that there was limited detail provided for the CVS support structure. This is relevant to both the structural integrity and civil engineering disciplines regarding selection and demonstration of relevant design code compliance. I raised RQ-01084 (ref. [48]) to seek clarification on the codes and standards applicable to the CVS support and where the applicable boundaries are for the CVS component supports (defined in ASME III under Subsection NF - Supports) and civil structural building supports.
11. The RP’s response explained that the CVS support is considered under E3S Chapter 9B because, despite being a support for a pressure vessel which is included within the scope of ASME III Subsection NF (Supports), the RP considers it to be better aligned to the civil engineering topic area. This is due to the materials and construction method of the CVS support (response to RQ-01084, ref. [48]), being considered under the claims on the ‘base mat and containment support structure’(E3S Chapter 9B). There is no reference to ASME III Subsection NF in this part of the E3S case.
12. I, along with the ONR civil engineering inspector, sought to determine the discipline-specific jurisdiction for the CVS support structure within GDA through raising RQ-01231(ref. [48]). The RP’s response explained that the CVS Support Structure and the anchorages will be designed to civil engineering codes (ACI-349 M) and that the CVS will be built to the applicable Subsection of ASME III (NE). The RP explained that that a future issue of the Structural Design Method Statement (ref. [67]) will be produced, which will include an expanded methodology for items forming interfaces between civil engineering and structural integrity claims, including the CVS to containment support structure interface.
13. There was no explicit reference of ASME III Section NF (supports for safety classified components) in the response to RQ-01231, however the ‘Containment System Design Description’ provided in response to RQ-01084 (ref. [48]) identifies that the CVS will be designed and manufactured in accordance with ASME III, Subsection NF - Supports.
14. I consider that the statements provided in the RQ responses and the information contained in the safety case are somewhat ambiguous as to which codes and standards will be followed for the CVS structure. This ambiguity appears to be a circumstance of the maturity of the E3S case to fully reflect the ongoing support structure design development. In my opinion, the ambiguity of referencing codes and standards for the CVS support structure is not a significant concern at this stage of GDA, but will need to be confirmed within the E3S case when the CVS support structure design is mature.
15. Overall, from my assessment of the structural integrity aspects of the Rolls-Royce SMR design CVS, I have been able to confirm that the E3S case considers the performance of the containment during severe accidents, including structural integrity and stability of CVS subcomponents, welds and supports using adequate codes and standards, with capacity to demonstrate withstand to internal and external hazards.
16. I expect specific evidence and detailed justification of structural integrity subclaims to be presented within the CVS CSR. At this stage of GDA, I do not consider the lack of available detailed evidence to be inhibitive, however I do consider it important for these aspects to be included as evidence for future assessment of the E3S case. I consider this to be a residual matter.

### Materials selection, ageing, degradation and through life reliability

1. ONR guidance provides the expectation that safety significant SSCs are constructed from materials with well-established properties and behaviour (sap EMC.13, ref. [11]). The use of proven materials is fundamental to the integrity of safety significant components from both a mechanical properties and environmental compatibility perspective. I therefore considered the RP’s approach to materials selection and management of ageing and degradation through life to be key aspects for sampling within GDA.

#### Materials selection

1. I sampled the E3S case to assess the RP’s approach to materials selection and how it ensures fit-for-purpose material is proven within its operating environment. SC 23.1.5 states that “materials are selected and specified to ensure they are well-understood and characterised, based on OPEX and RGP”.
2. The E3S case (ref. [8]) notes two distinct stages for defining materials; materials selection; and materials specification. I sought further clarification between these two aspects in RQ-01198 (ref. [48]). In its response, the RP explained that the materials selection stage determines the broader material grade, whereas materials specification is the development of the detailed specification of that material, where this differs from the commercial specification (for example, provided in ASME II). The latter considers additional, specific Rolls-Royce SMR design factors such as chemical composition ranges to be achieved or any additional mechanical testing. For ASME III components, materials have been selected from those allowed by ASME II and III, which I consider to be RGP. I am satisfied that this approach broadly aligns with ONR expectations for the use of proven materials in design.
3. I reviewed the materials selection process under SC 23.1.5 to assess how it meets ONR expectations for selecting materials that will enable an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility (EMC.13). The RP follows a generic design optioneering process, C3.2.2-2 (ref. [39]), which the E3S case (ref. [8]) explains is a structured design optioneering process for down-selecting between materials. I considered that the approach was logical, thorough with a requirement to consider OPEX and RGP. However, as a generic decision making process, it lacked specific materials selection criteria and guidance. I therefore reviewed the RP’s wider submission to understand the materials selection criteria.
4. The SIR (ref. [35]) states under SIR no. 211 that “the considerations made in materials selection beyond the duty requirements include through-life radiation dose implication, legislation, ALARP, Best Available Technology, cost, supply chain availability, standardisation and OPEX”. While the ALARP justification was part of the considerations, I regarded this was not in itself sufficient detail to define how a robust decision is made. Through my engagements with the ONR Chemistry inspector, I was made aware of guidance provided elsewhere in the E3S case on materials selection, provided under the ‘Primary Water Chemistry – Minimisation of Corrosion of Structural Materials’ (ref. [68]), which supports E3S case Chapter 20: Chemistry (ref. [7]). I reviewed this document, which provides a number of factors for consideration during materials selection, including functional and non-functional requirements of the SSC, through life degradation, ease of manufacture, fabrication, inspection and maintenance. I considered these factors for consideration aligned well with ONR expectations. In its response to RQ-01198 (ref. [48]), the RP explained that these factors were not a set of requirements for materials selection but presented as context for the document. While I consider that the RP has an understanding of the factors required for materials selection as demonstrated in (ref. [68]), it is currently unclear where the RP has incorporated this in their process. I therefore reviewed the RP’s design optimisation process (C3.2.2-2) in more detail.
5. The C3.2.2-2 (ref. [39]) process is structured using a preset format of steps or stages to inform decision making, with each step becoming more detailed as necessary. The process and outcome is captured under a decision record, which is a standardised format referenced as TS-DD-02 (ref. [69]). I reviewed this document to assess how and to what extent nuclear safety was considered as a contributing factor.
6. I noted that “impact on safety” was considered and used to inform the selection of the analysis tools, which include one or more options entitled: “Advantages and Disadvantages”; “Red-Amber-Green” scoring; and a “Pugh Matrix”, based on pre-determined weightings for each factor. In my opinion, the majority of the criteria on the Pugh matrix were not safety related or considered aspects important for materials selection.
7. From my review of the RP response to RQ-01198 (ref. [48]), the RP notes that specific guidance on using the C3.2.2-2 process for materials selection has been produced. The RP elaborated that this material-specific guidance is used by the materials engineers within the materials team when technically checking any decision records to provide consistency in approach. I considered that reference to this guidance as part of the supporting evidence for the E3S case structural integrity claim would be important to provide context and assurance in the substantiation of materials specification.
8. Overall, I consider the E3S case demonstrates that the RP are following a structured process for decision making, however the adequacy of this process for ensuring that a robust materials selection decision, based on important, technical aspects for the individual components, has not been fully demonstrated within Step 2. The RP’s proposal to produce material specific guidance to inform the decision making process is encouraging, which I consider to be reasonable at this stage of GDA.
9. In my opinion this lack of evidence within Step 2 is not a significant shortfall, however I expect further component-specific evidence to be presented within the E3S case to demonstrate how this approach has been applied for the design of components, and whether a reasonable level of material specific scrutiny has been applied through processes that may not currently be apparent in the E3S case. I expect further evidence to be presented within the E3S case to substantiate the RP’s materials selection approach, including the materials selection specific guidance being developed. I consider this to be a residual matter.
10. From my broader review of the C3.2.2-2 process, I considered that it lacked guidance on who makes the decisions and how the component lead/IPT members are selected. In its response to RQ-01198 (ref. [48]), the RP explained that the C3.2.2-2 process for materials selection is initiated by the relevant component IPT and is managed through the RP’s GR process (C3.2.1-3, ref. [39]) which ensures appropriate stakeholder engagement. I noted there is no specific guidance on the selection of the IPT, which in my opinion does not reflect the importance of having an appropriately represented team of Suitably Qualified and Experienced Persons (SQEP) to make materials selection decisions. I therefore expect the decision making records for materials selection and specification produced from the C3.2.2-2 process for a range of safety significant Rolls-Royce SMR design to be available as part of the E3S case. This should include evidence of a material-selection SQEP in the IPT. I consider this to be a residual matter.
11. SC 23.1.5 notes that ageing management is considered in the selection and specification of material for the design of components. The RP has produced an Ageing Management Plan (AMP), through which Technical Justifications (TJ) will be produced to outline the management of each identified ageing and degradation mechanism. I have sampled the AMP separately as part of my Step 2 assessment (see Section 4.2.5.2 below), however I wanted to understand how ageing and degradation considerations were accounted for in the materials selection and specification process, and vice versa. This link is important to minimise the known risk of degradation in the materials selected and inform design decisions. From my sampling, I was able to identify that, where the materials selection process highlights that materials are required outside the scope of the TJ for a credible degradation mechanism, then the TJ will be updated through a business change control process (C3.2.1-9). I am content that the RP’s approach enables TJ scope to be actively managed, to ensure all materials selected and specified are considered in the TJs and thus included in the ageing management programme.
12. Given the importance of safety classification in determining structural integrity controls, I considered it important to sample how safety classification influences the materials selection process. In the SIR (ref. [35]), SIRs no. 215 and no. 234 describe how greater scrutiny and tighter chemical composition limits will apply for VHR/HR materials and that for Safety Class 1 and 2 materials, consideration of any RGP beyond ASME II specifications is still accounted for in the material specification. In RQ-01198 (ref. [48]), I additionally asked where classification is considered in the materials selection/specification processes. The RP responded that the decision level is prescribed by the C3.2.2-2 process, and a VHR/HR classification would drive a more onerous decision level on materials selection. I was satisfied with the response, which confirms that the C3.2.2-2 process accounts for a range of safety classifications and is graded to enable more rigour in the selection and specification of materials for higher classification components.
13. In consideration of these points, I am generally content that there are no significant shortfalls with the RP’s approach to materials selection, to provide confidence that the Rolls-Royce SMR design uses proven materials in the broadest description. In my opinion, this approach generally aligns with Requirement 47 of the IAEA SSR2/1 (Ref. [28]) for quality of materials and SSG-56 for material properties and characteristics as a specific consideration in the design of the reactor coolant system pressure retaining boundary and associated systems (Ref. [30]).
14. Whilst I am broadly satisfied with the RP’s approach to materials selection and consider it to be in line with ONR expectations, I have noted several areas where I consider the E3S case would benefit from further clarification on the RP’s approach for refinement of material specification. I consider this to be a residual matters that I will take forward as part of my Step 3 assessment.

#### Management of ageing and degradation

1. Subclaim 23.3.1 claims that “the ageing and degradation of materials is actively managed and monitored where appropriate”. I have assessed the RP’s proposal for the management of ageing and degradation against SAPs EAD.1-5 and EMC.3 and IAEA standard SSG-48 (ref. [31]).
2. The RP’s ageing and degradation approach is provided in the Rolls-Royce SMR design AMP (ref. [44]). The scope covers all reactor island metallic components that are safety class 2 and above, noting the scope will expand to include other plant areas in later versions. I considered the current scope of the AMP to be acceptable for the purpose of my Step 2 assessment.
3. The RP claims that the AMP structure and content has considered RGP (including ASME III, EPRI and IAEA guidance) to “identify possible and credible degradation mechanisms, which have been used to inform the production of supporting information, identify knowledge gaps and define approaches on how to fill these gaps. Ageing management is a well-defined topic that has been developed from decades of PWR operation with a broad range of OPEX and RGP available to the industry”. I reviewed the scope of RGP identified in the AMP and considered it was thorough and relevant with no obvious sources missing. I am satisfied that the RP’s submission takes cognisance of this existing knowledge to structure its AMP and that the approach demonstrates a comprehensive examination of relevant scientific and technical issues, taking account of precedent.
4. The majority of OPEX for environmental degradation of components in light water reactor designs is on LiOH plus boron primary water chemistry. This presents a knowledge gap for the proposed operation of the Rolls-Royce SMR using a KOH non-boronated primary water chemistry regime. I therefore reviewed the AMP to understand how structural integrity aspects for material compatibility (SAP EMC.13) with the proposed primary water chemistry was considered (see Section 4.2.5.1 above).
5. The RP’s approach has conducted a literature review to identify mechanisms that could be affected by the proposed Rolls-Royce SMR design primary water chemistry regime. The RP’s approach prioritises production of TJs for relevant mechanisms as a priority, to counter any knowledge gaps that may impact the Rolls-Royce SMR design. The RP has recommended a programme of testing to verify that performance is no worse in the proposed Rolls-Royce SMR design primary chemistry than typical PWR chemistries. The scope and purpose of testing requirements are recorded in the TJs, which will be issued when complete. I am satisfied that the RP has considered the implications of the knowledge gaps associated with the proposed water chemistry on degradation and has implemented a systematic approach to manage this risk. Given the importance of material compatibility in demonstrating structural integrity of the primary circuit SSCs, I consider it necessary to sample how these knowledge gap risks have been identified and confirm they have been adequately managed through robust and evidential substantiation. I expect the relevant TJs to demonstrate how KOH–structural material compatibility has informed materials selection. I consider this to be a residual matter.
6. I noted that the AMP includes a strategy for dealing with the potential for interplay between different materials degradation mechanisms, which in my opinion broadly aligns with ONR expectations (SAP EAD.2). I sampled how degradation mechanisms were selected as in-scope in the AMP. The RP reviewed a number of sources from international OPEX, from which it compiled a list of potential degradation mechanisms. The RP explained that a TJ is produced for each degradation mechanism applicable to the Rolls-Royce SMR design and with respect to a range of materials defined in the AMP. I was content that the selection process of degradation mechanisms was sufficiently thorough and well-reasoned. I reviewed the approach, which I considered included a broad range of materials, including weld materials (noting that materials selection for the Rolls-Royce SMR design is still ongoing – see Section 4.2.5.1).
7. The AMP explained that the TJs include any associated risks, knowledge gaps and, if applicable, the test programme requirements to fill these gaps. The TJs will also consider whether inspection, routine replacement or mitigation during manufacture are required against a particular degradation mechanism. In my opinion, this approach is reasonable, however I considered it prudent to sample several TJs to understand how the content supports the higher level claim of being robust against a particular degradation mechanism, for a specified list of component materials. The information is presented in each TJ using an easy-to-follow claims, arguments and evidence formant, which I considered contained sufficiently in-depth descriptions of the degradation mechanisms, reference to relevant RGP, identified risks, knowledge gaps and risk management techniques. I judged that the contents and purpose of TJs outlined in the AMP is in line with ONR expectations.
8. I also sought clarification on the process by which TJs are produced to gain assurance that relevant and SQEP personnel are involved (RQ-01197, ref. [48]). The RP’s response explained that the GR process (C3.2.1-3,) is followed for the production of TJs, in which relevant stakeholders and a relevant lead (chair) are selected, with the resource (including author) for the task confirmed as SQEP. I am content that the RP has implemented an approach to enable sufficient governance and identify relevant SQEP personnel to produce the TJ.
9. Due to the importance of incorporating ageing and degradation considerations into the SMR design, I sought information on the process by which this is managed (RQ-01197, ref. [48]). The RP responded that appropriate design considerations identified in the TJs will be adapted into requirements (as part of the SIR) and then managed through the Design Review process (C3.2.1-2) for incorporation into the CSRs. The RP explained that other factors will support an overall ALARP justification for the component design or plant chemistry. I have elsewhere considered the link between materials selection and ageing and degradation and considered it acceptable (see Section 4.2.5.1). I am therefore content that the RP has considered the incorporation of ageing and degradation design considerations across the Rolls-Royce SMR design, with appropriate management processes in place. I consider it necessary to assess further how this approach has been implemented at a component-specific level to gain further assurance of this process. I consider this to be a residual matter.
10. The AMP states that any analysis should use conservative assumptions and limits to provide a margin between the operating and fault envelope for plant and the point at which failure from that degradation mechanism could first potentially occur. Considering ONR expectations in SAP EAD.2 (Lifetime Margins), I raised RQ-01197 (ref. [48]) to understand how these limits are defined. The RP responded that safe levels/limits will be defined within the TJs and that the appropriate analysis methods to calculate the safe life of a component or sub-component will be identified and justified. However, the margin between operational life and predicted safe working life will be demonstrated on a component basis. This will be captured within the CSRs, to ensure an adequate margin is demonstrated that is commensurate with the safety classification. I am satisfied with the RP’s response and consider it necessary to asses further how this is implemented at a component-specific level. I expect the CSRs to demonstrate how the RP’s approach for ageing and degradation has informed margins to demonstrate and justify safe working life of components. I consider this to be a residual matter.
11. Finally, I reviewed the AMP to understand how classification was considered in the application of the AMP to components design. I noted that the AMP outlined how highest reliability components could warrant additional claims to substantiate the impact of relevant degradation mechanisms on structural integrity and component design. These considerations, such as additional chemical composition restrictions, changes in manufacturing or further mechanical testing would be considered in the TJs. In addition, the AMP reported that for safety class 3 or non-safety classified components and materials, a simpler approach may be taken for degradation management, but this will be included in later editions of the AMP. I am content that the RP has developed an ageing management approach that is graded according to classification.
12. Overall, I am broadly satisfied that the RP’s proposed process for the management of ageing and degradation is in line with ONR expectations. In my opinion, this approach generally aligns with Requirement 31 of the IAEA SSR2/1 (Ref. [28]) for ageing management, SSG-48 on ageing management and SSG-56 for the consideration of ageing in the design of components and environmental qualification of items important to safety (Ref. [30]). I also consider that this approach satisfies Issue I of the WENRA reference safety levels and safety objectives for new plants (ref. [21] and [22]) on ageing management, such that the RP’s AMP approach identifies in a systematic and knowledge based manner, all relevant potential degradation mechanisms and their ageing effects, determine their possible consequences and the necessary activities to ensure and monitor the availability and reliability of in-scope SSCs.
13. As part of a balanced and thorough assessment, I expect the E3S case (CSR and TJs) to demonstrate how material compatibility risks associated with the proposed primary circuit water chemistry regime are managed. This should include demonstration of how the AMP is used to inform design decisions at a component level, how design considerations incorporate structural integrity ageing management requirements and how ageing management derives margins and safe working life of SSCs.

#### Irradiation embrittlement surveillance strategy

1. ONR guidance identifies that where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties (SAP EAD.3). As such, surveillance programmes should include representative surveillance material specimens and test programmes to provide adequate forewarning of detrimental material property changes throughout the life of the facility.
2. As part of my assessment, I sampled the RP’s consideration of structural integrity within the compact layout of the Rolls-Royce SMR design. I sought to understand how the risk of a compact RPV and core may affect the management of through life degradation and irradiation surveillance of SSCs, particularly with ensuring sufficient capacity for representative and robust through-life surveillance. I also sought to understand whether the compact design could increase the rate of irradiation damage or the uncertainty in the prediction of irradiation embrittlement of the RPV.
3. The RP identifies under SC 23.3 that ageing management of materials is managed through pro-active identification and consideration in the design phase, with a programme in place for irradiation surveillance of RPV materials. There is no further reference of how this will be delivered or where in the E3S case such an approach will be provided.
4. From my previous review of the RP’s AMP (see Section 4.2.5.2), I noted that irradiation embrittlement had been identified by the RP as a relevant degradation mechanism. In my opinion, a justification of how this type of degradation mechanism is managed through life should consider material surveillance. I therefore sampled the relevant TJ on irradiation embrittlement (ref. [70]), to understand how the RP has considered surveillance and whether the provision of sufficient surveillance specimens could be impacted by the compact design.
5. Claim 2.4 of the Irradiation Embrittlement TJ (ref. [70]) proposes a surveillance programme to monitor changes in the mechanical properties of RPV steels, for which the standard ‘ASTM E185’ will form the basis. I considered that this approach follows a recognised standard, although to meet the expectations in (ref. [24]), additional aspects should be considered to ensure test samples are representative. These may include test sample quantity, type (based on mechanical testing expectations), size/geometry and source location. Claim 2.4 also noted that all materials, including weld materials, exposed to peak neutron fluence at the end of design life, will form part of the Rolls-Royce SMR design surveillance programme. I am content at this stage of GDA that the RP’s proposed surveillance programme has considered RGP on the testing and surveillance of representative materials through life. In my opinion, this approach generally aligns with Requirement 30 of the IAEA SSR2/1 (Ref. [28]) for qualification of materials important to safety and SSG-56 for materials exposed to high neutron flux (Ref. [30]).
6. From my review of the high level information provided in the Irradiation Embrittlement TJ, I am content that the RP’s approach considers both the change in transition temperature and upper shelf energy due to irradiation embrittlement. However, I noted several technical aspects related to the RP’s methods for predicting and modelling irradiation embrittlement, in response to RQ-01238 raised on the topic (ref. [48]), that requires further consideration. In my opinion, these aspects are related to detailed technical evidence, which is not currently available within the E3S case. I expect the RP’s reasoning and justification for its approach and how it ensures risk is reduced ALARP to be presented within the E3S case. I consider this to be a residual matter
7. At the time of my assessment, the RP’s irradiation surveillance strategy (ref. [71]) was not available for assessment, I therefore consider it is necessary to assess the evidence of this within the E3S case, to support my assessment of the RP’s claims on the management of ageing and degradation mechanisms. I expect the E3S case to demonstrate how the approach, scope and content of testing in the surveillance program proposal has influenced design of the RPV internals to accommodate the surveillance specimens. This is a residual matter.

# Conclusions

## Conclusions

1. This report presents the Step 2 structural integrity 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 case 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 structural integrity against the expectations of ONR’s SAPs, TAGs and other guidance which ONR regards as relevant good practice.
2. Based upon my assessment, I have concluded the following:

* The Rolls-Royce SMR design E3S case explicitly considers structural integrity, with clear and traceable links from plant safety functional requirements to the structural integrity measures and controls applied to components. Structural integrity requirements are well substantiated, based on relevant OPEX and international guidance.
* The RP’s approach to component classification is reasonable, based on both direct and indirect consequences of gross failure, which aligns with ONR expectations. Where consequences are demonstrated to be intolerable, additional provisions for ‘beyond code’ integrity demonstration are proposed to ensure risk is reduced ALARP. I consider the RP’s approach of multiple beyond code classifications requires further assessment, to understand how proposed measures for inspection are proportionately reduced with the tolerability of gross failure. The adequacy of consequence analysis undertaken to support this approach should be demonstrated at a component-specific level.
* The RP has a good understanding of ONR expectations for highest reliability components and has established at a high level, an adequate beyond code approach through a conservative avoidance of fracture demonstration (AoFD). The AoFD is applied for the highest safety classification components, which combines; DTA using the R6 fracture mechanics methodology; representative materials testing to underpin conservative fracture toughness values used in the DTAs; and proposals to qualify the manufacturing NDE using the ENIQ methodology. This aligns with ONR expectations for such approaches, however further evidence is required to demonstrate how the technical inputs of avoidance of fracture are derived and reconciled to achieve a conservative and achievable target reserve factor.
* In my opinion, the RP has selected a well-established set of nuclear specific codes and standards to inform the design of the most safety significant SSCs, with provisions made in the RP’s approach to optimise design aspects and modify code requirements, where reasonably practicable. I consider this approach meets ONR expectations. A number of specific code cases and alternative standards for manufacturing assurance are proposed, the adequacy of which I expect to be presented within the E3S case in Step 3 of GDA.
* The RP’s approach to component design includes a set of structural integrity requirements to ensure access for inspectability. Whilst this approach is encouraging, the compact nature of this design means it is heavily reliant on the layout of major components and structures, for which a sufficient level of detail has not yet been submitted to enable assessment. I consider this requires further assessment when component-specific information is available, to confirm the requirement to ensure access for inspectability is appropriately implemented.
* The containment vessel structure design is based on an established nuclear code, which at a high level meets ONR expectations. The detailed substantiation of proposed structural integrity controls requires further evidence, such as the proposed use of a code case exemption to apply post weld heat treatment during manufacture and the selection and application of relevant civil engineering and mechanical engineering codes and standards for the CVS support structure.
* The RP’s proposed use of a seismic isolation safety feature directly influences the structural integrity design aspects of pressure boundary components associated with the Rolls-Royce SMR design primary circuit, secondary circuit and safety systems. Assessment of the adequacy of seismic isolation to fulfil the safety functional requirements is being undertaken by the ONR civil engineering and external hazards disciplines, meaning it is out of scope of the structural integrity assessment. Confirmation must therefore be sought that ONR is satisfied with the proposed seismic isolation system design, such that it can offer the reduction in dynamic loading of SSCs subject to structural integrity claims.
* The materials selection approach for the Rolls-Royce SMR design follows the RP’s generic decision making process, referencing the use of proven materials from codes and standards with provisions to optimise the material properties where necessary. This optimisation of material specification is related to SSC safety significance and is informed by ageing and degradation mechanisms, where reasonably practicable. I consider this approach is broadly aligned with ONR expectations, however further assessment is required to understand how multi-discipline engagement on materials selection is managed to ensure all relevant aspects for structural integrity are considered and that the outcomes are consistent and informed by SQEP judgement.
* The RP has established a robust approach for the management of ageing and degradation mechanisms, based on a comprehensive review of international OPEX, guidance and relevant good practice. In my opinion, the RP’s approach has sufficient provisions to identify knowledge gaps and implement satisfactory resolution plans through physical testing and technical substantiation, particularly for shortfalls in OPEX related to material compatibility with the selected primary coolant chemistry regime.
* The RP has stated that the compact nature of the Rolls-Royce SMR design does not inhibit the implementation of a robust and comprehensive material irradiation surveillance strategy. The scope, content and range of specimens indicated at this early stage of GDA has the potential to meet ONR expectations for ensuring data collected through life is suitably representative and bounding, however this information is not currently presented in the E3S case. Further assessment is required when the information is available, including substantiation of the RP’s approach for predictive modelling of irradiation damage, to ensure it is conservative and informed by OPEX to ensure risk is reduced ALARP.
* The E3S case makes reference to the use of multidiscipline, integrated project teams and requirements management to inform structural integrity provisions. The approach also ensures structural integrity requirements are considered in the design and safety substantiation of SSCs that inform other high level E3S case claims. In principle, this approach aligns with ONR structural integrity expectations for consideration of all relevant scientific and technical issues, however further evidence is required to understand how members of IPTs are selected and managed, with subsequent decisions recorded at a component-specific level. This should demonstrate how multiple design decisions for a single component are appropriately reconciled to ensure design features reduce risk ALARP and that the outcomes are promulgated to relevant E3S case chapters and stakeholders.
* Throughout my Step 2 assessment, the design status and maturity of the Rolls-Royce SMR design has resulted in a number of uncertainties associated with the RP’s E3S case. As such, key aspects to inform my assessment such as plant layout, component classification, bounding transient loadings and component design optimisation are still ongoing. Whilst I consider this acceptable for a Step 2 assessment, there remains a significant amount of detailed information required to underpin the E3S case structural integrity claims. I will assess this further as part of my GDA Step 3 structural integrity assessment.

1. 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.

## Recommendations

1. My recommendations are 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

| SAP No. | SAP Title |
| --- | --- |
| SC.4 | The regulatory assessment of safety cases. Safety case characteristics |
| EKP.3 | Engineering principles: key principles. Defence in depth. |
| ERL.1 | Form of Claims |
| EMC.1 | Integrity of metal components and structures: highest reliability components and structures.  Safety case and assessment |
| EMC.2 | Integrity of metal components and structures: highest reliability components and structures: Use of scientific and technical issues |
| EMC.3 | Integrity of metal components and structures: highest reliability components and structures: Evidence |
| EMC.4 | Integrity of metal components and structures: General. Procedural control |
| EMC.5 | Integrity of metal components and structures: General. Defects |
| EMC.6 | Integrity of metal components and structures: General. Defects |
| EMC.7 | Integrity of metal components and structures: Design. Loadings |
| EMC.8 | Integrity of metal components and structures: Design. Requirements for examination |
| EMC.9 | Integrity of metal components and structures: Design. Product form |
| EMC.10 | Integrity of metal components and structures: Design. Weld positions |
| EMC.11 | Integrity of metal components and structures: Design.  Failure modes |
| EMC.12 | Integrity of metal components and structures: Design.  Brittle behaviour |
| EMC.13 | Integrity of metal components and structures: Manufacture and installation. Materials |
| EMC.17 | Integrity of metal components and structures: Manufacture and installation. Examination during manufacture |
| EMC.21 | Integrity of metal components and structures: Operation.  Safe operating envelope |
| EMC.23 | Integrity of metal components and structures: Operation. Ductile behaviour |
| EMC.24 | Integrity of metal components and structures: Monitoring.  Operation |
| EMC.27 | Integrity of metal components and structures: Pre- and in-service examination and testing. Examination |
| EMC.28 | Integrity of metal components and structures: Pre- and in-service examination and testing. Margins |
| EMC.29 | Integrity of metal components and structures: Pre- and in-service examination and testing. Redundancy and diversity |
| EMC.30 | Integrity of metal components and structures: Pre- and in-service examination and testing. Control |
| EMC.32 | Integrity of metal components and structures: Analysis. Stress analysis |
| EMC.33 | Integrity of metal components and structures: Analysis. Use of data |
| EMC.34 | Integrity of metal components and structures: Analysis. Defect sizes |
| EAD.1 | Ageing and degradation. Safe working life |
| EAD.2 | Ageing and degradation. Lifetime margins |
| EAD.3 | Ageing and degradation. Periodic measurement of material properties |
| EAD.4 | Ageing and degradation. Periodic measurement of parameters |
| ECS.1 | Safety classification and standards. Safety categorisation |
| ECS.2 | Safety classification and standards. Safety classification of structures, systems and components |
| ECS.3 | Safety classification and standards. Standards |
| ELO.1 | Access |
| ELO.4 | Minimisation of the effects of incidents |