Partner Document Number	Partner Document Issue /Revision	Retention category:
n/a	**n/a**	**A**

Title

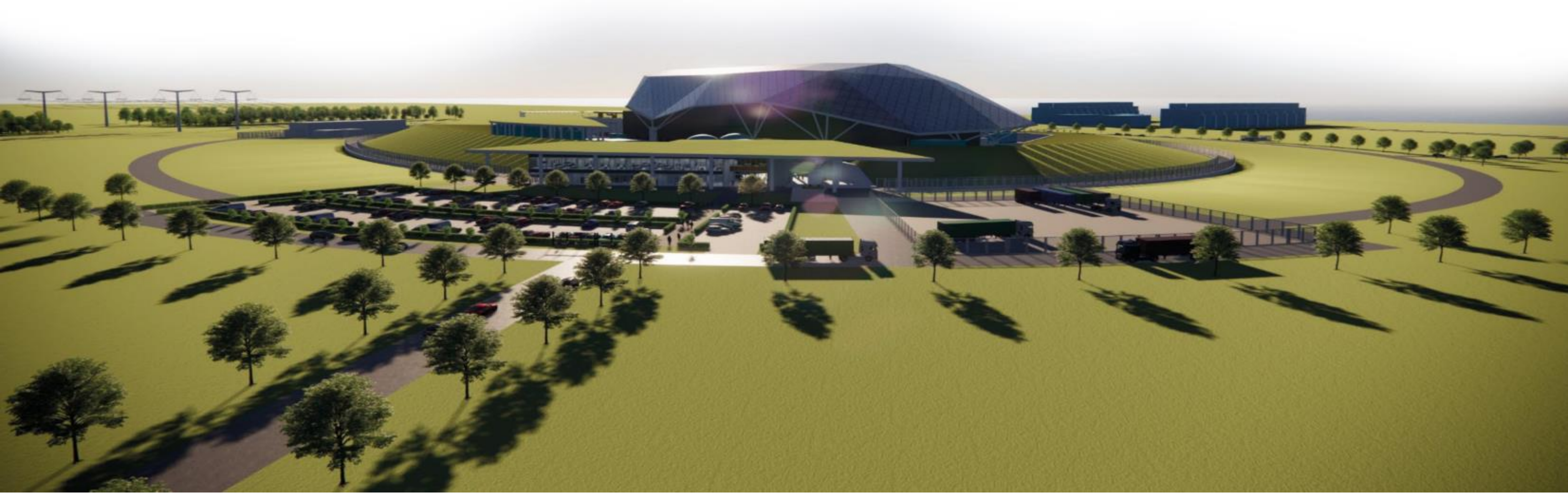
E3S Case Chapter 23: Structural Integrity

Executive Summary

This chapter of the Environment, Safety, Security, and Safeguards (E3S) Case presents the demonstration of structural integrity for the Rolls-Royce Small Modular Reactor (RR SMR).

The chapter outlines the sub-claims and preliminary arguments supporting the overall Claim that the structural integrity of Structures, Systems and Components (SSCs) is justified, and the risk of structural failure is minimised to As Low As Reasonably Practicable (ALARP).

At the Preliminary Concept Definition (PCD) design stage, the detailed arguments and underpinning evidence are still under development. The full suite of arguments and evidence will be developed and presented in future revisions of the E3S Case as the design programme matures, including definition of structural integrity requirements, methods for analysis and substantiation, and ultimately substantiation of specific components.



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23.1 Introduction

23.1.1 Introduction to Chapter

Chapter 23 of the Rolls-Royce Small Modular Reactor (RR SMR) Environment, Safety, Security & Safeguards (E3S) Case forms part of the Pre-Construction Safety Report (PCSR), as defined in E3S Case Chapter 1: Introduction, Reference [1].

Chapter 23 presents the overarching summary and entry point to the demonstration of structural integrity for the RR SMR, as defined at Reference Design (RD) 5 level of design maturity.

23.1.2 Scope

This chapter describes the Claims associated with the substantiation of 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; other functional requirements are addressed under the relevant chapter of the E3S Case for the component; an overview of the contents of each chapter is provided in E3S Case Chapter 1: Introduction, Reference [1].

Concrete structures and steel building structures are not within the scope of this chapter.

Design/Programme Maturity

RR SMR design information presented in this revision of the PCSR is largely based on the design definition at the end of Preliminary Concept Definition (PCD). PCD is an interim design stage representing RD5 level of design maturity, at which point structural integrity has presented the key Claims and Arguments that support demonstration of structural integrity for RR SMR, noting the underpinning Evidence will be developed as the RR SMR design progresses.

23.1.3 Claims, Arguments, Evidence Route Map

The Chapter level Claim for E3S Case Chapter 23: Structural Integrity is:

Claim 23: Structural Integrity of SSCs is justified and the risk of structural failure is minimised to As Low As Reasonably Practicable

This is decomposed into the following Sub-Claims, addressed in the corresponding sections of this report:

1. Sub-Claim 1: **Reliability is achieved** through high quality design, manufacture, and testing (Section 23.2).

2. Sub-Claim 2: **Reliability is demonstrated** for High Reliability (HR) and Very High Reliability (VHR) components through a robust avoidance of fracture case (Section 23.3).

3. Sub-Claim 3: **Reliability can be maintained** with the provision of effective in-service systems and procedures. (Section 23.4).

Sub-Claims 1 and 3 apply to all components within scope of this chapter, but with a graded approach applied, as described in Sub-Claim 1.1. Sub-Claim 2 (Section 23.3) is specific to higher reliability components (HR and VHR classifications – see Sub-Claim 1.1 (Subsection 23.2.2)); code compliance is considered sufficient for all other classifications.

This Claim structure has been informed by the UK Technical Advisory Group for Structural Integrity for High Integrity Plant (TAGSI) multi-legged approach described in Reference [2].

Specific Component Substantiation Reports (CSRs) will be produced to compile the component-specific arguments and evidence for key components or groups of components. These will implement this same Claims, Arguments, Evidence (CAE) structure for the SI function, with further decomposition as required.

Other Tier 2 documents will provide generic methods and policies which will be applied across multiple components. Key examples of these documents are:

1. Structural Integrity Substantiation Report, Reference [3]
2. ASME III Adaptation, Reference [4]
3. Defect Tolerance Assessment Guide, Reference [5]
4. Fatigue Assessment Guide, Reference [6]
5. Ageing Management Plan, Reference [7]
6. Inspection Qualification Strategy, Reference [8]
7. Non-Destructive Examination (NDE) Guide, References [9] and [10]

A full decomposition of this Claim into Sub-Claims, Arguments, and link to the relevant Tier 2 Evidence will be presented in future revision of this report. The complete suite of evidence to underpin the Claims in the E3S Case will be generated through the RR SMR design and E3S Case programme and documented in the CAE Route Map, Reference [11], described further in E3S Case Chapter 1: Introduction, Reference [1].

23.2 Achievement of Reliability

23.2.1 Achievement of Reliability Claim

Sub-Claim 1

Reliability is achieved through high quality design, manufacture, and testing.

Argument

High quality design, manufacturing and testing is fundamental to achieving the required reliability. The structural integrity measures set out under Sub-Claim 1 are largely informed by the safety classification and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) III code classification processes, providing a graded approach.

For Safety Class 1 and 2 components, ASME BPVC Section III sets the minimum requirements. Higher reliability classifications are defined where the consequences of failure are unacceptable, for which beyond-code measures are established. Conventional design codes may be used in lieu of ASME BPVC III for safety class 3 components.

Achievement of high-quality design and manufacture is comprised of a number of sub-claims covering the following areas:

1. Sub-Claim 1.1: Structural integrity measures are invoked based on the component's role in protecting people and the environment

1. Sub-Claim 1.2: Components are designed to appropriate standards and requirements which provide suitable reliability

2. Sub-Claim 1.3: Components are designed for structural integrity, including a review of OPEX and RGP

3. Sub-Claim 1.4: Materials are selected and specified to ensure they are well-understood and characterised, based on OPEX and RGP

4. Sub-Claim 1.5: Components are fabricated and installed to appropriate standards using qualified and controlled processes

5. Sub-Claim 1.6: Code-based examinations provide adequate demonstration that manufacturing processes have been performed to an acceptable standard and within expected parameters

6. Sub-Claim 1.7: Pressure testing provides assurance on the integrity of the pressure boundary

7. Sub-Claim 1.8: Appropriate controls are implemented in design and specified for manufacture to provide assurance of compliance with the relevant requirements

23.2.2 Component Classifications and Grading

Sub-Claim 1.1: Structural integrity measures are invoked based on the component's role in protecting people and the environment

Argument

The grading of structural integrity controls described throughout this chapter is based on four criteria:

1. The **Safety Classification**: A top-down categorisation of functions and classification of Structures, Systems and Components (SSCs) is carried out for each system and sub-system. For components this is then further refined to consider classifications of sub-components. Higher reliability classifications are defined where the consequences of failure are unacceptable

2. For Safety Class 3 components, the **dose** to workers and the environment directly as a result of the release of contents from the component (applicable to Class 3 components only)

3. For ASME III components, the **ASME III Code Classification**: Components for which ASME BPVC III is the design code are further assigned an ASME III Code Class, based on definitions provided by the US Nuclear Regulatory Commission (NRC)

4. The **Seismic Performance Classification**: Based on the importance to the components' role in safety in the event of an earthquake.

Safety Classification

Component functional requirements are decomposed from the plant fundamental safety functions, and assigned a category based upon the unmitigated dose consequence and the role of the measure in delivering defence in depth. Components are then classified based on gross consequences of failure, assuming they no longer deliver the associated safety function. The process is described in Reference [12].

This method is followed for the safety classification of components, but further refinement is allowed for sub-components following the RR SMR Integrated Management System (IMS) process C3.2.2-9, Reference [13]. This process adopts a Failure Modes and Effects Analysis (FMEA) approach to understand the complete consequences of gross failure of sub-components and regions, including accounting for secondary consequences. This process is used to identify higher reliability components and sub-components.

Two additional higher reliability classifications are established in Reference [12], based on acceptance criteria defined in the RR SMR E3S Principles, Reference [14]. These classifications apply when the consequences of failure are not acceptable and additional assurance against component failure is required; the definitions are summarised as:

Very High Reliability:

Structural failure would lead to either an off-site release of dose exceeding 100mSv or no physical barrier intact to confine any substantial relocation of radioactive material. It is not reasonably practicable to provide control of the resulting conditions either within or beyond the design basis.

High Reliability:

Structural failure would lead to exceeding **{REDACTED FOR PUBLICATION}** on-site release or limited relocation of radioactive material, but with off-site release limited to less than 100mSv. 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.

The following components have been identified as candidates for higher reliability classifications (VHR/HR):

1. Reactor Pressure Vessel (RPV) and closure head
2. Pressuriser (PRZ) shell
3. Steam Generator (SG) primary head, tubesheet and secondary shell
4. Reactor Coolant Pump (RCP) casing
5. Reactor Coolant Loop (RCL) pipework
6. Main Steam Line (MSL) pipework
7. Main Steam Isolation Valve (MSIV)

Analysis is currently underway to confirm whether the higher reliability classifications of these components are required, and to refine the regions within the components to which they are applicable.

HR and VHR classifications will not typically be applied to redundant features or sub-components (e.g. bolted assemblies) as the criteria for those classifications would not apply to a single failure. Instead, actions will be taken to reduce the likelihood of multiple failures, e.g. minimising the likelihood of common cause failure and demonstrating margin against any dynamic effects which may lead to subsequent failures.

Safety Class 3 Components – Dose Limits

The design code for Safety Class 3 pressure boundary components is informed by the role that the component pressure boundary plays in directly preventing the release of radioactivity. Where reasonably practicable, further physical defence should be provided (e.g. bunding) which would instead carry out that safety categorised function and allow the safety classification of the component to be reduced. Where that is not reasonably practicable, the criteria for whether the component is considered to lead to a release of radioactivity corresponds to “Low doses” from Reference [14], i.e. > 0.01mSv off-site or > 0.1 mSv on-site.

ASME III Code Class

Where ASME BPVC III, Reference [15], has been selected as the design code, the relevant subsection of the Code for pressure boundary components will be selected to provide assurance of structural integrity and quality commensurate with the relative importance assigned to the individual items of the nuclear power plant (ASME BPVC III NCA-2120). The RR SMR ASME III Code Classification process is defined in IMS C3.2.2-5, Reference [16]. A baseline code classification is based on RGP from NRC Regulatory Guide 1.26, Reference [17]. Additional rules are introduced in the process to ensure that:

1. The ASME III Code Class is not higher than the safety class

2. VHR/HR components are always designed to Code Class 1 rules

3. ASME III Code Class does not reduce the classification where it has been elevated due to secondary consequences.

Seismic Performance Categorisation

The method for deriving Seismic Performance Categorisation (SPC) is provided in SMR0001391, Reference [18]. The SPC affects the required demonstration of functionality following a seismic event, with the following definitions:

1. SPC1 - any SSC which has an important safety functional requirement in response to a seismic event within or beyond the design basis. SSC is to remain fully functional during and after a Design Basis Earthquake (DBE)

2. SPC2 - any SSC which unmitigated could have an undesirable impact on a seismic performance class 1 SSC or the long-term management of a seismic event within or beyond the design basis. SSC is to retain limited functionality during and after a DBE

3. SPC3 - all other SSC. No seismic withstand requirements are defined for SPC3 SSC with respect to the DBE. However, all SSC are to be unaffected by repeated ground motion at the Operating Basis Earthquake (OBE) level.

23.2.3 Design Standards

Sub-Claim 1.2: Components are designed to appropriate standards and requirements which provide suitable reliability

Argument

Component designs are in accordance with recognised and appropriate design codes, which confer a level of reliability based on Operating Experience (OPEX) and international consensus. Additional requirements are introduced based on RGP and project policy, managed through an appropriate requirements management system.

Design Code Selection

The codes, standards, and regulations specified to control quality of design and manufacture embody extensive knowledge and operating experience. Compliance with these standards provides a minimum baseline foundation for assuring that structural reliability of the RR SMR components will be achieved.

In accordance with the RR SMR E3S Principles, Reference [14], 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 implementation of this for structural integrity is shown in

Table 23.2-1.

Table 23.2-1: Pressure Retaining Component Design Code Selection

Safety Class	Design Code
VHR or HR	ASME III Division 1 Subsection NB and beyond code requirements.
1 or 2	ASME III Division 1, with the subsection based on the ASME III Code Class.
3 Directly leads to release of dose: ≥ 0.1mSv on-site, or ≥ 0.01mSv off-site.	Either ASME III Division 1, or Conventional design codes supplemented to give equivalent technical requirements to ASME III.
3 Does not directly lead to release of dose: < 0.1mSv on-site, and <0.01 mSv off-site.	Conventional design codes.

The 2021 edition of ASME BPVC, Reference [15], has been selected as the design basis code for 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 the defined threshold. ASME BPVC III provides rules for materials, design, structural analysis, manufacturing, examination, over-pressure protection and quality assurance for nuclear components. It has been widely adopted throughout the world as the code of practice for design and construction of nuclear pressure vessels and components. There has been successful long-term operation of many nuclear pressure vessels designed and manufactured to ASME III requirements, which demonstrates that the Code provides a sound basis for safe and reliable operation.

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 Pressure Equipment Safety Regulations (PE(S)R) are applied where either conventional standards are adopted or failure of the component would not lead to an emission of radioactivity.

Design Loadings

Conservative analyses are carried out in accordance with the design code requirements to demonstrate tolerance to all design basis loadings. Design loadings are captured in each component design specification, with loading combinations defined for credible combinations of loads.

For ASME III components, NCA-2142 and RGP are used to establish the loading specification, along with consideration of storage, transportation, or installation conditions. Each service condition to which a component may be subjected to is assigned a service level for use in the ASME III design analysis:

1. Level A – Normal service conditions

2. Level B – Upset conditions

3. Level C – Emergency conditions

4. Level D – Faulted conditions

Seismic loading includes an assessment of:

1. An OBE, where continued operation needs to be maintained and repeated occurrence are demonstrated not to cause damage to the plant requiring shutdown for inspection or repair

2. A DBE, in which safe shutdown of the plant must be demonstrated. Integrity of the pressure boundary is to be maintained but it may necessitate subsequent removal of components for repair

Implementation of ASME BPVC Section III

<u>ASME BPVC III Adaptation</u>

A number of the ASME BPVC III requirements relate to its operation with institutions, federal laws and practices in North America, which are less practicable to apply outside of this region. To ease its application in the UK, adaptations to the ASME III requirements are specified in Reference [4] to make it more consistent with UK practice, relating to aspects such as Quality Assurance (QA), inspection, responsibilities and duties, reference codes and standards, and certification. The adaptations will seek to provide equivalent or enhanced assurance, without affecting the technical requirements of ASME III. This approach is consistent with previous implementations and proposals for the adoption of ASME III for UK design and construction.

Where components are provided by suppliers who are already accredited to ASME III NCA, the Code rules do not require adaptation.

<u>ASME BPVC III Design Analysis Approach</u>

Design by Analysis requirements are defined in ASME III NB/CD/E/G-3200, which for Subsections NB and NCD refer to Mandatory Appendix XIII "Design Based on Stress Analysis". ASME III provides specified stress limits for Design, Service and Test Conditions, which provide protection against a range of failure modes, including:

1. Plastic collapse

2. Incremental collapse (thermal stress ratchet)

3. Fatigue

4. Brittle fracture

Code requirements for demonstration of prevention of nonductile fracture are through the application of ASME III Nonmandatory Appendix G, which is stated in G-2120 to ensure a margin of about a factor of 2 on defect size, and accounts for any through-life toughness degradation. The RR SMR implementation of this approach is described in Reference [19].

Guidance on the RR SMR implementation of the ASME III design by analysis requirements is given in the ASME Design By Stress Analysis Guide, Reference [20].

ASME BPVC III Fatigue Analysis

Fatigue analyses are carried out as specified by ASME III, with additional requirements to address environmental fatigue. Industry methods have been developed to account for environmental fatigue, the most comprehensive of which is described in US NRC document NUREG/CR-6909, Reference [21], which defines an environmental correction, Fen, factor. The Fen factors in CR-6909 are based on an extensive compilation of international test results. However, it is considered that the CR-6909 approach has shortcomings which result in its over-conservatism, which is evidenced by the extensive OPEX from the worldwide fleet of light water reactors, justified without any additional Fen factors applied.

Fatigue analyses will therefore account for the detrimental effects of a PWR environment accounting for the latest methods and understanding. CR-6909, Reference [21], Fen factors will be calculated, but for austenitic stainless steel two refinements to the approach will be made which have not yet been adopted as ASME BPVC Code Cases but have been subject to extensive validation by Rolls-Royce plc and internationally:

1. The Fen-threshold approach, Reference [22], which effectively amends the design factors used to construct the design fatigue curve, based on an understanding of differences in effects in a PWR environment compared to an air environment

2. The SNW method, Reference [23], which provides a more accurate method of accounting for the parameters in the Fen equations where they vary during a loading cycle

The RR SMR fatigue assessment method is defined in Reference [6], with the basis for the method described in Reference [24].

To minimise the risk of high cycle fatigue due to rapid thermal mixing of fluid streams with a significant temperature difference, the guidelines developed by the Network for Evaluating Structural Components project in Reference [25] are adopted, supplemented by guidance from the Electrical Power Research Institute (EPRI).

Requirements Management

RR SMR use of requirements management processes C3.1.1, Reference [26], and tools ensure that requirements are invoked on the components based on the classification system. These are then embedded in the technical requirements for internal design or in the supply chain, and later embedded in the Design Specification.

23.2.4 Design Process and Decisions

Sub-Claim 1.3: Components are designed for structural integrity, including a review of OPEX and RGP

Argument

The new design of the RR SMR plant allows current best practice and technologies to be accommodated, considering improvements in material forming, welding and examination and incorporating lessons learned from OPEX in these areas. Systems engineering approaches ensure design is carried out in a systematic and structured way.

Design Process

Systems engineering approaches are adopted, implementing tools such as robust design and a structured design optioneering process defined in Reference [27] for any significant decisions, which ensures a balanced consideration of design options, accounting for factors which influence structural integrity along with RGP and relevant OPEX.

Technical risks to the design delivering its functions are identified and managed, using tools such as Design Failure Modes and Effects Analysis (DFMEA).

Design for X

Design for X (DfX) is embedded in working practices through the use of Integrated Project Teams (IPTs) which are made up from a range of functions such as design, structural analysis, procurement, manufacturing, examination and verification. Examples of implementation of DfX are given below.

<u>Examination Access</u>

The RR SMR E3S Principles in Reference [14] require designs to facilitate access for inspections. Components have therefore been designed for Ultrasonic Testing (UT), on the assumption that UT will be required for in-service inspection as well as manufacturing examination for higher reliability components, and for any potential deployment of UT in lieu of Radiographic Testing (RT).

<u>Material Forming</u>

The RR SMR E3S Principles in Reference [14] require designs to minimise the number and length of welds where reasonably practicable. In general this is achieved by maximising forging sizes while still achieving appropriate material properties. Forgings will also be used in preference to castings where practicable. Application of these principles reduces the likelihood of structurally significant defects; examples are:

1. Major components are constructed from forged material and not rolled plates, avoiding the requirement for seam welds

2. The RPV upper shell is a single forging including nozzles, without the need to weld nozzles into the shell

3. The RPV head is a single piece forging without the need for a circumferential weld to an additional torus forging

4. The RCP pump bowl will be constructed from a forging, not casting

<u>Material Degradation</u>

Material degradation mechanisms are considered in the design process, with requirements introduced from specific degradation mechanism justifications. For example crevices are minimised to reduce Stress Corrosion Cracking (SCC) concerns.

23.2.5 Material Selection & Specification

Sub-Claim 1.4: Materials are selected and specified to ensure they are well-understood and characterised, based on OPEX and RGP

Argument

Material selection is carried out through a systematic and robust process that accounts for OPEX and RGP, ensuring a fit-for-purpose material is used which is proven in the relevant environment. Materials are specified with appropriate mechanical and physical tests, manufacturing controls and chemical element limits based on recognised international standards, and some provisions are enhanced to provide additional assurance for HR and VHR components in particular to improve weldability, inspectability or address specific degradation mechanisms.

Material Selection

For ASME BPVC III components, materials have been selected from those allowed by ASME BPVC Section III and Section II, which ensures the use of materials with proven mechanical properties. Down-selecting between materials follows the structured design optioneering process C3.2.2-2, Reference [27], which for major decisions, e.g. component pressure boundary, provides an output of a structured decision record. This considers a range of factors such as impact to safety, legislative requirements, As Low as Reasonably Practicable (ALARP), Best Available Technique (BAT), cost, supply chain availability and standardisation; previous OPEX in a similar environment is a key consideration.

SA-508M Grade 3 has been selected for major pressure boundary components based on its OPEX, and acceptability under the ASME Code. Surfaces of ferritic components that are exposed to primary coolant are protected by corrosion resistant weld-deposited cladding.

Ageing management is considered in the selection and specification of material, based on the RR SMR Ageing Management Plan in Reference [7] and RGP, implemented through requirements. For example, Alloy 600 and its associated weld metals are not used in contact with primary coolant, and low carbon grade austenitic stainless steels are used to reduce susceptibility to sensitization.

Material Specification

Material specifications are based on international standards, which provide appropriate mechanical and physical tests, manufacturing controls and chemical element limits. For HR and VHR components, additional requirements are applied beyond the ASME BPVC II specifications. For lower safety class components, the design code material specifications are generally employed, with only minimal changes introduced based on RGP. Materials are ordered to ensure that sufficient material is available for testing, surveillance and archiving material, as appropriate.

For HR and VHR 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. In particular, controls are introduced to reduce irradiation embrittlement in the RPV beltline region. Homogeneity of large forgings is demonstrated through chemical analysis and mechanical testing in different locations, which provides assurance for the absence of carbon macro segregation.

Material certification requirements are defined by ASME BPVC III where applicable, or equivalent requirements as provided by the ASME BPVC III Adaption, Reference [4]). Traceability requirements are graded based on classification.

The ASME III nil-ductility reference temperature (RT_{NDT}) fracture toughness testing is used to index the ASME III K_{IC} curve to define the fracture toughness transition region. This is a well-established method of defining the fracture toughness, derived from a large database of testing as a below all points curve. ASME BPVC III provides a method to be used as the basis for defining allowable operating temperatures based on the fracture testing.

23.2.6 Fabrication and Installation

Sub-Claim 1.5: Components are fabricated and installed to appropriate standards using qualified and controlled processes

Argument

For ASME III components, the requirements specified in Sections III and IX set baseline requirements to ensure robust fabrication methods are employed. Only qualified and proven techniques are adopted, using approved and controlled procedures in accordance with ASME NQA-1 requirements.

Evidence

For ASME BPVC components, welding will be carried out in accordance with a written procedure which has been approved and qualified in accordance with ASME BPVC IX by satisfactory completion of a welding procedure qualification test. The welding will be performed by a welder/operator who has demonstrated competence by satisfactory completion of an appropriate performance qualification test also in accordance with ASME BPVC IX.

Design for Manufacture is employed to ensure designs are optimised with manufacturability in mind. Fabrication processes are developed accounting for RGP and refined using techniques such as Process Failure Modes and Effects Analysis (PFMEA).

Robust process controls are imposed, such as special process controls in accordance with ASME NQA-1. Independent inspections against witness and hold points are identified and agreed, with HR and VHR components receiving the highest level of inspection and surveillance. The supplier selection, approval and auditing processes are used to assess evidence of previous experience and capability and ensure requirements can be implemented.

Repairs are only allowed within the ASME BPVC III framework, with more stringent requirements introduced for HR or VHR components. Any deviations from design intent will be recorded and justified, with defects managed to ensure continuous improvement.

23.2.7 Examination

Sub-Claim 1.6: Code-based examinations provide adequate demonstration that manufacturing processes have been performed to an acceptable standard and within expected parameters

Argument

Code-based examinations ensure that the manufacturing processes are behaving as expected and consistent with their qualification. Acceptance criteria are typically based on some characteristic of the defect rather than being linked to a specific acceptable defect size, the acceptance based on experience of what produces components of the required quality (e.g., indication length or ultrasonic response exceeding a generic reference level amplitude). ASME BPVC III sets the minimum requirements for the examination of welds, with additional requirements introduced on a case-by-case basis to address specific concerns. Options to invoke UT in lieu of RT will be explored to reduce dose to workers. Additional UT may be invoked in specific instances to provide further assurance, or diversity or redundancy.

For HR and VHR components, additional objective-based examinations are also carried out as identified under Claim 2.

ASME III Examinations

All manufacturing examination as a minimum will meet the design code requirements, which for ASME III components generally specify both surface and volumetric examinations.

Code-Based Ultrasonic Examination of Welds

The extent to which persons are exposed to ionising radiation needs to be reduced as far as practicable, which is a legislative requirement in the UK, Reference [28]. The volumetric examination of welds required by ASME III has historically mandated Radiographic Testing (RT). However the use of UT in lieu of RT is now permitted through ASME III Code Case N-659-3.

UT is considered to offer a better capability for the planar through-wall flaws (e.g. due to lack of fusion) which pose the greatest threat to component integrity. However RT is effective in detection of volumetric defects such as cavities, solid inclusions, and incomplete penetration. UT will be adopted in lieu of RT where appropriate justification can be made on its capability, and the diversity and redundancy of the overall examination package. The extent of application of UT in lieu of RT will consider the reduced effectiveness of UT for austenitic stainless steel welds due to the coarse grain structure.

Weld-Deposited Cladding

There is no structural reliability Claim on the cladding itself, with its function only to provide a corrosion protection layer to the inside surface of ferritic vessels. However there have been historic issues of under-clad cracking, threatening the integrity of the underlying base material. Accounting for RGP, methods of manufacture will address this potential issue by including sufficient pre-heat and post-heating and improved controls to prevent moisture and hydrogen ingress. Examinations may be carried out to detect either cladding disbond from the parent material or inspect the Heat Affected Zone (HAZ) beneath the clad surface for under-clad cracking.

The requirement for and extent of such inspections will be informed by the manufacturing process development and qualification.

Quality Examinations

The examination schedule is likely to be supplemented with additional inspections at various stages in manufacture at the behest of the manufacturer, which are supplementary to the Code-mandated inspections. These 'Quality' examinations confirm that processes are operating within their expected performance and mitigate the risk of defects being detected at the end of manufacture.

23.2.8 Pressure Testing

Sub-Claim 1.7: Pressure testing provides assurance on the integrity of the pressure boundary

Argument

Compliance with the design code pressure test requirements verifies the integrity and leak-tightness of the pressure boundary under a pressure loading greater than that which will be seen in service. This limiting loading could potentially highlight flaws prior to the vessel entering service and provide other benefits to the component's structural integrity.

Evidence

Hydrotest demonstrates that no gross defects exist at the time the procedure is carried out that threaten the integrity of the vessel. A number of historic pressure vessel failures have occurred under hydrotest, Reference [29].

A study carried out by TAGSI on the value of a hydrotest for an RPV in Reference [30] indicated three primary potential benefits:

1. Provision of some confidence that the construction quality assurance programme has been realised

2. Provision of stress relief in welded regions

3. Discounting the presence of certain defect size and fracture toughness combinations

However, Reference [30] does also acknowledge that the benefit is limited by various factors, such as a lack of knowledge of the defects present at the time of the hydrotest and the material properties at different conditions.

Pressure tests will be specified to be compliant with the design code requirements, and the testing procedure and records will be retained to form the Lifetime Quality Assurance Records.

23.2.9 Quality Assurance

Sub-Claim 1.8: Appropriate controls are implemented in design and specified for manufacture to provide assurance of compliance with the relevant requirements

Argument

The quality assurance activities specified ensure that design, manufacturing processes, commissioning, in-service and decommissioning activities are carried out in accordance with their requirements. A graded approach is employed to ensure appropriate and proportionate activities.

Graded Approach to Quality

A graded approach to quality will ensure a proportionate approach is taken, such that processes of greatest risk will receive the highest levels of assurance, largely based on the consequences of the activity being incorrectly performed. This is consistent with Requirement 7 of IAEA GSR Part 2, Reference [31], and the guidance provided in IAEA GS-G-3.5, Reference [32].

Independent Inspection and Surveillance

Inspection and surveillance will be graded according to the safety consequences and classification.

For ASME III components of Safety Class 1 and below, the Authorised Nuclear Inspector / Authorised Inspector will perform surveillance as required by the Code, except where alternative arrangements are invoked by the ASME III Adaptation. A conformity assessment body will be used where required by the PE(S)R.

For HR and VHR components, an Independent Third Party Inspection Agent (ITPIA) is likely to be appointed to act independently on behalf of the Owner. The ITPIA will be accredited by a recognised national accreditation service to the requirements of BS EN ISO 17020, Reference [33], for a Type A body. The ITPIA will provide assurance that the ASME III requirements are satisfied and provide inspection of the additional higher reliability requirements.

Management of Suppliers

For suppliers of ASME III components, requirements are set in technical specifications to ensure compliance with ASME III NCA and NQA-1, and the RR SMR ASME III adaptations where relevant. Additional RR SMR requirements are specified in the RR SMR Supplier Management System Requirements, Reference [34], which are graded based on the scope of supply and the quality grading of the supplied items. Risk assessments, audits and gated reviews are used to manage suppliers.

The primary responsibility for the overall safety of the design of the RR SMR and the adequacy of the supporting E3S analyses rests with Rolls-Royce SMR as the Principal Designer and supplier of the plant to customers. Where design activities are undertaken in the supply chain, Rolls-Royce SMR carry out the role of intelligent customer.

Design

The RR IMS engineering processes govern internal design activities, ensuring appropriate governance and independent verification. Analysis which is critical to the substantiation of VHR components, e.g. the defect tolerance assessment of the RPV, will be subject to full independent verification.

Materials & Manufacture

Manufacturing quality plans will be used to identify the sequence of activities, specific quality practices and the responsible persons for undertaking, reviewing and approving activities at all stages in product construction. This will identify the governance stages and any witness or hold-points which require external acceptance, which will be agreed with the ITPIA where appropriate.

Processes which are critical to the integrity of the finished product will be strictly controlled to ensure that the required properties are consistently achieved.

Material testing will be carried out at test houses with an appropriate QA system which is accredited to ISO/IEC 17025, Reference [35], by a recognised national accreditation body.

23.3 Demonstration of Reliability

23.3.1 Achievement of Reliability Claim

Sub-Claim 2

Reliability is demonstrated for HR and VHR components through a robust avoidance of fracture case.

Argument

Specifically for HR and VHR components, a demonstration of fracture avoidance is required to ensure appropriate conceptual defence in depth. This requires additional material fracture toughness testing and assurance of examination capability, linked by conservative fracture analysis to provide a balanced demonstration of fracture avoidance. An illustration of this inter-relationship is given in Figure 23.3-1.

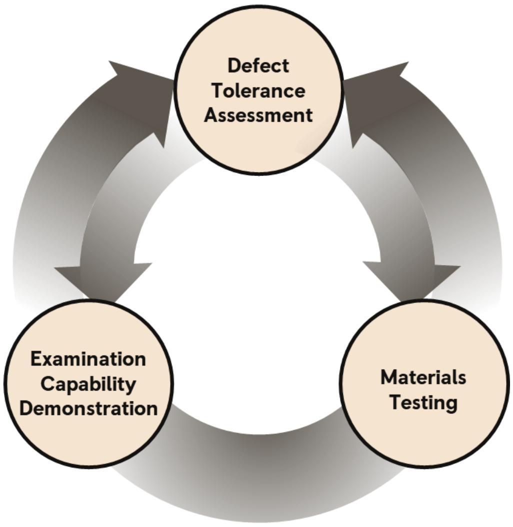

Figure 23.3-1: Avoidance of Fracture Case

Objective-based examinations with relevant assurance of their capability provide confidence in defects of a specified size not being present. Fracture analysis is used in a Defect Tolerance Assessment (DTA) to determine the Limiting Defect Size (LDS) which could threaten the integrity of the component and demonstrate an appropriate margin between this and the specified start of life defect, accounting for through life propagation. Assurance of the LDS is provided by carrying out fracture testing to demonstrate that the material properties used in the DTA are conservative. A balance is sought between these activities so that there is not too great an onus placed on any one individually in the safety case.

Appropriate bounds and expectations for VHR components are informed by UK RGP. A reduced level of demonstration is specified for HR components, to recognise the reduced burden on the avoidance of fracture case, due to the physical defence in depth available.

The Claim is comprised of the following Sub-Claims:

1. Sub-Claim 2.1: Objective-based manufacturing examinations are capable of reliably detecting defects of structural concern

2. Sub-Claim 2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound

3. Sub-Claim 2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern

23.3.2 Examination

Sub-Claim 2.1: Objective-based manufacturing examinations are capable of reliably detecting defects of structural concern

Argument

Examinations are designed to meet the specific objectives based on the credible defects identified for the manufacturing process and the depth as defined by the allowable defect obtained from the DTA. Robust demonstration of inspection assurance is provided by European Network of Inspection Qualification (ENIQ)-based Inspection Qualification (IQ) for VHR welds and the production of capability statements for HR welds and both HR and VHR forging examinations.

Implementation of Objective-Based Examinations

Further to the methods-based examinations described in Sub-Claim 1.6 (Subsection 23.2.7), for HR and VHR components examination requirements are set to detect specific structurally significant defects, with a depth of the Inspection Target Defect Size (ITDS). The ITDS is determined by the DTA incorporating the Target Reserve Factor (RF), with target defect characteristics and morphology informed by an Expert Elicitation (EE) process. The EE process defines the credible defects based on an understanding of the manufacturing process, including potential process breakdowns; the method and framework for EEs is defined in Reference [36].

These HR and VHR inspections are developed to achieve specific capability or performance objectives and are referred to as objective-based examinations. Objective-based examinations are introduced for all HR and VHR welds and wrought material.

The objective-based examinations are specified at the end of production, to supplement the Code-based examinations. For welds this is when the component is in its finished state, after final stress relief and factory hydrostatic test.

For forgings, it is expected that minimal changes will be needed to ASME III Code examinations to demonstrate that the objectives can be met.

The approach for NDE of forgings and welds is described in References [9] and [10].

Examination Capability Assurance

The capability of the examination procedures to meet the objectives as specified in the examination specification will be demonstrated using one of two methods:

1. IQ in accordance with ENIQ (Reference [37]) through a combination of practical trials and a theoretical assessment presented in a Technical Justification

2. Inspection Capability Assessment (ICA), which demonstrates capability largely through the use of existing evidence and literature, with a Capability Statement produced

Further details of the approach will be provided in the Examination Capability Assurance Guide (Reference [38]).

The level of assurance is informed by the significance of the targeted defects, and the likelihood of those defects being generated. Generally VHR welds will be subject to IQ. HR welds, VHR forgings and HR forgings will be subject to ICA.

23.3.3 Fracture Toughness

Sub-Claim 2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound

Argument

Sufficient Upper Shelf Fracture Toughness Testing (USFTT) is carried out on forgings and representative welds to provide confidence that the toughness value used in the DTA is a suitably conservative lower bound.

Sufficient testing will be carried out to ensure that the transition region is appropriately characterised, and with the ASME III K_{IC} curve providing a suitable lower bound to the material behaviour.

Upper Shelf Fracture Toughness

USFTT is specified for all VHR ferritic forgings and welds. Testing is specified to limiting or sample components for both austenitic materials (due to their inherently high toughness and ductility) and HR forgings and welds (due to their reduced consequences).

USFTT of materials is specified with an acceptance criterion based on the initiation of 0.2 mm of tearing ($K_{J0.2mm}$). Testing is carried out in accordance with ASTM E1820-20, Reference [39]. This ensures that the material toughness is greater than that assumed in the DTA.

USFTT will be specified on prolongations of each production forging and representative welds and the associated HAZ, such as those used in the welding procedure qualification. Process controls, Code-based acceptance testing and sample USFTT using the production weld consumable are used to ensure that the production weld output is consistent with that of the process qualification.

Transition Region Fracture Toughness

Claim 1 (Section 23.2) covers Code-mandated drop-weight and Charpy tests used to index the transition region. For HR and VHR ferritic materials, supplementary Charpy testing will be specified to better characterise the transition and upper shelf regime.

Embrittlement

Through-life embrittlement is accounted for according to the approach described in Sub-Claim 3.1 (Sub-section 23.4.2).

23.3.4 Fracture Analysis

Sub-Claim 2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern

Argument

RR SMR defines an assessment method based on the R6 procedure, Reference [40], which has substantial provenance and precedent for its application in the UK. The RR SMR DTA method ensures a range of conservatisms in the inputs and limits and sets a further target reserve factor on defect size to demonstrate margin consistent with OPEX and RGP.

DTA Calculation Overview

For HR and VHR components a demonstration of defect tolerance is carried out through fracture analysis in accordance with the R6 procedure, Reference [40]. This demonstrates appropriate margin based on the demonstrated toughness and defect characterisation and examination capability covered under Sub-Claims 2.1 and 2.2 (Sections 23.3.2 and 23.3.3). R6 is the accepted basis for assessment 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.

Calculations are carried out in accordance with the RR SMR DTA Guide, Reference [5], which identifies an appropriate and standardised approach for the application of the R6 method and the selection of inputs. The basis behind the RR SMR DTA Guide is provided in Reference [41].

Analysis will be carried out using dedicated RR SMR software which is being developed in accordance with the RR SMR software development process C3.3.2, Reference [42]. This ensures it is developed against defined requirements with robust validation and verification activities carried out throughout development using unit test cases, system tests and comparison of results against manual calculations. The use of this software will allow for a standardised and robust implementation of the defined approach and guidance. This will allow for assessment of more locations to be carried out efficiently, reducing the reliance on judgement and read across.

The acceptance criterion requires a Target RF between the end of life defect size, calculated from the ITDS with Fatigue Crack Growth (FCG) through life, and the LDS corresponding to the position on the R6 failure assessment curve, calculated using lower bound material properties. The Target RF for VHR components is 2 and for HR components is 1.4.

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.

Key DTA Calculation Conservatisms

Inputs are generally upper bounds for loading and lower bound for material resistance, with the following identified as key conservatisms in the analysis:

1. Lower bound material resistance will be used. In the transition region the ASME III Appendix G K_{IC} curve will be used, which is a lower bound curve and materials testing will demonstrate the RT_{NDT} to be conservative. On the upper shelf a minimum toughness value will be used corresponding to 0.2 mm of tearing ($K_{J0.2}$), with testing as described in Claim 2.2 used to

demonstrate a higher toughness than this. For infrequent events, an increased lower bound toughness corresponding to up to 2 mm of tearing is allowed.

2. Loading will be conservatively characterised, such as through the use of bounding thermal transient definitions for both the LDS and FCG calculations, and the number of events for FCG calculations. Weld residual stress is calculated using conservative approaches from design codes

3. FCG is calculated using the latest reference curves fully accounting for environmental effects. ASME XI FCG relationships typically represent mean data behaviour; in such cases bounding laws will be used based on the application of additional factors to the reference curves to provide additional conservatism

4. The ITDS will be a conservative characterisation of potential defects, postulated to be planar and through-wall

5. The chosen method is generally based on the simpler rules and correlations provided in R6. These approaches are designed to be conservative compared to real behaviour or detailed representative analysis

23.4 Maintenance of Reliability

23.4.1 Maintenance of Reliability Claim

Sub-Claim 3

Reliability can be maintained with the provision of effective in-service systems and procedures.

Argument

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 a pro-active 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 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.

They are expressed by three Sub-Claims:

1. Sub-Claim 3.1: The ageing and degradation of materials is actively managed and monitored where appropriate

2. Sub-Claim 3.2: In-Service Inspection will detect any defects which may propagate to a critical size

3. Sub-Claim 3.3: Plant Monitoring will provide forewarning of failure

23.4.2 Materials Ageing Management

Sub-Claim 3.1: The ageing and degradation of materials is actively managed and monitored where appropriate

Argument

Material ageing degradation mechanisms will be systematically identified based on OPEX and RGP. These mechanisms will be individually justified. An irradiation surveillance programme will be implemented to validate the irradiation embrittlement model and analysis employed.

Ageing Management Plan

The RR SMR Ageing Management Plan (AMP), Reference [7], outlines the proactive approach taken to identification of potential ageing threats. This identifies potential degradation mechanisms and their applicability based on industry best-practice such as information from EPRI and guidance in ASME III Non-mandatory Appendix W. Individual technical justifications will be produced for each specific degradation mechanism to demonstrate RR SMR components are robust to that degradation, identifying any requirements placed on components to ensure this and any evidence gaps requiring testing.

Embrittlement

An irradiation surveillance programme will be implemented to measure changes to fracture toughness and mechanical properties of material in the beltline region of the RPV throughout the plant life. The strategy is defined in Reference [43]. A shift to the fracture toughness transition curve will be applied in the DTA to account for irradiation embrittlement, which will be validated by the surveillance programme.

An appropriate thermal ageing model will also be used to account for a shift due to thermal embrittlement, with the pressuriser expected to be most significantly affected.

23.4.3 In-Service Examination

Sub-Claim 3.2: In-Service Inspection will detect any defects which may propagate to a critical size

Argument

ASME BPVC XI provides rules for ISI of nuclear plant components with extensive successful application for identifying and managing issues, demonstrating it to be suitable and effective. Additional assurance may be provided for higher reliability components demonstrating a very high confidence in detecting the structurally significant defects.

ASME XI PSI / ISI

ASME BPVC XI, Reference [44], Division 1 provides prescriptive rules for the ISI of components in nuclear power plants. The examination extent is defined based on good practice for different components or types of component and their ASME III Code Classification. Components are examined at a set frequency of 10 years, with pipework subjected to representative sampling such that all locations are examined after 40 years of service. Rules are also provided should it be necessary to take corrective action to repair components.

The following Commitment on Future Dutyholder/Licensee is captured:

***Commitment on Future Dutyholder/Licensee C23.1**: In-Service Inspections should be carried out in accordance with ASME BPVC Section XI Division 1*

ASME XI is based on an objective-based approach with a statistically based qualification of the operator, equipment and procedure for both the detection of and the sizing of flaws, using Performance Demonstration described in Appendix VIII. This has been shown to significantly improve the examinations compared to early iterations such as those discussed by Marshall and studied in the PISC I experimental trials, Reference [45].

A Pre-Service Inspection (PSI) will be carried out using equivalent procedures and equipment as the ISI, to allow for a baseline against which any changes during service can be measured.

Risk-Informed ISI (RI-ISI) approaches have been developed by EPRI, and task groups for ASME BPVC XI and ENIQ. These seek to define a more effective ISI regime with no overall increase in risk, by deploying resources in the areas of greatest risk. It is expected that RI-ISI approaches will be adopted for the RR SMR, with further information and substantiation to be provided in future.

Higher Reliability PSI / ISI Assurance

A future decision by the licensee will be required to determine whether alternative assurance, e.g. ENIQ qualification, is required for PSI and ISI. The components are designed to provide adequate provision for in-service UT which should enable either approach.

Access for NDE

All components are specified and designed for examination to ASME BPVC XI. No access issues have been identified in order to carry out this ISI programme. This will be reviewed as the design matures.

Hydrotests

No periodic in-service hydrotests will be specified, consistent with the requirements of ASME XI.

23.4.4 Plant Monitoring

Sub-Claim 3.3: Plant Monitoring will provide forewarning of failure

Argument

Monitoring of plant conditions ensures that the plant is operating within the defined operating parameters. Monitoring methods will enable confirmation of the absence of unacceptable degradation mechanisms and provide sufficient forewarning of potential deviations from design intent or degraded reliability. This will allow corrective action or mitigation to be implemented so that acceptable reliability can be maintained throughout life. Sufficient monitoring should be in place to enable the use of a digital twin approach.

Transient Temperature and Pressure Monitoring

Monitoring of the number of events and the severity of pressure-temperature transients ensures that the plant operation is within the design specification.

For the Reactor Coolant System (RCS), pressure-temperature limits are stipulated to avoid over-pressurisation particularly at low temperatures, where the potential for brittle fracture could exist. Pressure systems are designed with pressure relief devices, with associated controls, complying with ASME BPVC III over-pressure protection requirements.

Reactor Coolant Chemistry Monitoring

Monitoring of plant chemistry will ensure that operation within the limits of the chemistry specification can be achieved and minimise potential abnormal chemistry conditions.

The RR SMR primary and secondary water chemistry will be monitored and maintained within the specification which will ensure any materials degradation is within expected bounds.

Leakage Monitoring

Instrumentation will be provided to detect leakage from the RCS, which can provide forewarning of a more catastrophic failure.

Loose Parts Monitoring

A Loose Parts Monitoring System will monitor the RCS for the presence of loose parts. This system will respond to any small component which becomes detached which could cause damage to safety critical components.

Vibration Monitoring

Vibration monitoring systems will be provided where required to manage the risk of high cycle fatigue.

23.5 Conclusions

23.5.1 Summary

Claims and preliminary arguments are presented to support the overall chapter Claim that 'Structural Integrity of SSCs is justified and the risk of structural failure is minimised to As Low As Reasonably Practicable', which contributes to the overall E3S objective to protect people and the environment from harm, and the demonstration that risks are reduced ALARP.

At PCD design stage, there is limited underpinning evidence presented within this report. The full suite of Evidence to underpin the Claims and Arguments will be developed in line with the CAE Route Map and reported in future revisions of the E3S Case, including definition of structural integrity requirements, approaches and methods for analysis and substantiation such as avoidance for fracture demonstration, stress analysis and NDE, and ultimately CSRs for specific components or groups of components.

23.5.2 ALARP

The design of the RR SMR has been developed in accordance with the systems engineering design process, which includes alignment to RGP and OPEX, design to codes and standards according to the safety classification, and a systematic optioneering process with down-selection of design options based on assessment against relevant safety criteria (as described in PSCR Chapter 3: E3S Objectives & Design Rules, Reference [46]). Particular focus is placed on the structural integrity requirements for HR and VHR components to ensure that overall risks are reduced to ALARP.

23.5.3 Assumptions & Commitments on Future Dutyholder/Licensee

Table 23.5-1: Assumptions & Commitments on Future Dutyholder/Licensee

Assumption/ Commitment	**ID**	**Description**
Commitment	C23.1	In-Service Inspections should be carried out in accordance with ASME BPVC Section XI Division 1

23.6 References

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[7] RR SMR Report, Doc Number TBC, “Ageing Management Plan,” To be Issued.

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23.7 Acronyms and Abbreviations

ALARP	As Low As Reasonably Practicable
AMP	Ageing Management Plan
ASME	American Society of Mechanical Engineers
BAT	Best Available Technique
BPVC	Boiler & Pressure Vessel Code
CAE	Claims, Arguments, Evidence
CSR	Component Substantiation Report
DBE	Design Basis Earthquake
DFMEA	Design Failure Modes and Effects Analysis
DfX	Design for X
DR	Definition Review
DTA	Defect Tolerance Assessment
E3S	Environment, Safety, Security & Safeguards
EE	Expert Elicitation
ENIQ	European Network of Inspection Qualification
EPRI	Electrical Power Research Institute
Fen	Environmental correction factor
FCG	Fatigue Crack Growth
FMEA	Failure Modes and Effects Analysis
HAZ	Heat Affected Zone
HR	High Reliability
ICA	Inspection Capability Assessment
IMS	Integrated Management System
IPT	Integrated Project Team
ISI	In-Service Inspection
ITDS	Inspection Target Defect Size

ITPIA	Independent Third Party Inspection Agent
IQ	Inspection Qualification
K_{IC}	Critical stress intensity factor
$K_{J0.2}$	Fracture Toughness, corresponding to 0.2 mm of tearing
LDS	Limiting Defect Size
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OPEX	Operating Experience
PCD	Preliminary Concept Definition
PCSR	Pre-Construction Safety Report
PE(S)R	Pressure Equipment Safety Regulations
PFMEA	Process Failure Modes and Effects Analysis
PRZ	Pressuriser
PSI	Pre-Service Inspection
QA	Quality Assurance
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RD5	Reference Design 5
RF	Reserve Factor
RI	Risk Informed
RPV	Reactor Pressure Vessel
RR SMR	Rolls-Royce Small Modular Reactor
RT	Radiographic Testing

RT_{NDT}	Reference Nil-Ductility Temperature
SCC	Stress Corrosion Cracking
SG	Steam Generator
SPC	Seismic Performance Categorisation
SSC	Structure, System or Component
TAGSI	UK Technical Advisory Group for Structural Integrity for High Integrity Plant
USFTT	Upper Shelf Fracture Toughness Testing
UT	Ultrasonic Testing
VHR	Very High Reliability