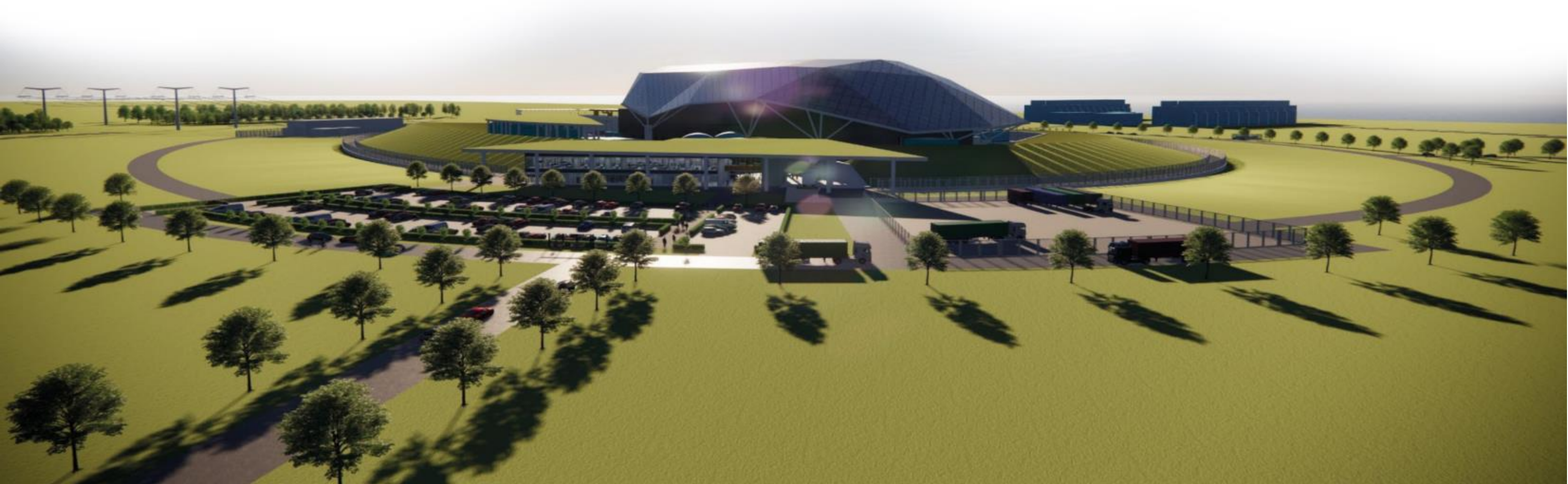
Environment, Safety, Security, and Safeguards Case Version 2, Tier 1, Chapter 23: Structural Integrity



Record of Change

<table>
<tr><th>Date</th><th>Revision Number</th><th>Status</th><th>Reason for Change</th></tr>
<tr><td>March 2023</td><td>1</td><td>Issue</td><td>First issue of E3S Case</td></tr>
<tr><td>February 2024</td><td>2</td><td>Interim Version</td><td>Incorporates revised approaches defined at Reference Design 7, aligned to Design Reference Point 1, including:
Simplified requirements for HR Classification components.
Updates to Tier 2 references.
Structural analysis has been separated from sub-claim 23.1.3 (Design Standards), to be a sub-claim in its own right (sub-claim 23.1.5). Subsequent sub-claims renumbered; other documentation will align to the new sub-claim structure in future updates.</td></tr>
<tr><td>May 2024</td><td>3</td><td>Issue</td><td>Updated to correct revision history status at Issue 2.
Chapter changes include:
The Assumption & Commitment on Future Dutyholder / Licensee / Permit Holder relating to in-service inspection has been removed and is captured instead in the more generic commitment in E3S Case Version 2 Tier 1 Chapter 13 (Conduct of Operations).
Minor wording changes.
Also minor template/editorial updates for overall E3S Case consistency.</td></tr>
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Executive Summary

Chapter 23 of the generic 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 top-level 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).

Version 2 of the generic E3S Case is developed in support of the reference design 7 (RD7) design, corresponding to design reference point 1 (DRP1) for the generic design assessment (GDA). At this 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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23.0 Introduction to Chapter

23.0.1 Introduction

Chapter 23 of the Rolls-Royce Small Modular Reactor (RR SMR) generic Environment, Safety, Security and Safeguards (E3S) Case presents the overarching summary and entry point to the demonstration of structural integrity for the RR SMR.

23.0.2 Scope and Maturity

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 engineering chapter of the E3S Case for the component; an overview of the contents of each chapter is provided in E3S Case Tier 1 Chapter 1: Introduction [1].

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

At Version 2 of the generic E3S Case, the case describes the requirements that are placed on component designs. These requirements are reasonably mature, and components are currently being developed against them. Component substantiation considering these structural integrity requirements are provided in component substantiation reports (CSRs); Appendix A (section 23.6) currently references a single CSR; future updates to the E3S Case for reference design 9 (RD9)/design reference point 3 (DRP3) will incorporate additional components to meet the stated scope.

23.0.3 Claims, Arguments and Evidence Route Map

The overall approach to claims, arguments, evidence (CAE) and the set of fundamental E3S claims to achieve the E3S fundamental objective are described in E3S Case Tier 1 Chapter 1: Introduction [1]. The associated top-level chapter 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 ALARP

A decomposition of this claim into sub-claims, and mapping to the relevant Tier 2 and Tier 3 information containing the detailed arguments and evidence, is presented in the E3S Case Route Map [2]. Given the evolving nature of the E3S Case alongside the maturing design, the underpinning arguments and evidence may still be developed in future design stages; the trajectory of this information, where possible, is also illustrated in the route map.

A proportionate summary of the arguments and evidence from lower tier information, available at the current design stage, is presented within this chapter.

The chapter claim is decomposed into the following sub-claims, addressed in the corresponding sections of this chapter:

- Sub-claim 23.1: Reliability is achieved through high quality design, manufacture, and testing (Section 23.1)
- Sub-claim 23.2: Reliability is demonstrated for components of VHR/HR classification through a robust avoidance of fracture case (Section 23.2)
- Sub-claim 23.3: Reliability can be maintained with the provision of effective in-service systems and procedures. (Section 23.3).

Sub-claims 23.1 and 23.3 apply to all components within scope of this chapter, but with a graded approach applied as described in sub-claim 23.1.1. Sub-claim 23.2 is specific to higher reliability components (VHR and HR classifications – see sub-claim 23.1.1, section 23.1.1); code compliance is considered sufficient for all other classifications for the demonstration of reliability.

The basis of the approach for SI described in this chapter is UK relevant good practice (RGP), and the ONR Safety Assessment Principles [3], in particular EMC.1 to EMC.3 for higher reliability components, and the relevant Technical Assessment Guide [4]. The claim structure employed has been informed by the UK Technical Advisory Group for Structural Integrity for High Integrity Plant (TAGSI) multi-legged approach [5]. The adoption of this RGP is used to define the as low as reasonably practicable (ALARP) position for components and consistent with the claim made in E3S Case Tier 1 Chapter 24: ALARP Summary [6].

Tier 2 CSRs compile the component-specific arguments and evidence for key components or groups of components. These implement this same CAE structure for the SI function, with further decomposition as required. The CSRs are identified in Appendix A (section 23.6).

Other Tier 2 documents provide generic methods and policies which apply to multiple components. Key examples of these documents are:

- Structural Integrity Requirements [7]
- Permitted Adaptations to ASME BPVC III [8]
- Defect Tolerance Assessment Guide [9]
- ASME BPVC III Fatigue Assessment Method [10]
- Ageing Management Plan [11]
- Non-Destructive Examination (NDE) Framework [12, 13, 14, 15, 16, 17, 18, 19].

23.0.4 Applicable Regulations, Codes and Standards

Codes and standards are discussed in sub-section 23.1.2.

23.1 Achievement of Reliability

Sub-Claim 23.1: Reliability is achieved through high quality design, manufacture, and testing.

High quality design, manufacturing and testing is fundamental to achieving the required reliability. The structural integrity measures set out under Sub-claim 23.1 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:

- Sub-Claim 23.1.1: Structural integrity measures are invoked based on the component's role in protecting people and the environment
- Sub-Claim 23.1.2: Components are designed to appropriate standards and requirements which provide suitable reliability
- Sub-Claim 23.1.3: Components are designed for structural integrity, including a review of OPEX and RGP
- Sub-Claim 23.1.4: Structural assessments are carried out to show that the design code limitations are satisfied for the applicable loading conditions
- Sub-Claim 23.1.5: Materials are selected and specified to ensure they are well understood and characterised, based on OPEX and RGP
- Sub-Claim 23.1.6: Components are fabricated and installed to appropriate standards using qualified and controlled processes
- Sub-Claim 23.1.7: Code-based examinations provide adequate demonstration that manufacturing processes have been performed to an acceptable standard and within expected parameters
- Sub-Claim 23.1.8: Pressure testing provides assurance on the integrity of the pressure boundary
- Sub-Claim 23.1.9: Appropriate controls are implemented in design and specified for manufacture to provide assurance of compliance with the relevant requirements.

23.1.1 Component Classifications and Grading

Sub-Claim 23.1.1: Structural integrity measures are invoked based on the component's role in protecting people and the environment.

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. The **dose** to workers and the environment directly as a result of the release of contents from the component (applicable to safety class 3 components only)

3. The **ASME III Code Classification** (for ASME BPVC III components only): 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 Categorisation**: A categorisation based on the importance of the components' role in safety in the event of an earthquake.

The classifications and categorisations for the different components are defined in the relevant CSR, and where appropriate summarised in the relevant Tier 1 engineering chapters.

23.1.1.1 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. This is covered in E3S Case Tier 1 Chapter 3: E3S Objectives and Design Rules for SSCs [20] with the process is described in the Categorisation and Classification Method [21].

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 [22]. This process adopts a failure modes and effects analysis approach 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). This process is used to identify higher reliability sub-components or regions.

The safety classification is used to invoke some SI requirements and informs on selecting the appropriate design standards (e.g. ASME BPVC III or non-nuclear standard), following the principles described in E3S Case Tier 1 Chapter 3 [20].

Higher Reliability Classifications

Two additional higher reliability classifications are established in the Categorisation and Classification Method [21], based on acceptance criteria defined in the RR SMR E3S design principles [23]. These classifications apply when the consequences of failure are not acceptable and additional assurance against component failure is required; the definitions are summarised as:

- **VHR:** 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. It is not reasonably practicable to provide control of the resulting conditions either within or beyond the design basis

- **HR:** Structural failure could lead to relocation of radioactive material, but with off-site doses limited to less than 100 mSv. 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.

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 (sub-claim 23.2.1, section 23.2.1). However the additional defence in depth may be considered in determining specific ALARP positions.

The following components have been identified as containing higher reliability classification (VHR/HR) forgings or welds based on preliminary component assessments following C3.2.2-9 [22]:

- Reactor pressure vessel (RPV) body and closure head
- Pressuriser shell
- Steam generator primary head, tubesheet and secondary shell
- Reactor coolant pump casing
- Reactor coolant loop pipework
- Main steam line pipework
- Main steam isolation valve body.

At this stage these classifications include a number of assumptions and judgements which are subject to further review and refinement. In particular, further work is being carried out on the layout for the main steam line pipework and main steam isolation valve to minimise potential consequences with an aim to reducing the classification.

HR/VHR classifications are not typically to 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 shall 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.

23.1.1.2 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, with the outcome described in section 23.1.2.1. 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” [23], i.e. > 0.01 mSv off-site or > 0.1 mSv on-site.

23.1.1.3 ASME III Code Class

Where ASME BPVC III [24] 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 [25]. A baseline code classification is based on RGP from NRC Regulatory Guide 1.26 [26]. Additional rules are introduced in the process to ensure that:

- The ASME III Code Class is not higher than the safety class
- VHR/HR components are always designed to Code Class 1 rules
- ASME III Code Class does not reduce the classification where it has been elevated due to secondary consequences.

23.1.1.4 Seismic Categorisation

Seismic performance classification is described in E3S Case Tier 1 Chapter 3 [20] with the method defined in SMR0001391 [27].

23.1.2 Design Standards

Sub-Claim 23.1.2: Components are designed to appropriate standards and requirements which provide suitable reliability.

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.

23.1.2.1 Design Standard 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 design principles [23], 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.

The 2021 edition of ASME BPVC [24] 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 BPVC III requirements, which demonstrates that the Code provides a sound basis for safe and reliable operation.

ASME publishes Code Cases that provide additions and revisions to clarify the intent of the Code or provide rules for methods not previously considered. These Code Cases go through extensive review through the ASME Committees and are subjected to additional review by Rolls-Royce SMR to provide confirmation that they are appropriate before being adopted as part of the design basis.

Table 23.2-1: Pressure Retaining Component Design Code Selection

Safety Class	Design Code
VHR/HR	ASME BPVC III Division 1 Subsection NB and beyond code requirements.
1 or 2	ASME BPVC III Division 1, with the subsection based on the ASME III Code Class.
3 Directly leads to release of dose which is: ≥ 0. 1mSv on-site, or ≥ 0.01 mSv off-site.	Either ASME BPVC III Division 1, or Conventional design codes supplemented to give equivalent technical requirements to ASME BPVC III.
3 Does not directly lead to release of dose, such that the following limits are met: < 0.1 mSv on-site, and <0.01 mSv off-site.	Conventional design codes.

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.

ASME BPVC III Adaptation

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, permitted adaptations to the ASME BPVC III requirements are defined in SMR0008428 [8].

The adaptations in SMR0008428 [8] only relate to assurance and organisational requirements, and not technical requirements. They will deliver an equivalent (or enhanced) level of output compared to the ASME BPVC III requirement with no increase in nuclear safety risk or reduction in quality as a result of their use.

23.1.2.2 Requirements Management

RR SMR use of requirements management process C3.1.1 [28] and the relevant software ensures that requirements are invoked on the components based on the classification system. The structural integrity requirements [7] capture the relevant technical requirements aligned to this E3S Case chapter, decomposing them to a greater level of detail. These are then embedded in the technical requirements for each component for both internal design or in the supply chain, and subsequently embedded in the component design specification.

23.1.3 Design Process and Decisions

Sub-Claim 23.1.3: Components are designed for structural integrity, including a review of OPEX and RGP.

The new design of the RR SMR plant allows RGP to be accommodated, while incorporating modern technologies, practices and lessons learned from OPEX in areas such as material forming, welding and examination. Systems engineering approaches ensure design is developed and optimised in a systematic and structured way.

23.1.3.1 Design Process

Systems engineering approaches are adopted, implementing tools such as robust design and a structured design optioneering process (C3.2.2-2 [29]) is adopted for any significant decisions. This approach 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.

23.1.3.2 Design for X

Design for X, i.e. the consideration of various functions and lifecycle stages in the design and optimisation of a product (e.g. manufacturing, assembly, through life maintenance etc), is embedded in working practices through the use of integrated project teams which are made up from a range of functions such as design, structural analysis, procurement, manufacturing, examination and verification. This aspect of Sub-claim 23.1.3 interfaces with various other sub-claims.

Examples of implementation of Design for X are given below.

Examination Access

The RR SMR E3S design principles [23] 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 for the code examination.

Access requirements will consider human factors as described in E3S Case Tier 1 Chapter 18 [30], ensuring compliance with the target audience description [31].

Material Forming

The RR SMR E3S design principles [23] 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:

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

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

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

- The reactor coolant pump casing will be fabricated from a forging rather than a casting.

Material Degradation

Material degradation mechanisms are considered in the design process, with requirements introduced from specific degradation mechanism justifications, the identification of which is discussed in section 23.3.1. For example, crevices are minimised to reduce stress corrosion cracking concerns.

23.1.4 Structural Assessments

Sub-Claim 23.1.4: Structural assessments are carried out to show that the design code limitations are satisfied for the applicable loading conditions.

Design specifications capture the loading which the component shall be demonstrated to withstand based on design, service and test conditions as well as relevant hazards. Assessments are carried out against the design limits in the relevant design code using conservative inputs while accounting for relevant through-life degradation mechanisms.

23.1.4.1 Loading Conditions

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. Loadings are defined by the relevant safety analysis as described in E3S Case Tier 1 Chapter 15: Safety Analysis [32].

For ASME BPVC III components, NCA-2142 and RGP are used to establish the loading specification, along with consideration of storage, transportation, or installation conditions. Design, service and test loadings are defined; each service condition is assigned a service level for use in the ASME BPVC III design analysis:

- Level A – Normal service conditions

- Level B – Upset conditions

- Level C – Emergency conditions

- Level D – Faulted conditions.

Seismic loading includes an assessment of:

- An operating basis earthquake (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

- A design basis earthquake (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.

23.1.4.2 Design Limits

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

- Plastic collapse
- Incremental collapse (thermal stress ratchet)
- Fatigue
- Brittle fracture.

23.1.4.3 Design by Analysis

Guidance on the RR SMR implementation of the ASME BPVC III design by analysis requirements is given in the Appendix XIII Design Based on Stress Analysis Guidance [33].

ASME BPVC III Fatigue Analysis

The RR SMR fatigue assessment method and its technical basis is defined in SMR0005839 [10].

SMR0005839 [10] describes a fatigue analysis method based on the requirements specified by ASME BPVC III with additional requirements to address environmental fatigue. Environmental fatigue is accounted for using environmental correction, Fen, factors from US NRC document NUREG/CR-6909 [34] which are based on an extensive compilation of international test results.

It is considered that the CR-6909 environmental fatigue approach has shortcomings which result in over-conservatism in the overall assessment. This is evidenced by comparison to the extensive OPEX from the worldwide fleet of light water reactors which have been justified without consideration of Fen factors and have not seen failures attributed to environmental fatigue. Therefore, Rolls-Royce SMR is also applying variants on the CR-6909 approach to reflect the latest methods and understanding for austenitic stainless steel. These variants are still pending approval as ASME BPVC Code Cases but have good pedigree and supporting evidence.

Thermal stratification or striping and turbulent temperature oscillations will be avoided by design where possible. Guidelines from the Network for Evaluating Structural Components and the Electrical Power Research Institute (EPRI) are considered in SMR0005839 [10] for the screening and assessment of such phenomena.

23.1.4.4 Prevention of Nonductile Fracture

Code requirements for demonstration of prevention of nonductile fracture are through the application of ASME BPVC 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 will be described in [35].

23.1.5 Material Selection & Specification

Sub-Claim 23.1.5: Materials are selected and specified to ensure they are well understood and characterised, based on OPEX and RGP

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 then 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 VHR/HR components in particular to improve mechanical properties, weldability, inspectability or address specific degradation mechanisms.

23.1.5.1 Material Selection

For ASME BPVC III components, materials have been selected from those allowed by ASME BPVC Section III [24] and Section II [36], which ensures the use of materials with proven mechanical properties. Down-selecting between materials follows the structured design optioneering process C3.2.2-2 [29] 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, 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 [11] 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. In particular, material selection and specification consider the adoption of a boron-free primary chemistry regime as discussed in E3S Case Tier 1 Chapter 20: Chemistry [37].

23.1.5.2 Material Specification

Material specifications are based on international standards, which provide appropriate mechanical and physical tests, manufacturing controls and chemical element limits. For VHR/HR 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 where appropriate based on RGP. Materials are ordered to ensure that sufficient material is available for testing, surveillance, and archiving material.

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. In particular, controls are introduced to reduce irradiation embrittlement in the RPV. 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 [8]. Traceability requirements will be graded based on classification.

Materials testing is used to index the ASME BPVC 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 and pressures based on the K_{IC} curve.

23.1.6 Fabrication and Installation

Sub-Claim 23.1.6: Components are fabricated and installed to appropriate standards using qualified and controlled processes.

For ASME BPVC III components, the requirements specified in Sections III [24] and IX [38] 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 BPVC III NCA [24] and ASME NQA-1 [39] requirements.

23.1.6.1 Process Qualification and Controls

For ASME BPVC III components, welding will be carried out in accordance with a written procedure which has been approved and qualified in accordance with ASME BPVC IX [38] by satisfactory completion of welding procedure qualification. 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 [38].

Design for Manufacture (sub-claim 23.1.3) ensures 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.

Robust process controls are imposed, such as special process controls based on the requirements in ASME NQA-1 [39]. Independent inspections against witness and hold points are identified and agreed, with VHR/HR 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 VHR/HR components. Any deviations from design intent will be recorded and justified, with defects managed to ensure continuous improvement.

23.1.7 Examination

Sub-Claim 23.1.7: Code-based examinations provide adequate demonstration that manufacturing processes have been performed to an acceptable standard and within expected parameters.

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). Further details are provided in the Rolls-Royce SMR NDE Framework [12].

ASME BPVC III sets minimum requirements for the examination of welds, with additional requirements introduced on a case-by-case basis to address specific concerns. The use of ultrasonic testing (UT) in lieu of radiographic testing (RT) is allowed which can enable a reduction in dose to workers. Additional UT may be invoked in specific instances to provide further assurance, or diversity or redundancy.

For VHR/HR components, additional objective-based examinations are also carried out as identified under Sub-claim 23.2.

23.1.7.1 ASME BPVC III Examinations

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

23.1.7.2 Ultrasonic Testing in lieu of Radiographic Testing of Welds

The volumetric examination of welds required by ASME BPVC III has historically mandated RT. However, the use of UT in lieu of RT is now permitted through ASME III BPVC Code Case N-659-3 [40]. Rolls-Royce SMR endorse the use of Code Case N-659-3, subject to further additional requirements and considerations [13]. This is based on an assessment of precedent, the technical capability of UT and the recognition that the extent to which persons are exposed to ionising radiation needs to be reduced as far as practicable [41].

23.1.7.3 Weld-Deposited Cladding

There is no structural reliability claim on 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 or dis-bond of the cladding. Requirements placed on the material and process controls address these potential issues, supplemented by an appropriate extent of UT where the mechanism cannot be discounted as described in SMR0009142 [42].

23.1.7.4 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. Further explanation on the role of these examinations is provided in the NDE Framework [12].

23.1.8 Pressure Testing

Sub-Claim 23.1.8: Pressure testing provides assurance on the integrity of the pressure boundary.

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.

23.1.8.1 Benefits of Pressure Testing

Pressure testing demonstrates that no gross defects exist at the time the procedure is carried out that threaten the integrity of the component. A number of historic pressure vessel failures have occurred under hydrotest [43].

A study carried out by TAGSI on the value of a hydrotest for an RPV [44] indicated three primary potential benefits:

- Provision of some confidence that the construction quality assurance programme has been realised
- Provision of stress relief in welded regions
- Discounting the presence of certain defect size and fracture toughness combinations.

However, TAGSI 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 are specified to be compliant with the design code requirements, and the testing procedure and records will be retained to form part of the Lifetime Quality Assurance Records.

23.1.9 Quality Assurance

Sub-Claim 23.1.9: Appropriate controls are implemented in design and specified for manufacture to provide assurance of compliance with the relevant requirements.

The quality assurance (QA) 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.

23.1.9.1 Graded Approach to Quality

A graded approach to quality ensures 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 [45], and the guidance provided in IAEA GS-G-3.5 [46]. Quality requirements are cascaded to suppliers or products and associated services based on this graded approach. Additional information is provided in E3S Case Tier 1: Chapter 17 [47].

23.1.9.2 Independent Inspection and Surveillance

Inspection and surveillance is graded according to the safety consequences and classification.

For ASME BPVC 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 BPVC III Adaptation [8]. A conformity assessment body will be used where required by the PE(S)R.

For VHR/HR components, an independent third party inspection agent (ITPIA) will 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 [48] for a Type A body. The ITPIA provides assurance that the ASME BPVC III requirements are satisfied and provide inspection of the additional higher reliability requirements.

23.1.9.3 Management of Suppliers

For suppliers of ASME BPVC III components, requirements are set in technical specifications to ensure compliance with ASME BPVC III NCA [24] and NQA-1 [39], and the RR SMR ASME BPVC III adaptations where relevant [8]. Additional RR SMR requirements are specified in the RR SMR Supplier Management System Requirements [49], 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 shall carry out the role of intelligent customer.

23.1.9.4 Design

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

23.1.9.5 Materials and 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 [50], or by a recognised national accreditation body.

23.2 Demonstration of Reliability

Sub-Claim 23.2: Reliability is demonstrated for HR and VHR components through a robust avoidance of fracture case.

Specifically, for VHR/HR 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 inspection capability, linked by conservative fracture analysis to provide a balanced demonstration of fracture avoidance. This balance in crucial in demonstrating an ALARP case consistent with the principles in E3S Case Tier 1 Chapter 24: ALARP Summary [6]. An illustration of this inter-relationship is given in Figure 23.3-1.

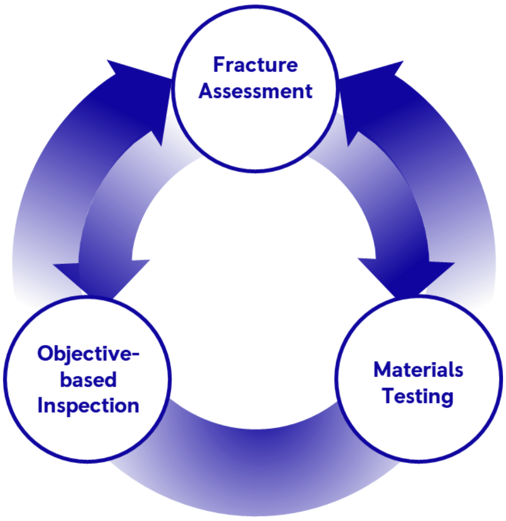

Figure 23.3-1: Avoidance of Fracture Case

Objective-based inspections, 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.

The avoidance of fracture demonstration approach, including bounds and expectations, is informed by UK precedent and RGP as described in the ONR Technical Assessment Guide [4].

Claim 23.2 is comprised of the following sub-claims:

- Sub-Claim 23.2.1: Objective-based manufacturing inspections are capable of reliably detecting, characterising and sentencing defects of structural concern
- Sub-Claim 23.2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound

- Sub-Claim 23.2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern.

23.2.1 Inspection

Sub-Claim 23.2.1: Objective-based manufacturing inspections are capable of reliably detecting, characterising and sentencing defects of structural concern.

Inspections 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 the highest risk VHR/HR locations. A reduced level of assurance will be provided for lower risk VHR/HR locations, for example production of capability statements.

23.2.1.1 Implementation of Objective-Based Inspections

Further to the methods-based inspections described in Sub-claim 23.1.7 (section 23.1.7), for VHR/HR components inspections requirements are set to detect, characterise and sentence specific structurally significant defects with high reliability. The inspection target defect size (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 SMR0008075 [18].

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

The objective-based inspections 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, ASME BPVC III Code examinations are enhanced to ensure that the objectives can be met.

The approach for NDE of VHR/HR forgings is described in the NDE Framework: Annex 2 [14], and for objective-based inspection of VHR/HR welds in NDE Framework: Annex 3 [15].

23.2.1.2 Inspection Capability Assurance

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

- IQ in accordance with ENIQ [51] through a combination of practical trials and a theoretical assessment presented in a Technical Justification, further described in the NDE Framework Annex 7 [19]. This may employ a proportionate approach to qualification where appropriate

- Inspection capability assessment (ICA), which demonstrates capability largely through the use of existing evidence and literature, with a capability statement produced

The level of assurance is informed by consequences of failure (considering a distinction between that for VHR and HR safety classifications), the significance of the targeted defects, and the likelihood of those defects being generated. Further details of the approach will be provided in RR SMR NDE Framework: Annex 5 [17], which will be issued in 2024.

23.2.2 Fracture Toughness

Sub-Claim 23.2.2: The fracture toughness value used in the limiting defect size calculation is a suitably conservative lower bound.

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 is specified to ensure that the transition region is appropriately characterised by a suitable lower bound to the material behaviour.

23.2.2.1 Upper Shelf Fracture Toughness

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

USFTT of materials is specified with an acceptance criterion based on the initiation of 0.2 mm of tearing ($K_{J0.2}$). Testing is carried out in accordance with ASTM E1820-20 [52]. This ensures that the material toughness is greater than that assumed in the DTA.

USFTT is specified on prolongations of each production forging and representative welds and the associated heat affected zone, such as those used in the welding procedure qualification. Process controls, Code-based acceptance testing and sample USFTT using the production weld consumable provide assurance that the production weld output is consistent with that of the process qualification.

23.2.2.2 Transition Region Fracture Toughness

Sub-claim 23.1.5 (section 23.1.5.2) covers Code-mandated tests used to index the transition region. For VHR/HR ferritic materials, supplementary Charpy testing is specified to better characterise the transition and upper shelf regime.

23.2.2.3 Through-Life Degradation of Toughness

The RR SMR Ageing Management Plan [11], which is described under sub-claim 23.3.1 (section 23.3.1), identifies through-life degradation mechanisms which may act to reduce the start of life toughness.

A shift to the fracture toughness transition curve and a reduction in upper shelf toughness is applied in the DTA to account for irradiation embrittlement based on the technical justification [53]. The applied shifts will be validated by an irradiation surveillance programme [54].

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. The approach will be defined in a thermal ageing technical justification for RD9.

23.2.3 Fracture Analysis

Sub-Claim 23.2.3: Fracture mechanics analysis provides a conservative method of determining start of life defects of structural concern.

RR SMR defines an assessment method based on the R6 procedure [55] which has substantial provenance and precedent for its application in the UK. The RR SMR DTA method [9] 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.

23.2.3.1 DTA Calculation Overview

For VHR/HR components, a demonstration of defect tolerance is carried out through fracture analysis in accordance with the R6 procedure [55]. This demonstrates appropriate margin based on the demonstrated toughness and defect characterisation and inspection capability covered under Sub-claims 23.2.1 and 23.2.2 (sections 23.2.1 and 23.2.2). R6 is the accepted basis for assessment of safety critical structures and components in the UK, representing RGP. 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 [9], 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 SMR0002121 [56].

Analysis is carried out using dedicated RR SMR software which is being developed in accordance with the RR SMR software development process C3.3.2 [57]. This ensures the software 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 allows for a standardised and robust implementation of the defined approach and guidance. This also allows 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 of 2 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.

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.

23.2.3.2 Inputs and Methods Used to Ensure a Conservative DTA

To ensure a conservative result is obtained from the DTA, inputs are generally upper bounds for loading and lower bound for material resistance. The key means of achieving this conservatism are highlighted below:

- Lower bound material resistance is used. In the transition region the ASME BPVC III Appendix G K_{IC} curve is a lower bound, and specified materials testing demonstrates the indexing reference temperature to be conservative. On the upper shelf, a minimum toughness value is used corresponding to $K_{J0.2}$, with testing as described in Sub-claim 23.2.2 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

- Loading is 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

- FCG is calculated using the latest reference curves fully accounting for environmental effects. ASME BPVC XI FCG relationships typically represent mean data behaviour; in such cases additional factors are applied to ensure the FCG laws are appropriately bounding

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

- 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

23.3 Maintenance of Reliability

Sub-Claim 23.3: Reliability can be maintained with the provision of effective in-service systems and procedures.

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 surveillance of irradiated RPV materials. In-service inspection (ISI) in accordance with ASME BPVC XI provides a forewarning of failure, through inspections 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 approach can be expressed by three Sub-claims:

- Sub-Claim 23.3.1: The ageing and degradation of materials is actively managed and monitored where appropriate
- Sub-Claim 23.3.2: In-Service Inspection will detect any defects which may propagate to a critical size
- Sub-Claim 23.3.3: Plant Monitoring will provide forewarning of failure.

23.3.1 Materials Ageing Management

Sub-Claim 23.3.1: The ageing and degradation of materials is actively managed and monitored where appropriate.

Material ageing degradation mechanisms are systematically identified based on OPEX and RGP. These mechanisms are being individually justified.

23.3.1.1 Ageing Management Plan

The RR SMR Ageing Management Plan [11] outlines the proactive approach taken to identification of potential ageing threats. This identifies potential degradation mechanisms and their applicability based on sources such as EPRI and guidance in ASME BPVC III Non-mandatory Appendix W. Individual technical justifications are being 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.

23.3.2 In-Service Inspection

Sub-Claim 23.3.2: In-Service Inspection will detect any defects which may propagate to a critical size.

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.

23.3.2.1 ASME BPVC XI PSI / ISI

ASME BPVC Section XI [58] Division 1 provides prescriptive rules for the ISI of components in nuclear power plants. This is discussed further in the RR SMR NDE Framework [12]. The inspection extent is defined based on good practice for different components or types of components and their ASME III Code Classification. Components are examined at a set frequency of 10 years in accordance with the requirements and methods specified in ASME BPVC XI, 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/permit holder is captured:

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

ASME BPVC 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 inspections compared to early iterations such as those discussed by Marshall and studied in the PISC I experimental trials [59].

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.

23.3.2.2 Higher Reliability PSI / ISI Assurance

Further work will be carried out for RD9 to determine where alternative assurance, e.g. ENIQ qualification, is appropriate for PSI and ISI. The components are designed to provide adequate provision for in-service UT which should enable either approach.

23.3.2.3 Access for NDE

All components are specified and designed for inspection to ASME BPVC XI. VHR/HR component designs also consider access requirements to allow for enhanced inspections which could be demonstrated to provide a higher reliability. No access issues have been identified in order to carry out this ISI programme. This will be reviewed as the design matures.

23.3.2.4 Hydrotests

No periodic in-service hydrotests are required by ASME BPVC XI, with no additional testing expected to be specified.

23.3.3 Plant Monitoring

Sub-Claim 23.3.3: Plant Monitoring will provide forewarning of failure.

Monitoring of plant conditions ensures that the plant is operating within the defined operational limits and conditions [60]. 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. The location of monitoring and operations will be defined under E3S Case Tier 1 Chapter 7: Instrumentation & Control [61], informed by E3S Case Tier 1 Chapter 18: Human Factors Engineering [30]. This will allow corrective action or mitigation to be implemented so that acceptable reliability can be maintained throughout life. Sufficient monitoring enables the use of a digital twin approach.

23.3.3.1 Transient Temperature and Pressure Monitoring

Monitoring of the number of events and the severity of pressure-temperature transients provides evidence that the plant operation is within the design specification and allows for appropriate action to be taken if any potential deviations are identified.

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 defined in the relevant engineering chapters, complying with ASME BPVC III over-pressure protection requirements.

23.3.3.2 Reactor Coolant Chemistry Monitoring

Monitoring of plant chemistry described in E3S Case Tier 1 Chapter 20: Chemistry [37] will check that operation is within the limits of the chemistry specification and allow potential abnormal chemistry conditions to be identified and minimised. 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.

23.3.3.3 Leakage Monitoring

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

23.3.3.4 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.

23.3.3.5 Vibration Monitoring

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

23.4 Conclusions

23.4.1 ALARP, BAT, Secure by Design, Safeguards by Design

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 E3S Case Tier 1 Chapter 3: E3S Objectives & Design Rules [20]).

Particular focus is placed on the structural integrity requirements for VHR/HR components to ensure that overall risks are reduced to ALARP considering the principles of E3S Case Tier 1 Chapter 24: ALARP Summary [6], and apply BAT in accordance with E3S Case Tier 1 Chapter 27: Demonstration of Best Available Techniques [62]. This claim is made on the combination of the sub-claims within this chapter and their subsequent requirements, aligning to RGP. Demonstration of a balanced avoidance of fracture case against claim 23.2 for VHR/HR components ensures that any defects generated through construction could either be tolerated or would be detected and further action taken.

23.4.2 Assumptions and Commitments on Future Dutyholder / Licensee / Permit Holder

No specific assumptions or commitments are identified. A commitment for the operational safety programme is captured in C13.1 in E3S Case Chapter 13: Conduct of Operations [63].

23.4.3 Conclusions and Forward Look

The generic E3S Case objective at Version 2 is 'to provide confidence that the RR SMR design will be capable of delivering the E3S fundamental objective as it developed from a concept design into a detailed design' [1]. This confidence is built through development and underpinning of top-level claims across each chapter of the E3S Case, through supporting arguments and evidence. The top-level claim for chapter 23 is 'Structural Integrity of SSCs is justified and the risk of structural failure is minimised to ALARP'.

The arguments and evidence presented to meet the generic E3S Case objective at Version 2 cover the requirements which are placed on component designs to ensure that that overall risks are reduced to ALARP. The arguments and evidence are organised under the following claims:

- Claim 23.1 describes the high quality design, manufacturing and testing which is fundamental to achieving the required reliability. Compliance with the appropriate design code (ASME BPVC III for safety class 1 and 2 components) provides a minimum expectation

- Claim 23.2 describes the additional avoidance of fracture demonstration for VHR/HR components, where the consequences of failure are not acceptable and additional assurance against component failure is required. This demonstration is provided through a combination of objective-based inspections, fracture toughness testing and fracture analysis

- Claim 23.3 describes the maintenance of reliability through pro-active planning in the design phase and through monitoring of information through-life.

Further arguments and evidence to underpin claims will be developed in line with the E3S Case Route Map [2] and reported in future revisions of the generic E3S Case, which will further build confidence that the RR SMR can deliver its fundamental E3S objective. The future developments include work to define the more detailed requirements for components, e.g. the approach for inspection assurance to be defined in SMR0008572 [17]. The majority of the additional evidence that will be generated will be in the applications of the requirements to components, and the -component specific cases to be made in the CSRs.

23.5 References

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23.6 Appendix A: Component-Specific Substantiation

23.6.1 Component Substantiation Report Approach

The case for specific components is made in a CSR where the generic claims in this chapter are applied for the component. The CSR compiles the justification and supporting evidence to substantiate that the component is fit-for-purpose and reduces risk to As Low as Reasonably Practicable. The CSRs also support the engineering chapters of the E3S Case and supporting system design descriptions.

23.6.2 Component Substantiation Report Index

Table 23.6-1 lists the CSRs which fall within the scope of this chapter which have been issued to date. This will be expanded in future updates to the E3S Case.

Table 23.6-1: List of Component Substantiation Reports

Document Number	Scope	Engineering E3S Case Tier 1 Chapter	Reference
SMR0007896	Reactor Pressure Vessel	5	[64]

23.7 Abbreviations

ALARP	As Low As Reasonably Practicable
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
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
ICA	Inspection Capability Assessment
IMS	Integrated Management System
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
NDE	Non-Destructive Examination

NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OPEX	Operating Experience
PE(S)R	Pressure Equipment Safety Regulations
PSI	Pre-Service Inspection
QA	Quality Assurance
RCS	Reactor Coolant System
RD	Reference Design
RF	Reserve Factor
RGP	Relevant Good Practice
RI	Risk Informed
RPV	Reactor Pressure Vessel
RR SMR	Rolls-Royce Small Modular Reactor
RT	Radiographic Testing
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