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18.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the generic Small Modular Reactor (SMR-300) can be constructed, commissioned, operated, and decommissioned on a generic site in the United Kingdom (UK) to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment, as defined in Part A Chapter 1 Introduction [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the Preliminary Safety Report (PSR), which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and evidence to demonstrate that the generic SMR-300 design risks to people are likely to be tolerable and As Low As Reasonably Practicable (ALARP) [1].

Part B Chapter 18 of the PSR presents the Claims, Arguments and Evidence (CAE) for the Structural Integrity topic.

18.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in Part A Chapter 3 Claims, Arguments and Evidence [2].

This chapter (Part B Chapter 18) links to the overarching claim through Claim 2.2:

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

As set out in Part A Chapter 3, Claim 2.2 is further decomposed across several engineering disciplines which are responsible for development of the design of relevant Structures, Systems, and Components (SSCs).

This chapter presents the Structural Integrity aspects for the generic SMR-300 and therefore directly supports a claim focused on the design and assessment of Structural Integrity SSCs, Claim 2.2.9.

Claim 2.2.9: Higher Reliability SSCs have been justified using appropriate methods, demonstrating that risk is tolerable and As Low As Reasonably Practicable (ALARP).

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in sub-chapter 18.3.

The scope of this chapter covers the Candidate Higher Reliability components as set out in sub-chapter 18.2.4.1.

Sub-chapter 18.2.1 covers the identification of Higher Reliability components. Sub-chapter 18.4 covers the codes and standards associated with Structural Integrity. Sub-chapter 18.5 covers an overview of the method for Higher Reliability demonstration. Sub-chapter 18.6 covers a broad look at the achievement of integrity. Sub-chapter 18.7 presents a brief discussion of how Structural Integrity will approach the demonstration of integrity. Sub-chapter 18.8 covers monitoring of integrity.

Finally, sub-chapter 18.9 provides a technical summary of how the claims for this chapter have been achieved, together with a summary of key contributions from this chapter to the overall ALARP argument. Sub-chapter 18.9.3 discusses the GDA commitments that have arisen.

Excluded from the Part B Chapter 18 Structural Integrity scope are all the other mechanical SSCs not identified as requiring the Higher Reliability designation. The Structural Integrity requirements for these SSCs are covered through code compliance and specific additional work identified as needed through the Part B Chapter 19 Mechanical Engineering [3]. The justification of the integrity of the Containment Structure (CS) is excluded from the scope of this chapter and is covered under Part B Chapter 20 Civil Engineering [4].

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Part A Chapter 2 General Design Aspects and Site Characteristics [5].

18.1.2 Assumptions

Assumptions which relate to this topic have been formally captured in the Commitments, Assumptions and Requirements process [6]. Further details of this process are provided in Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [7].

The single identified assumption (A_Stru_041) is: The currently identified Candidate Higher Reliability Components are assumed to be correct at this stage in the project. This will be confirmed through safety analysis topics.

18.1.3 Interfaces with other SSEC Chapters

The Structural Integrity chapter interfaces with the following PSR chapters.

A description and the functional and safety functional claims for the systems of the Structural Integrity SSCs of the generic SMR-300 reactor design is presented in PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [8], PSR Part B Chapter 2 Reactor [9] and PSR Part B Chapter 5 Reactor Supporting Facilities [10].

PSR Part A Chapter 2 General Design Aspects and Site Characteristics [5] outlines the current approach to categorisation and classification and discusses the UK approach as the design develops. It also provides a detailed account of the layout / configuration process and design of the Nuclear Steam Supply System and plant.

PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance (MSQA) [7] describes how safety and quality tasks such as Examination, Inspection, Maintenance and Testing (EIMT) and Verification and Validation (V&V) will be undertaken to the appropriate standard. It also describes how the design controls and processes ensure that the physical configuration of the plant design conforms with layout design requirements (i.e. inspection access etc.).

PSR Part A Chapter 5 Summary of ALARP and SSEC [11] provides the ALARP overview for the PSR and takes the ALARP summary from sub-chapter 18.9 of this chapter.

PSR Part B Chapter 9 Conduct of Operations [12] details the EIMT requirements, which have expectations related to Structural Integrity.

PSR Part B Chapter 12 Nuclear Site Health and Safety and Conventional Fire Safety [13] interfaces with this chapter as Structural Integrity plays a role in mitigating mechanical failures and chemical leaks. Structural Integrity expectations for operations such as inspection impacts Nuclear Site Health and Safety captured in the chapter.

PSR Part B Chapter 14 Design Basis Accident Analysis (Fault Studies) [14] details the Design Basis Accident Analysis (DBAA) which provides input to the Structural Integrity chapter by identifying the conditions that the SSCs would be subjected to during a Design Basis Accident (DBA).

PSR Part B Chapter 15 Beyond DBA (BDBA), Severe Accidents Analysis and Emergency Preparedness [15] details the BDBA and Severe Accidents Analysis to assess extreme scenarios and these are used to assess the effects on Structural Integrity and enhance structural robustness against severe accidents.

PSR Part B Chapter 16 Probabilistic Safety Analysis [16] interfaces with this chapter as it provides input to the assessment of the containment structural analysis (metallic parts) for the Level 2 Probabilistic Safety Analysis (PSA). It also affects the likelihood of various failure modes.

PSR Part B Chapter 17 Human Factors [17] considers human interactions with SSCs during EIMT.

PSR Part B Chapter 19 Mechanical Engineering [3] and this chapter capture all SSCs that have Structural Integrity claims. Any components with Higher Reliability claims are in the scope of the Structural Integrity chapter, any other structural integrity claim that only requires code compliance is covered in the mechanical chapter.

PSR Part B Chapter 22 Internal Hazards [18] provides details on the internal hazards, which is used to maintain the structural integrity of SSCs throughout their lifecycle and also contributes to the mitigation of these hazards.

PSR Part B Chapter 23 Reactor Chemistry [19] interfaces with this chapter as the ageing and degradation mechanisms considered by the Structural Integrity chapter will be cross cutting with Reactor Chemistry.

18.2 SSCS WITHIN THE SCOPE OF STRUCTURAL INTEGRITY

Structural Integrity is a key element in demonstrating the safety of a new nuclear facility. It underpins and interfaces with several topics to provide a full safety case. The Office for Nuclear Regulation (ONR) Safety Assessment Principles (SAPs) [20] place particular scrutiny on the demonstration of safety for the Structural Integrity of metallic components and introduces a classification above that for other topic areas i.e., highest reliability.

While the classification of SSCs is explained in PSR Part A Chapter 2 [5], the need for identification of SSCs with Higher Reliability claims is driven by the process of hazard identification and fault studies analysis. An overview of the process is provided along with a summary of the assumed Candidate SSCs within the scope of Structural Integrity.

18.2.1 Identification of Higher Reliability Components

The identification of Higher Reliability Components is an outcome of an iterative process which will be informed by in-depth assessment of components and their Safety Functional Requirements (SFRs) by Fault Studies, Internal Hazards, and Structural Integrity as presented in the Safety Assessment Handbook [21]. The Fault Studies work is detailed in Part B Chapter 14 [14]. The ONR expectations on Structural Integrity, lessons learnt from the previous GDAs, and the UK Relevant Good Practice (RGP) has informed the development of identifying and assigning Higher Reliability claims on SSCs.

At this stage, engineering judgement has informed the identification of the Higher Reliability candidates based on theorised unmitigated consequences of gross failure. As the Fault Studies work progresses, the postulated gross consequences of failure of SSCs will be further investigated to further develop the identification of Candidate Higher Reliability SSCs. These candidate SSCs, highlighted in Table 1, have a primary safety function for maintaining structural integrity to ensure plant safety. Their direct or indirect consequences of gross failure may result in unacceptable / highly undesirable conditions with release of radioactivity, as no practicable means of providing an engineered physical safety measure is possible. The severity of the consequences and available physical protection will distinguish the class of the component.

The Higher Reliability SSCs are identified as two separate Structural Integrity designations above Safety Class 1 (SC1):

- Very High Reliability (VHR).
- High Reliability (HR).

18.2.2 Very High Reliability Components

VHR is assigned to components for which the consequences of gross failure would be catastrophic and unacceptable. There is no line of protection to mitigate gross failure of these components to prevent release of radioactive material, and it is not reasonably practicable to provide further defence-in-depth. High levels of evidence are required to substantiate the Higher Reliability claims for these components based on sound engineering and beyond design code assessments, thus enabling a claim that the corresponding failure frequency is less than 10^{-7} per annum in line with the Safety Assessment Handbook [21].

18.2.3 High Reliability Components

HR claims are assigned when the gross failure of a component would result in “highly undesirable” consequences. For these components, there is limited protection to prevent unacceptable levels of radiological release but still would result in “highly undesirable” consequences. Appropriate level of demonstration will be provided through a Component Safety Report (CSR) by introducing additional measures beyond normal practice compared to SC1 components. However, compared to VHR, a lower range of evidence provision would be adequate to substantiate the claims for HR, as the consequences of gross failure are less severe. Similarly, the probability of failure of HR components would be expected to be in the region between 10^{-5} and 10^{-7} per annum.

18.2.4 SSC Safety Categorisation and Classification

The safety categorisation and classification methodology currently defined for the SMR-300 by Holtec International utilises the United States Nuclear Regulatory Commission (US NRC) Regulatory Guide 1.26, Revision 6 [22] and related guidance documents.

The design of the generic SMR-300 will require the adoption of appropriate national and international nuclear specific codes and standards to meet regulatory expectations so that it is fit for use as the starting point for a future licensee’s site-specific project. Holtec International acknowledge the existence of differences in the approach to safety categorisation and classification between the NRC Regulatory Guides and other national and international standards.

The differences in the approach to safety categorisation and classification have been identified via the UK GDA Gap Analysis Report [23] and the US / UK Regulatory Framework and Principles Report [24]. Part A Chapter 2 [5] provides the context to the UK Categorisation and Classification approach for safety functions. Work will be conducted moving forward to ensure that the future SSEC presents a viable case for UK deployment, specifically in relation to categorisation and classification.

In line with the SAPs [20], it is recognised that there are SSCs within the generic SMR-300 that will require Higher Reliability claims. These Higher Reliability SSCs are a key part of the safety categorisation and classification.

18.2.4.1 Higher Reliability SSCs

The SSCs in Table 1 have been identified as Candidate Higher Reliability components. This list of SSCs has been modified since Revision 0 of this chapter and the rationale for the removal of specific SSCs has been captured in the table.

Table 1: List of Candidate Higher Reliability Components

SSC Reference	American Society of Mechanical Engineers (ASME) Section III Division I Classification	Higher Reliability Designation	Rationale	Comments
Reactor Pressure Vessel (RPV)	Class 1	VHR	[REDACTED]	N/A
Steam Generator (SGE) with integrated Pressuriser (PZR)	Class 1 – Primary Side and Supports Class 2 – SGE Secondary Side*	VHR	[REDACTED]	N/A
Reactor Coolant Pump (RCP) Bowl	Class 1	VHR	[REDACTED]	[REDACTED]
Hot-Leg pipework	Class 1	VHR	[REDACTED]	N/A
Cold-leg Pipework (either side of RCP)	Class 1	VHR	[REDACTED]	[REDACTED]
RPV Internals	Class CS	None	[REDACTED]	Any requirement for VHR/HR classification will be confirmed through the Safety Assessment process led primarily by [14].
RCP Flywheel and Shaft	Class 1	VHR / HR	[REDACTED]	[REDACTED]
Main Steam System (MSS) Pipes (inside CS)	Class 2*	VHR / HR	[REDACTED]	[REDACTED]
Main Steam Penetration	Class 2*	VHR / HR	[REDACTED]	[REDACTED]
Main Steam Pipes (outside CS, upstream of Main Steam Isolation Valve (MSIV))	Class 2*	VHR / HR	[REDACTED]	[REDACTED]
Passive Core Makeup (PCM) Accumulators	Class 2	None	[REDACTED]	Any requirement for VHR/HR classification will be confirmed through the Safety Assessment process led primarily by [18].
Steel Containment Vessel (SCV)	Class MC	None	[REDACTED]	Any requirement for VHR/HR classification will be confirmed through the Safety Assessment process led primarily by [18].

Note*: Design Challenge Paper [25] and associated Commitment (C_Stru_088) (see sub-chapter 18.9.3) have been raised recognising the gap between the ASME classifications and Higher Reliability Designation, see sub-section 18.9.2.3.

18.2.4.2 Higher Reliability Design Philosophy

Where possible, the application of a Higher Reliability claim should be avoided. It is preferential to the generic SMR-300 that an engineered safety solution is provided, but where this is deemed not practicable, then a Higher Reliability claim will be explored. It is understood that a high burden of proof is attached to the Higher Reliability claim and for the safety case to be acceptable, beyond-code measures must be undertaken. Structural Integrity Principles and Methodology report [26] and the following sections in this chapter provide an overview of the evidence needed to create a robust CSR with Higher Reliability claims.

18.3 STRUCTURAL INTEGRITY CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the Structural Integrity aspects for the generic SMR-300 and therefore directly supports Claim 2.2.9.

Claim 2.2.9: Higher Reliability SSCs have been justified using appropriate methods, demonstrating that risk is tolerable and As Low As Reasonably Practicable (ALARP).

Claim 2.2.9 has been further decomposed within Part B Chapter 18, to provide confidence that the relevant requirements on structures will be met during all lifecycle phases.

Table 2 shows the breakdown of Claim 2.2.9 and identifies in which sub-chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 2: Claims Covered by Part B Chapter 18

Level 4 Claim No.	Claim Wording	Sub-chapter Section
2.2.9.1	Structural Integrity SSCs are designed using appropriate Codes and Standards.	18.4 Codes and Standards
2.2.9.2	Higher Reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing.	18.6 Achievement of integrity
2.2.9.3	Higher Reliability components will be tolerant of defects demonstrated by the avoidance of fracture.	18.7 Demonstration of integrity
2.2.9.4	Through life monitoring, maintenance and inspection will provide forewarning of failures.	18.8 Monitoring of integrity

Appendix A provides a full Claims, Arguments and Evidence mapping for Part B Chapter 18, which includes any lower-level claims, arguments and evidence needed to support the Claims in the table above. This includes identification of evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

18.4 STRUCTURAL INTEGRITY CODES AND STANDARDS

Claim 2.2.9.1: Structural Integrity SSCs are designed using appropriate Codes and Standards.

This sub-chapter outlines the codes and standards used in the design of the generic SMR-300 Structural Integrity SSCs.

Claim 2.2.9.1 has been expanded upon with the following argument:

<u>Argument</u>

The design of Structural Integrity SSCs has been undertaken using best practice nuclear industry codes and standards by use of the ASME Boiler and Pressure Vessel Code (BPVC) design codes.

<u>Evidence</u>

The Codes and Standards for Structural Integrity report [27]. This report identifies the design codes and standards used for the Higher Reliability SSCs.

<u>Overview</u>

The Requesting Party has recognised that UK nuclear safety regulations are based on a non-prescriptive regime and consequently the technical codes and standards that must be used for nuclear power plant are not prescribed. However, codes and standards used should reflect the industry practice and comply with UK regulation. New codes and standards can be introduced where needed and use of such codes will be justified in each case.

The scope of the sub-chapter covers aspects relevant to main metallic components and their reference codes and standards. The SSCs in the scope of Structural Integrity, Mechanical Engineering Part B Chapter 19 [3], and Civil Engineering Part B Chapter 20 [4] will be covered in their respective codes and standards report while referencing the relevant topic area.

18.4.1 Codes and Standards

The design philosophy for SSCs within the scope of Structural Integrity is to design in accordance with ASME BPVC Section III [28]. The principal regulations, codes, and standards are presented in Table 3.

Table 3: Principal Reference Regulations, Guides, Series, Codes and Standards

Label	Title	Revisions
ASME BPVC	Boiler and Pressure Vessel Code [28]	2023 (2021 for Section III Engineering)
ASME NQA-1	Quality Assurance Requirements for Nuclear Facility Applications [29]	2015
U.S.NRC 10 CFR	Title 10, Code of Federal Regulations [30]	Not Applicable
U.S.NRC NUREG-0800	Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition [31]	2016
U.S.NRC RG	Regulatory Guides	Not Applicable

The Codes and Standards for Structural Integrity report [27] presents the codes and standards used for the structural integrity demonstration of the Generic SMR-300 and evaluates their suitability for application in the United Kingdom.

The standards summarised in Table 3 reflect current industry practice and are harmonised with relevant international standards. When compared with UK regulations and RGP, they are judged to be compliant. The report [27] shows that Quality Group A standards (primarily ASME BPVC Section III Subsection NB Class 1 Components) is expected to be used to set the minimum requirements for Higher Reliability SSCs.

The report also maps the standards and regulatory guides applied to key structural-integrity activities: design and manufacture; materials; examination, inspection, EIMT; analysis; quality assurance; and in-service monitoring and surveillance.

Complementing this, SMR-300 Project References for Design and Licencing report [32] lists the complete project-wide set of codes and standards, including the mechanical requirements. It is noted that the 2021 edition of the ASME BPVC Section III has been selected as this conforms with the NRC endorsed edition.

18.4.2 CAE Summary

The design of Structural Integrity SSCs has been undertaken using best practice nuclear industry codes and standards by use of the ASME BPVC design codes. For the current design maturity, Claim 2.2.9.1 and its associated argument demonstrate that the generic SMR-300 SSCs with structural integrity claims are designed to appropriate codes and standards. The codes and standards for structural integrity report [27] identifies the codes used and acknowledges that they are appropriate for use in the UK and are RGP.

18.5 HIGHER RELIABILITY DEMONSTRATION

This sub-chapter introduces the approach to demonstrate how the Higher Reliability SSCs are to be substantiated using appropriate methods.

The established nuclear codes referred to in sub-chapter 18.4 describe the normal practices for components with Structural Integrity claims and are accepted as the minimum specifications for the SSCs with highest reliability claims. However, it is acknowledged that these recognised design code assessments cannot be used alone to discount gross failure of an SSC.

Therefore, a CSR to discount gross failure calls for additional measures beyond normal practices. The CSRs will show conceptual defence-in-depth where it is not reasonably practicable to provide a physical defence-in-depth. The beyond-code demonstration is not required by the US regulatory body. This work will be undertaken moving forward to meet the UK regulatory expectations. The approach for Higher Reliability components is discussed in the SMR-300 Structural Integrity Design Challenge Paper [33]. Through this paper it has been agreed that a UK approach is to be adopted for the UK deployment regarding structural integrity of identified Higher Reliability SSCs.

Avoidance of fracture will be demonstrated through conservative Defect Tolerance Assessment (DTA), qualified manufacturing inspections, and confirmatory fracture toughness testing. This demonstration will be applied to the selected limiting locations of VHR / HR components.

To establish the Higher Reliability CSRs, conceptual defence-in-depth will be presented by the multi-legged approach advised by UK Technical Advisory Group on the Structural Integrity of High Integrity Plant (TAGSI) [34]. This will allow provision of a range of arguments that will substantiate the Higher Reliability claims and enable a failure rate claim of less than 10^{-7} per annum. The three legs of UK TAGSI arguments are:

- Leg A: Achievement of Integrity. Integrity is achieved through quality design and manufacture based on Operational Experience (OPEX) and functional testing (see sub-chapter 18.6).
- Leg B: Demonstration of Integrity. The demonstration that the component is tolerant of failure against a given defect size is achieved through DTA, qualified inspections and additional material testing (see sub-chapter 18.7).
- Leg C: Monitoring of Integrity. The monitoring to provide forewarning of failure is achieved through plant monitoring, In-service Inspections, and testing (see sub-chapter 18.8).

In the UK context, the following ONR SAPs are specifically applicable to Higher Reliability SSCs:

- ONR SAP EMC.1: Safety case and assessment. The safety case should be especially robust and the corresponding assessment suitably demanding, in order that a properly informed engineering judgement can be made that: (a) the metal component or structure is as defect-free as possible; and (b) the metal component or structure is tolerant of defects.

- ONR SAP EMC.2: Use of scientific and technical issues. The safety case and its assessment should include a comprehensive examination of relevant scientific and technical issues, taking account of precedent when available.
- ONR SAP EMC.3: Evidence. Evidence should be provided to demonstrate that the necessary level of integrity has been achieved for the most demanding situations identified in the safety case.

18.6 ACHIEVEMENT OF INTEGRITY

Claim 2.2.9.2: Higher Reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing.

Claim 2.2.9.2 has been broken down into three arguments, each addressing key areas of the claim, that together provide the holistic approach to the achievement of integrity. This sub-chapter will present the generic SMR-300 approach to the achievement of integrity in the form of these arguments and will refer to the lower tier documentation that supports this.

This sub-chapter outlines the engineering methodologies used in the design of the generic SMR-300 Structural Integrity SSCs. The arguments and engineering methodologies are broken down as follows:

- Argument 2.2.9.2-A1
  - Materials (sub-chapter 18.6.1).
- Argument 2.2.9.2-A2
  - Design (sub-chapter 18.6.2).
  - Operational Limits and Design Parameters (sub-chapter 18.6.3).
  - Welds (sub-chapter 18.6.4).
  - Code Analysis (sub-chapter 18.6.5).
  - Manufacturing and Installation (sub-chapter 18.6.6).
- Argument 2.2.9.2-A3
  - Pre-Service Examination, Inspection, Maintenance and Testing (sub-chapter 18.6.7).
  - Quality Assurance (sub-chapter 18.6.8).

The Principles and Methodology report [26] presents the methodology for creating a CSR for Higher Reliability SSCs. This includes details of what is required to support the achievement of integrity leg of the safety case.

18.6.1 Materials

Argument 2.2.9.2–A1: Appropriate materials will be selected with consideration given to compatibility and representative testing.

The material selection for the generic SMR-300 will be conducted through a systematic approach by considering the behaviour of equipment in the manufacturing, operation, inspection, and maintenance stages, as well as previous OPEX in similar environments. The evaluation of material suitability will involve characterising the applicable environment, identifying potential degradation modes, and assessing potential hazards that could impact the continued effectiveness of the selected material. For instance, a material selection process has been followed for the material of the RPV, considering water chemistry, irradiation embrittlement and operational modes with temperature and pressure transients. Scoping calculations using both RG 1.99 [35] and the more conservative American Society for Testing and Materials (ASTM) E900-15 [36] have been undertaken to investigate the effects of irradiation embrittlement on [REDACTED] and show tolerable margins at this stage.

The material selection process used for the generic SMR-300 has been driven primarily by the requirements of the Electrical Power Research Institute (EPRI) Utility Requirements

Document (URD) [37] for the primary circuit and the Top Level Design Requirements [38], specifically Requirement 1173 (3.5.15): 'Proven materials shall be selected by design with well-established operating histories' and Requirement 1174 (3.5.15): 'Materials shall be selected from those Code or industry accepted consensus standards to the extent practical'. The material selection process is also covered in more detail in PSR Part B Chapter 23 Reactor Chemistry [19], which discusses the Nuclear Energy Institute (NEI) requirements on material selection and management.

A Commitment (C_MSQA_109) to create an Ageing and Degradation strategy has been raised in Part A Chapter 4 [7]. This will contribute to the material selection for the Higher Reliability SSCs going forward.

Several documents contribute to the explanation of the material selection at this stage of the project for SSCs in the scope of structural integrity:

1. Overview of Holtec SMR-300 Structural Integrity Main Metallic Structures and Components [39] lists the material selection of individual SSCs.
2. Holtec SMR-300 GDA Structural Integrity Demonstration of ALARP [40] discusses the material selection process and identifies where further work in the scope of Structural Integrity is required to reduce risks to ALARP.
3. The Design Basis Report [41] describes the work conducted to date and highlights where UK specific material work is required.
4. The Structural Integrity Principles and Methodology report [26] outlines how the safety case will be evidenced to demonstrate that the material is acceptable for an SSC with Higher Reliability claims.

18.6.2 Design

Argument 2.2.9.2–A2: Manufacturing and design will be aligned with relevant codes and will build on lessons learnt from OPEX and RGP.

Holtec International have an extensive in-house capability to manufacture, load test, conduct Non-Destructive Examination (NDE), perform analyses and many other areas key to designing a Nuclear Power Plant (NPP). Having in-house capability and established processes results in a well-rounded approach to design. These processes, as well as existing procedures, drive informed design decisions when considering inspection accessibility, component material selection (see sub-chapter 18.6.1 above) or weld minimisation and arrangement. Holtec International have engaged with several companies who have extensive industry experience when designing SSCs. This draws extra OPEX from experienced designers and manufacturers ensuring that a wide breadth of OPEX informs design requirements and decisions.

The Top Level Design Requirements document [38] outlines the three design philosophies for SMR-300: safety, performance, and constructability. This document along with the plant objectives and EPRI URD [37] guidance provides the high level design requirements including design aspects such as maintenance, access, and replacement. PSR Part A Chapter 4 [7] provides details of the design management arrangements in place for delivery of a safe and secure UK SMR-300, which includes requirements management and layout considerations.

During the design of the generic SMR-300, Holtec International's organisational experience has been applied to the SSC design process considering their future pre-service and In-service Inspections (ISI), including access arrangements and geometry allowing for effective

NDE. As the generic SMR-300 progresses through the defined and stage gated design process, continued regard will be given to the positioning, size, number, and environment of welds.

Initially the space for inspection is considered for ASME BPVC expected inspections, but as UK requirements are further developed these will be fed back into the design process loop, and where design change / deviation is needed then the Design Adaptation Committee (DAC) via the Design Management process [42] will be used to enact these. Considering ISI, the SSCs are designed for inspectability of welds and where possible the minimisation of welds is balanced appropriately with the challenges of manufacturing large forgings. This minimisation of vessel welds and reduced number of components results in a simplified design. The Structural Integrity Demonstration of ALARP [40], shows where design decisions have been made balancing the pros and cons of decisions such as forging size versus minimisation of welds.

OPEX of known technical issues, degradation and established design codes are all utilised to inform design decisions. The SSCs have been designed to produce failure modes that are gradual and predictable, while avoiding brittle behaviour, as per Generic Design Criteria (GDC) 31 [30]. For example, the RPV beltline section does not have welds in the high neutron fluence regions to avoid irradiation embrittlement of the welds. The pressure retaining components and any other support structures, in the scope of Structural Integrity, are designed in line with the corresponding ASME BPVC, Section III class requirements [28] following the guidance given in US NRC NUREG-0800 SRP 3.2.2 [31].

The design of components has been investigated for the different service levels with a suitably conservative combination of loads. Loads are combined in line with the NUREG-0800 SRP 3.9.3 [31] and ASME BPVC to calculate the stress limits. This combination will include effects of internal pressure, dead weight, thermal expansion, and dynamic loads due to seismic motion. Conservative assumptions are made for both load values and frequency of events combined.

While the design of SSCs has been conducted to the US regulatory guidance, there is no formal process that documents why particular design decisions are ALARP. The process of providing evidence for the ALARP argument will continue as the design develops. To date, structural integrity SSCs ALARP arguments can be found in Structural Integrity Demonstration of ALARP [40].

18.6.3 Operational Limits and Design Parameters

Components and structures will be designed and analysed within specific defined limits. Therefore, they should be operated and controlled within these limits and conditions. Throughout the plant’s service life, operating conditions will be monitored to ensure they are maintained within these limits. The Overview of Holtec SMR-300 Structural Integrity Main Metallic Structures and Components Report [39] specifies the normal operating conditions for the SSCs within the scope of the Structural Integrity chapter.

Postulated events that the plant might credibly experience during design life and subsequent maintenance activities are considered, to establish the design condition for SSCs with Structural Integrity claims. The design transients, along with the consequent loads and load combinations, will be evaluated in accordance with the appropriate ASME code service limits. This ensures that the SSCs are designed to remain within acceptable limits to accommodate the pressures and temperatures anticipated during normal operation and accident loading conditions such as Safe Shutdown Earthquake (SSE).

SSCs will be designed with appropriate safety margins to ensure their structural integrity, considering service lifetime. The overpressure protection features have been designed with enough capacity to prevent the Reactor Coolant Pressure Boundary (RCPB) from exceeding 110% of design pressure during Normal Operations (NOs) and Anticipated Operational Occurrences (AOOs).

18.6.4 Welds

Welds in the generic SMR-300 plant will be categorised based on their safety classification to ensure that all welds shall perform their intended function. The following requirements will be met:

- The weld joint configuration is selected in accordance with the function of the joint.
- The welding procedure specifications comply with ASME BPVC, Section IX [43] for materials used in the system selected from ASME BPVC Section II [44].
- The quality assurance requirements applied to the welding process correspond to the highest safety classification of the parts being joined.
- The non-destructive examination of every code weld is conducted using quality procedures that comply with ASME BPVC, Section V [45].

The welding operations are performed in accordance with the requirements of codes and standards depending on the design and functional requirements of the components. Component specific weld methodology can be found in the Design Basis Report – Structural Integrity Analysis and Design [41]. The welder will be qualified in accordance with ASME BPVC, Section IX, or AWS D1.1 [46]. The controls on all ASME BPVC welds will be based on Section III.

To supplement the above process, where welds are classified as HR or VHR, beyond-code substantiation is required. This will follow the structure in sub-chapter 18.7, further explained in the Structural Integrity Principles and Methodology report [26].

18.6.5 Code Analysis

The central objective of the Structural Integrity Code Analyses is to ensure that the generic SMR-300 plant possesses sufficient structural capability to withstand normal, off-normal loads, and the worst-case loadings under extreme environmental phenomena or accident events. The input parameters to these analyses are suitably conservative, to provide an adequate margin for uncertainty. These inputs include conservative material properties that will be confirmed by representative testing.

SSC classification will be used to identify the appropriate level of analysis. The classification of the SSC will determine analysis to the accepted codes and standards. Where Higher Reliability claims are made, beyond-code methods of analysis will be undertaken, details of these can be found in the Structural Integrity Principles and Methodology report [26] and are further explain in sub-chapter 18.7.

The Reactor Coolant System will be designed in line with 10 CFR 50 Appendix G [47] and therefore ASME BPVC Section III, Non-mandatory Appendix G requirements [28]. ASME BPVC Section III Sub-section NB and Appendix XIII [28] will be used to investigate plastic collapse, local failure, buckling, and cyclic loading of the system.

A Design Challenge Paper [25] was raised to identify the Candidate Higher Reliability components of the SGE secondary shell and the MSS that required ASME Classification changes from the US design. As a prerequisite the Candidate Higher Reliability pressure retaining SSCs should be SMR Class A or ASME Section III Subsection NB Class 1 to allow the appropriate level of analysis and quality demonstration as a precursor for beyond-code assessments. The recommended option to change the SMR-300 generic design, with additional fall-back options, is to be investigated going forward. See sub-chapter 18.9.2.3 for more details.

Holtec International has a comprehensive internal procedure for V&V of computer codes and calculation methods. V&V problems and sensitivity studies will be applied to computer codes.

18.6.6 Manufacturing and Installation

The manufacturing and installation process for ASME BPVC, Section III components will be largely guided by Section III [28] and Section IX [43], as they provide the baseline requirements following the US NRC NUREG-0800 SRP 3.2.2 [31] guidance on codes and standards application on classes. Where no advantage can be taken from novel technological advancements, only proven techniques will be employed, using approved and controlled procedures in line with ASME NQA-1 requirements. Where HR and VHR components are identified, appropriate beyond-code evidence in line with Structural Integrity Principles and Methodology report [26] will be presented. The manufacturing processes employed will be chosen such that defects of concern can be reliably avoided. This, combined with the appropriate inspection, will assure that the SSCs achieve the intended integrity when entering service.

Holtec International's background in manufacturing and installation gives them a wealth of experience with utilising RGP and OPEX. In addition to this, Holtec International have engaged with industry experts when designing SSCs. This aids in adding further OPEX where outside resource can enhance design quality. The manufacturing methodology adopted will facilitate examination during the manufacturing of SSCs. At agreed hold points, independent inspections will be conducted for HR and VHR components.

18.6.7 Pre-service Examination, Inspection, Maintenance and Testing

Argument 2.2.9.2–A3: Quality manufacturing inspection and testing will be conducted.

An examination strategy will be developed to assure adequate manufacturing of components has been achieved. The defect size that can be confidently measured by the selected examination method will be used as an input to the DTA along with a factor, assuring margins between the acceptable defect size and the ability to detect the defect.

Where ASME NDE methods are applied, the acceptance criteria for the NDE of weld materials for the pressure retaining SSCs will be in accordance with the requirements of the ASME BPVC, Section III, Division 1, NB-5000 [28]. Where beyond-code measures are required, qualified inspection will be conducted in accordance with the Structural Integrity Principles and Methodology report [26]. This qualified inspection is discussed in more detail in sub-chapter 18.7.3.

Holtec International have formalised programmes with criteria determining when testing is required to demonstrate that performance of equipment and plant systems is in accordance with the design parameters. Test results will be documented and evaluated to assure that the test requirements have been satisfied. Test programmes will ensure appropriate retention of test data in accordance with the record requirements of Holtec International's Quality Assurance Plan (QAP). Further description of the measures applied for inspection are provided in the CD-03, Nuclear Quality Assurance Manual [48] and subordinate procedures Topical Report [49].

The US deployment of Leak Before Break (LBB) differs from the application in the UK. In the UK LBB will not be the primary means of demonstrating that gross failure of a component can be discounted. However, LBB is recognised as a supporting argument which can be used to aid the demonstration of conceptual defence-in-depth.

Pre-service EIMT will be undertaken on appropriately sourced, representative test pieces to verify as-built material specifications for vessels and piping including, toughness (J-R curves), tensile strength (stress-strain curves), yield and ultimate strength, and welding process / methods used to ensure that the actual plant-specific stress analysis uses appropriate material properties.

18.6.8 Quality Assurance

With reference to Part A Chapter 4 [7], the Quality Assurance (QA) requirements will be applied to the design, procurement, manufacture, operation, and testing to ensure the safety-related work is performed in accordance with approved QA procedures as described in the Topical Report on the QAP for Holtec International's SMR Design and Construction [49]. This will ensure that inputs to Structural Integrity are accurate.

During the design process, measures are in place to manage inputs, outputs, changes, interfaces, and records within Holtec International and its suppliers, subject to the provisions of the QAP [49]. These controls ensure that design inputs are accurately translated into design outputs, with the final design output referencing suitable acceptance criteria that allows for verification through inspection and testing, as necessary.

Necessary measures and governing procedures have been established to control procurement, ensuring compliance with defined requirements. The interface requirements for procured equipment and supports will be provided to the supplier in the purchase order. Holtec International commits to requirements of ASME NQA-1 2015 for the control of purchased material, equipment, and services with an inspection programme to verify their quality, as outlined in QAP [49]. All safety-related materials require a certified material test report, alongside an independent review for conformance to the specification.

Holtec International will establish and implement measures to assess the quality of purchased items and services, whether purchased directly or through contractors, at intervals and to a depth consistent with the item or service importance to safety, complexity, quantity, and the frequency of procurement. Verification actions may include testing, as appropriate, during design activities. Verifications occur at the appropriate phases of the procurement process, and may include, as necessary, verification of activities of suppliers.

As detailed in Part A Chapter 4 [7], for the UK SMR-300 design and GDA quality assurance is set out in a specific Holtec SMR-300 GDA Project Quality Plan (PQP) [50].

18.6.9 CAE Summary

The achievement of integrity has outlined the need for further work beyond that expected from code. The base level achievement of integrity is guided using ASME BPVC Section III and the beyond-code measures are outlined in the Structural Integrity Principles and Methodology report [26]. This beyond-code work will be undertaken as the design develops, further supporting the claim. Claim 2.2.9.2 has therefore been met to the extent consistent with the maturity of this project at this time; further work in the scope of Structural Integrity will be undertaken and reported in the next revision of the safety case.

18.7 DEMONSTRATION OF INTEGRITY

Claim 2.2.9.3: Higher Reliability components will be tolerant of defects demonstrated by the avoidance of fracture.

Claim 2.2.9.3 has been broken down into two arguments, each addressing a key area of claim 2.2.9.3 that together provide the holistic approach to the demonstration of integrity.

This sub-chapter outlines the approach to demonstration of integrity to be used on the generic SMR-300 Structural Integrity SSCs. It considers the avoidance of fracture demonstration through the following arguments and topics:

- Argument 2.2.9.3-A1
    - Defect Tolerance Assessment (sub-chapter 18.7.1).
    - Fracture Toughness Testing (sub-chapter 18.7.2).
- Argument 2.2.9.3-A2
    - Qualified Inspection (sub-chapter 18.7.3).

18.7.1 Defect Tolerance Assessment

Argument 2.2.9.3-A1: Defect tolerance assessment utilising conservative material properties, fracture toughness testing, and through life inspections will be utilised to demonstrate the avoidance of fracture.

The Principles and Methodology report [26] presents the methodology for creating a CSR for Higher Reliability SSCs. This includes details of what is required to support the demonstration of integrity leg.

As a precursor to beyond-code DTA, code checks and analysis shall have been completed, see sub-chapter 18.6.5. The stresses defined for these analyses may be used as inputs to the beyond-code DTA.

For the avoidance of fracture demonstration for VHR / HR SSCs, conservative DTA will be conducted through the R6 approach [51]. The R6 code for DTA has been through Holtec International's QA Validation [52] and will ensure adequate margin is achieved between concerning defects and the defect size that can be detected reliably by the qualified inspections. The R6 program aids the calculations of the defect assessment R6 Procedure in which the limiting condition of a structure is evaluated by reference to two criteria, brittle fracture, and plastic collapse. Structural integrity relative to the limiting condition is evaluated by means of a Failure Assessment Diagram (FAD), see Figure 1 below for an example of a generic FAD.

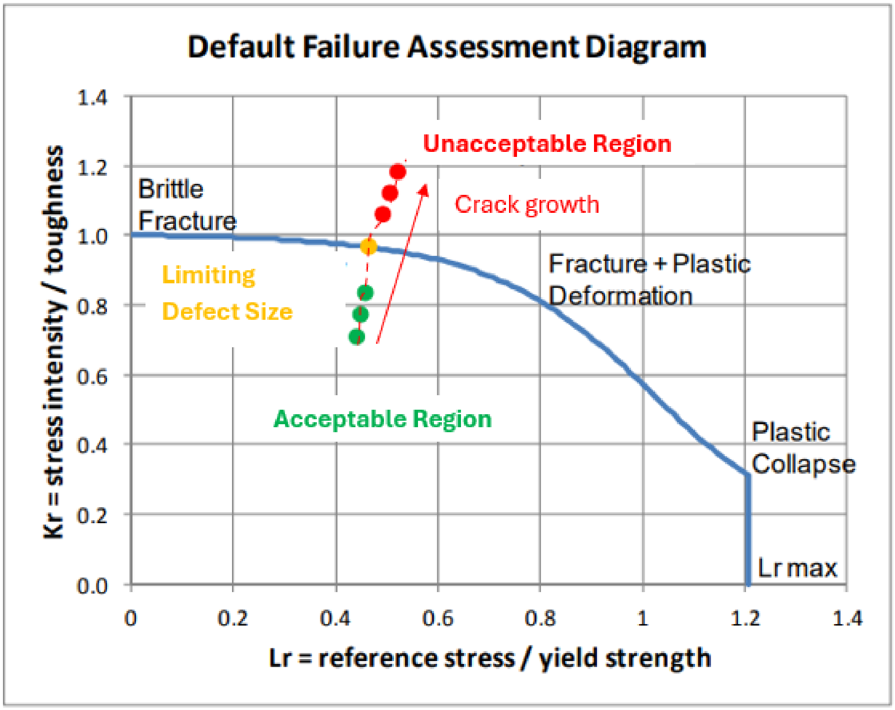

Figure 1: Example Failure Assessment Diagram

As detailed in the Principles and Methodology report [26], the End-of-Life Limiting Defect Size (ELLDS) is compared to the Qualified Examination Defect Size (QEDS) (see sub-chapter 18.7.3) plus the Lifetime Fatigue Crack Growth (LFCG). The acceptance criteria for the VHR components is a Defect Size Margin (DSM) of no less than two (2):

$$DSM = \frac{ELLDS}{QEDS + LFCG} \geq 2$$

The lower bound material properties will be calculated based on ASME BPVC Section XI, Non-mandatory Appendix G – Fracture Toughness Criteria for Protection Against Failure [53] requirements. The values used will be confirmed to be achievable and conservative through fracture toughness testing.

For the RPV beltline region (i.e. the active fuel zone), irradiation embrittlement is a known degradation mechanism that effects the fracture toughness. Appropriate adjustments to calculate the end-of-life fracture toughness's shall be made based on the estimated fluence in the region. Scoping calculations using both RG 1.99 [35] and the more conservative ASTM E900-15 [36] have been undertaken to investigate the effects of irradiation embrittlement on [REDACTED] and show tolerable margins at this stage, see Design Basis Report [41]. The toughness values shall be validated through-life via material surveillance monitoring (see sub-chapter 18.8.3).

18.7.2 Fracture Toughness Testing

Conservative fracture toughness properties will be used for DTA of VHR / HR components. The values used will be confirmed to be lower bounds by representative fracture toughness tests. The methodology of testing weld and parent material properties of VHR / HR components will be reported in the future CSRs. Both drop weight and Charpy V-notch testing shall be performed for each of the ferritic forgings and welds to establish the Reference Temperature for the Nil-Ductility Transition (RT_{NDT}) and standard compact tension (C(T)) specimens shall be tested in accordance with ASTM E1921 [54] to establish the reference temperature, T_0, for critical forgings, see Purchase Specification for Major Forgings [55].

18.7.3 Qualified Inspection

Argument 2.2.9.3-A2: Qualified inspections to detect critical Start-of-Life Defects will be developed.

Inspections are fundamental for establishing that the components are as defect free as possible, especially for welds which have a higher likelihood of defect occurrence. The manufacturing inspections of Higher Reliability SSCs will be qualified with an appropriate arrangement such as European Network for Inspection and Qualification (ENIQ) methodology [56]. This qualification will allow detection of defects of concern by NDE with higher confidence. The inspection scheme will include the qualification of Higher Reliability manufacturing NDE and Pre-Service Inspection (PSI) / ISI of selected Higher Reliability locations. Qualification of the NDE system will require assessment of the following items:

- Qualification of equipment / inspection procedure.
- Qualification of personnel.

Achievement of qualification is through production of a Technical Justification (TJ) and practical trials. It is expected that multiple different NDE techniques will be applied to selected Higher Reliability locations including volumetric and surface breaking techniques to ensure redundancy and diversity.

18.7.4 CAE Summary

The demonstration of integrity claim has recognised the need for further work beyond the code assessment to evidence that it has been successfully achieved. The claim outlines the planned approach to DTA using appropriate methodologies. Claim 2.2.9.3 has therefore been met to the extent consistent with the maturity of this project at this time; further work in the scope of Structural Integrity will be undertaken and reported in the next revision of the safety case.

18.8 MONITORING OF INTEGRITY

Claim 2.2.9.4: Through-life monitoring, maintenance and inspection will provide forewarning of failures.

This sub-chapter outlines the approach to be used in the monitoring and in-service inspection of the generic SMR-300 Structural Integrity SSCs. It considers:

- Plant monitoring.
- In-service Inspection.
- Environmental Material Surveillance.

Claim 2.2.9.4 has been expanded upon with the following argument:

Argument

The design will allow for through-life monitoring, maintenance, and ability to be inspected to the maximum extent practicable.

Evidence

The Principles and Methodology Report [26]. This report provides the methodology to be followed for the Monitoring of Integrity leg of the argument.

The Structural Integrity Demonstration of ALARP [40]. This report provides details of the inspectability aspects of the Higher Reliability SSCs.

The Design Basis Report [41]. This report provides details of the EIMT required for the RPV and SGE SSCs.

18.8.1 Plant Monitoring

Monitoring systems will be in place to provide the operating status of metallic components and structures in the generic SMR-300. Details of the control and instrumentation systems is provided in Part B Chapter 4 [57]. Temperature, pressure, and vibration of the SSCs will be monitored to verify that they are within the bounding conditions identified in the analyses. The primary monitoring systems that will provide assurance on maintaining Structural Integrity and forewarning of failure are:

- The Leak Detection System (LDS), which will be implemented to monitor the Reactor Coolant System for any leak and forewarning of further safety significant failures and consequences, this is detailed in PSR Part B Chapter 1 [8].
- Vibration Monitoring System (VMS), which measures vibration in locations with high cycle fatigue risk, such as SGE tubes and pumps.
- Temperature Monitoring, which will collect reactor coolant temperature data to inform Embrittlement Trend Curves.
- In-core and ex-core instrumentation will measure neutron flux during operation. The output of the instruments will be used to validate the calculated fluence values.
- Vessel surveillance programme, which will monitor the condition of the Higher Reliability pressure vessels.

18.8.2 In-service Inspection

The reactor coolant pressure boundary components are designed with access for periodic inspection and testing of critical areas and features to assess their structural and leak tight integrity. Further details of the Candidate Higher Reliability components inspectability considerations within the design are provided in the ALARP report [40].

As the baseline requirement for class 1 components, the periodic inspection and testing are in accordance with ASME BPVC, Section XI Division 1 [53] pursuant to 10 CFR 50.55a(g). Furthermore, Materials Reliability Program (MRP) requirements will be implemented for managing degradation mechanisms impacting materials in PWR primary systems. A commitment to create an Ageing and Degradation strategy, where these requirements will be captured, is raised in Part A Chapter 4 (C_MSQA_109) [7].

The beyond-code measures for Candidate Higher Reliability SSCs, are the further qualified in-service inspections. Where defects were catalogued during pre-service inspections, this initial fingerprint is compared to the results of the in-service qualified inspection. The results of this inform the necessary action and records are updated and kept for future inspections. Further details of EIMT for the SMR-300 plant are provided in Part B Chapter 9 [12] whilst Part A Chapter 4 [7] provides details on the design controls to ensure that the design conforms with the EIMT requirements.

18.8.3 Environmental Material Surveillance

An optimal Material Surveillance Program, in line with 10 CFR 50 Appendix H [58], will allow monitoring of RPV material of the generic SMR-300, abiding by the rules of ASTM E185-82 [59] and thus changes in fracture toughness properties. More recent revisions of ASTM E185 provide more relevant guidance for modern materials used in longer-lived RPVs. An exemption from Appendix H will be requested to allow implementation of a more appropriate surveillance programme (e.g. ASTM E185-21 and E2215-19). The surveillance programme will include specimens representative of the lower shell forging and the weld(s) that receive the highest fluence. This will provide representative knowledge with regards to the ageing and degradation of RPV material and inform / confirm the analysis.

[REDACTED]

18.8.4 CAE Summary

The monitoring of integrity claim recognises the systems and processes which will contribute towards the forewarning of component failures. The evidence to support this claim also draws upon the relevant good practice defined in the ASME BPVC. Claim 2.2.9.4 has therefore been met to the extent consistent with the maturity of this project at this time; further work in the scope of Structural Integrity will be undertaken and reported in the next revision of the safety case.

18.9 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Structural Integrity chapter and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [11] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This sub-chapter therefore consists of the following elements:

- Technical Summary.
- ALARP Summary:
  - Demonstration of RGP.
  - Evaluation of Risk and Demonstration Against Risk Targets.
  - Options Considered to Reduce Risk.
- GDA Commitments.
- Conclusion.

A review against these elements is presented below under the corresponding headings.

18.9.1 Technical Summary

This PSR Part B Chapter 18 aims to demonstrate the following Level 3 claim and its associated sub-claims to a maturity appropriate for a PSR:

Claim 2.2.9: Higher Reliability SSCs have been justified using appropriate methods, demonstrating that risk is tolerable and As Low As Reasonably Practicable (ALARP).

The design of Structural Integrity SSCs has been undertaken using best practice nuclear industry codes and standards by use of the ASME BPVC Section III design codes (Claim 2.2.9.1).

The achievement of integrity has outlined the need for further work beyond that expected from code. This beyond-code work will be undertaken as the design develops, supporting Claim 2.2.9.2. Material properties testing and transient definitions ensure that data used during assessments will be appropriate and suitably conservative. The claim has been met to an appropriate level for the maturity for this project at this time; further work will be undertaken and reported in the next revision of the safety case.

Development of the demonstration of integrity leg has identified the need for further work to evidence the beyond-code assessment. The Structural Integrity Principles and Methodology report [26] outlines the approach to the DTA and appropriate supporting evidence needed to develop the safety case for components with Higher Reliability claims. Claim 2.2.9.3 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of the safety case.

The monitoring of integrity claim recognises the systems and processes which will contribute towards the forewarning of component failures. The evidence to support this claim also draws upon the relevant good practice defined in the ASME BPVC. Claim 2.2.9.4 has therefore been met to the extent consistent with the maturity of this project at this time; further work will be undertaken and reported in the next revision of the safety case.

18.9.2 ALARP Summary

18.9.2.1 Demonstration of RGP

The design of the generic SMR-300 Structural Integrity SSCs complies with RGP and US NRC requirements applicable in the US. The design adopts nuclear-specific codes and standards endorsed by the US NRC and internationally recognised bodies such as the International Atomic Energy Agency (IAEA). The principal codes and standards identified within sub-chapter 18.4 are considered RGP by the UK nuclear industry. This is based on existing practices adopted on UK nuclear licensed sites and application in earlier and successful GDAs.

The Higher Reliability Structural Integrity approach is not undertaken in the US but is a UK regulatory requirement for identified Higher Reliability SSCs. The adoption of the UK approach regarding structural integrity for the UK deployment has been accepted through a Design Challenge Paper [33]. DTA will be undertaken using the R6 approach [51]. The R-CODE R6 module for DTA has been through QA Validation [52]. The acceptance criteria for the R6 assessment will be adequate margins between the ELLDS and the defect size that can be detected confidently by the qualified inspections with the addition of predicted LFCG. The use of the R6 approach is considered RGP and has been used for the safety cases of both operating UK reactors and as part of recent GDAs.

18.9.2.2 Evaluation of Risk and Demonstration Against Risk Targets

The numerical targets against which the demonstration of ALARP is considered can be found in PSR Part A Chapter 2 [5]. Structural Integrity SSCs, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

- By fulfilling safety functions for normal operations (e.g., shielding and containment), and thereby contributing to achieving Targets 1-3;
- By achieving their target probability of failure they will contribute to the achievement of accident risk, Targets 4-9.

Risks below the Basic Safety Objectives (BSOs) are considered broadly acceptable; however, the Requesting Party is still required to identify further risk reduction measures in line with the ALARP approach. Risks between the BSOs and Basic Safety Levels (BSLs) require a consideration of risk reduction options. Risks above the BSLs are not acceptable for new plants.

Evaluation of risk is not directly applicable to the Structural Integrity SSCs. The Higher Reliability designation of the Structural Integrity SSCs will be associated with a probability of failure, which is then used to calculate the overall comparison against the risk targets as described above.

At this time, the evaluation of the normal operations and accident risks against Targets 1-9 has not been provided. This information will be presented in PSR Part B Chapter 10 Radiological Protection [60] for normal operations, and Part B Chapter 14 Design Basis Accident Analysis [14], Part B Chapter 15 Beyond Design Basis and Severe Accident Analysis and Emergency Preparedness [15] and Part B Chapter 16 Probabilistic Safety Analysis [16] for accident conditions.

18.9.2.3 Options Considered to Reduce Risk

The Structural Integrity Demonstration of ALARP report [40] explores the ALARP arguments for four fundamental areas for the Candidate Higher Reliability SSCs:

- Design: Design decisions made to minimise the safety related impact on the complete system, and to recognise where design decisions improve whole plant safety.
- Pressure Boundary: Reduction in pressure boundary area for a compact design and to limit failure modes.
- Welding: Minimising the number and length of welds to reduce susceptibility of defects in design.
- Inspectability: Considerations made to allow access for inspection in the design.

The ALARP report [40] looks at what design decisions have been made to reduce the risks at this stage of the project and highlights areas where design development is ongoing. The report highlights where risks or differences between US and UK cannot be mitigated, and need to be addressed, and where Design Challenges from the GDA team may be raised following the process for the assessment of risk reduction options presented in Design Management Process [42].

Design Challenge Papers that have been raised at this stage in support of the GDA for the Structural Integrity topic include the following:

- HI-2250234 – SMR-300 Structural Integrity [33]: This Design Challenge Paper focuses on the fundamental concept of Higher Reliability components and how any future deployment of the SMR-300 in the UK should address regulatory expectations for this area. It notes the differences in regulatory regimes between the US and UK for the structural integrity discipline.

  The agreed way forward is that the UK will adopt the approaches and methodologies described in this chapter and supporting documents for the UK deployment. This Design Challenge Paper presents the approach to be followed to conduct the beyond-code assessments for the identified Higher Reliability SSCs beyond the GDA timescales.

- HI-2250236 – Structural Integrity – Candidate Higher Reliability SSC Classification [25]: [REDACTED]

  A Commitment (C_Stru_088) has been raised in sub-chapter 18.9.3 to ensure that the Design Challenge is progressed through the Design Management Process [42] to completion.

18.9.3 GDA Commitments

At Revision 1 there is one GDA commitment identified for Part B Chapter 18 Structural Integrity. GDA commitments have been formally captured in the Commitments, Assumptions and Requirements process [6]. Further details of this process are provided in Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [7]. The Structural Integrity Commitment is the following:

- C_Stru_088 - The Design Challenge Paper 'ASME Classification SGE and MSS' (HI-2250236-R0.0), associated with PSR Part B Chapter 18 Claim 2.2.9.2 'Higher Reliability SSCs Structural Integrity is achieved through quality design, analysis, manufacture, and testing', is with the Design Authority for Design Decision.

  [REDACTED]

  A Commitment is raised to progress this Design Challenge through the Design Management process (HPP-3295-0017-R1.0) to completion. Target for resolution: Issue of Long Lead Item and SSC Procurement Specifications.

In the process of developing the Structural Integrity Chapter, the supporting documents have identified areas of further work to develop the Structural Integrity safety justification. These have either been progressed via Design Challenge Papers, captured as GDA Commitments or are considered as part of normal business within the scope of Structural Integrity topic.

18.9.4 Conclusion

This chapter summarises the Structural Integrity aspects for the generic SMR-300, focused on the design and assessment of the Candidate Higher Reliability SSCs. It identifies the claims and arguments that will form the basis of the safety case for the Structural Integrity topic throughout the lifecycle of the generic SMR-300 to a maturity aligned to a PSR at this stage.

As the design and safety case are developed, further evidence will be provided to substantiate these claims and arguments.

Structural Integrity is a key element in demonstrating the safety of a new nuclear facility, it underpins and interfaces with several topics to provide a full safety case. It is recognised that the identification of Higher Reliability Components is an outcome of an iterative classification process, which is still to be fully completed. It is understood that a high burden of proof is attached to the Higher Reliability claim to discount gross failure of an SSC. Therefore, for the safety case to be acceptable, beyond-code measures must be undertaken to meet the UK regulatory expectations. This will be conducted in accordance with the ALARP principle.

The development of the CSRs will adopt the three-legged TAGSI approach to demonstrate 'conceptual' defence-in-depth. These legs are:

- Leg A – Achievement of Integrity.
- Leg B – Demonstration of Integrity.
- Leg C – Monitoring of Integrity.

The CSRs will demonstrate that the Structural Integrity Higher Reliability SSCs are as defect free as possible and are defect tolerant throughout the life of the plant.

It is therefore demonstrated through this chapter that robust CSRs will be developed to substantiate the Higher Reliability SSCs through this Structural Integrity approach.

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18.11 LIST OF APPENDICES

Appendix A PSR Part B Chapter 18 CAE Route Map

Table 4: PSR Part B Chapter 18 CAE Route Map

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