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UK ABWR Generic Design Assessment

Generic PCSR Chapter 8 : Structural Integrity

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

This chapter describes the safety case that demonstrates sufficient structural integrity of UK ABWR metal Systems, Structures and Components (SSCs). In particular it describes the enhanced design, manufacture, inspection and testing requirements for those SSCs that are safety classified as Very High Integrity (VHI). It lists the high level Safety Functional Claims related specifically to structural integrity that is made on metal SSCs. No Safety Property Claims have been identified as necessary for the demonstration of structural integrity requirements.

The information provided includes: system design requirements for adequate structural integrity; safety categorisation and classification; load combinations considered in normal operation and during faults; materials selection to minimise degradation; safety case assumptions, Limits and Conditions for Operation; resistance to hazards that could potentially challenge structural integrity; and compliance with the ALARP principle.

The overall PCSR justification that the UK ABWR is safe and satisfies the ALARP principle is underpinned by hazards assessments, design basis analysis, probabilistic safety analysis, beyond design basis analysis and human factors analysis (described in PCSR Chapters 6, 7 and 24 to 27), which demonstrate that the designs of SSCs are fault tolerant. These analysis chapters specify the high level safety functional claims related to structural integrity, but do not specify requirements for design parameters on individual SSCs related to structural integrity. Instead they apply analysis conditions and assumptions that are based on, and fully consistent with, the design information and safety claims for structural integrity aspects that are presented in this chapter, in order to substantiate claims on the SSCs.

The structural integrity aspects of the designs of Safety Class 1 and 2 SSCs are well advanced for GDA, being largely based on proven technology from the Japanese ABWR reference design. Additional risk reduction measures have been introduced (with reference to the J-ABWR design) in response to safety assessments undertaken in GDA, and these include uprating of the safety classification of some SSCs to VHI.

This chapter demonstrates that the risks associated with the structural integrity of the SSCs for the UK ABWR are ALARP. It is acknowledged that further work will be required post-GDA to develop the design and fully incorporate site specific aspects. This work will be the responsibility of any future licensee.

8.1 Introduction

The subject of this chapter is the structural integrity of metal Systems, Structures and Components (SSCs) of the United Kingdom (UK) Advanced Boiling Water Reactor (ABWR). This chapter provides an overview of the safety justification for the structural integrity topic area, describing the approach taken to justifying structural integrity within the Generic PCSR. It also describes the overall structure of the structural integrity safety case based on a document map (Appendix C to this chapter) showing where the more detailed evidence to substantiate structural integrity claims is described in a series of supporting Topic Reports (TRs).

8.1.1 Background

The genesis of ABWR design is described in [Ref-8-1] and design summaries of the most safety significant ABWR components are provided as follows:

- Reactor Pressure Vessel [Ref-8-2]
- Main Steam Piping [Ref-8-3]
- Feedwater Piping [Ref-8-4]
- Main Steam Isolation Valve [Ref-8-5]

The UK ABWR Preliminary Safety Report on Structural Integrity [Ref-8-6] describes the approach for demonstration of structural integrity. This is founded on understanding the potential radiological consequences of failure, following which structural integrity classification is assigned. This classification scheme provides a method for establishing the level of rigour applied during the design assessment, material procurement, fabrication, in-manufacture inspection, testing and in-service testing, maintenance and inspection of the component, and the development of the safety case arguments. This classification scheme applies to safety related metal SSCs within the nuclear island and across the Balance of Plant (BOP).

The references in this chapter identify evidence of sound engineering practice to justify structural integrity by sound design, quality of manufacture and effective control throughout operation. For components with a redundancy or diversity of protection, where the radiological consequences of failure are minimal, Hitachi-GE proposes that a structural reliability claim based on the demonstration of good design, and manufacture in accordance with recognised nuclear design standards will be appropriate.

Where limited or no protection can be provided, i.e. where it is not practicable to adequately mitigate the consequences of failure, it is necessary for a higher degree of reliability to be demonstrated. For the UK ABWR, this will require measures to be implemented in excess of design code requirements, in particular to demonstrate that a component is free of significant defects and also tolerant of defects that may be present. For components where it is necessary to claim that the likelihood of gross failure is so low that it can be discounted, the highest levels of structural reliability will be substantiated. Detailed classification is described in section 8.4 which sub-divides Class 1 into three; Very High Integrity (VHI), High Integrity (HI) and Standard Class 1.

8.1.2 Document Structure

This chapter stands as the head document for the Structural Integrity topic area, providing the links to relevant other chapters within the PCSR and its own network of Topic Reports and other supporting reports, as applicable. Thus each chapter consists not only of its own content, but the content of all the linking documents.

Chapter 8 Overview

The remainder of this chapter comprises the following sections:

Section 8.2 Purpose and Scope: Explains the purpose of the chapter and items that are considered within and out of scope of the safety justification for this topic area.

Section 8.3 Safety Functional Claims: Describes the safety functional claims structure, briefly explains the role of the claims in the structural integrity safety justification and provides links to the documentation providing the detailed arguments and evidence.

Section 8.4 Structural Integrity Classification: This section describes the method of assigning required structural integrity levels for SSCs and the criteria used for classification.

Section 8.5 Topic Reports: Provides a description of the main underpinning topic reports that provide the detailed evidence to substantiate the claims made for each of the structural integrity levels.

Section 8.6 Load Condition: Description of the plant operating conditions that defines the load conditions under which structural integrity must be evaluated and ensured to be adequate.

Section 8.7 Assumptions, Limits and Conditions for Operation: The general principles for the identification of Assumptions, Limits and Conditions for Operation (LCOs), are described in Generic PCSR Chapter 4 : Safety Management throughout Plant Lifecycle, section 4.12.

Section 8.8 Summary of ALARP Justification: This section provides a summary of the ALARP justification that appears within the UK ABWR GDA Topic Reports [Ref-8-16], [Ref-8-17], [Ref-8-18].

Section 8.9 Conclusions: This section provides a summary of the main aspects of this chapter.

Section 8.10 References: This section lists documents referenced within this chapter.

Appendices A through C: Provide a comprehensive table of the Safety Functional Claims (SFCs), the equivalent for Safety Property Claims (SPCs), and the document map showing the structure of the Levels 1 and 2 documents for the structural integrity topic area.

Appendices D through J: Provide a summary of the underpinning TRs for this chapter (see below).

Chapter 8 Supporting Documents

The approach to justify structural integrity varies between components according to the consequences of failure and the nature of any available protection. It is determined by the outcome of classification, described in Section 8.4, and the various approaches are summarised in Section 8.5.

The structural integrity chapter does not include a BSC. Instead, a series of TRs underpin this chapter and provide detailed justification of structural integrity for a particular component, or group of components as follows:

TR No.	Component(s)	Level	Reference
1	Reactor Pressure Vessel (RPV)	VHI	[Ref-8-16]
2	Main Steam Piping	VHI	[Ref-8-17]
3	Main Steam Isolation Valve (MSIV)	VHI	[Ref-8-18]
4	Feedwater Piping	Class 1	[Ref-8-20]
5	Reactor Internals	Class 1	[Ref-8-19]
6	Standby Liquid Control (SLC) Storage Tank	Class 2	[Ref-8-21]
7	High Pressure (HP) Turbine Casing	Class 3	[Ref-8-22]

These TRs are summarised as Appendices D through J to this chapter.

Links to Other Chapters

This chapter also provides links to other key chapters within the safety case that form part of or link to the case for this topic area. The most significant links for the structural integrity case are listed below:

Chapter 4 – Assumptions and Limits and Conditions for Operation: Chapter 4 identifies the general principles of Assumptions and Limits and Conditions for Operation.

Chapter 5 – General Design Aspects: Within Chapter 5, Section 5.6 (Categorisation of Safety Functions and Classification of SSCs) sets down the principles for the safety classification of SSCs. The design requirements for (in particular) Class 1 SSCs are an input to the design principles for HI and VHI components defined in this chapter.

Chapter 12 – Reactor Coolant Systems, Reactivity Control Systems and Associated Systems: Chapter 12 provides the system description of the Reactor Pressure Vessel (RPV), the reactor internals, control rod mechanisms, main steam system (including main steam isolating valves), the Feedwater System and core cooling systems. The principles of design set down in this chapter for the design requirements of the HI and VHI SSCs will be applied in Chapter 12.

Chapter 16 – Auxiliary Systems: Chapter 16 provides the system description of the cooling water systems (e.g. ultimate heat sink, reactor building cooling water, etc.). Although there is no HI or VHI SSCs on the systems described, all of the metal SSCs will be designed against the principles set down in this chapter.

Chapter 17 – Steam and Power Conversion Systems: Chapter 17 provides the system description of the: Turbine Generator; Turbine Main Steam, Turbine Auxiliary Steam and Turbine Bypass System; Extraction Steam System; Turbine Gland Steam System; Feedwater Heater Drain and Vent System; Condenser; Circulating Water System; Condensate and Feedwater System; and Condensate Purification System. Although there is no HI or VHI SSCs on the systems described, all of the metal SSCs will be designed against the principles set down in this chapter.

Chapter 23 – Reactor Chemistry: Materials selection and optimisation, which influence the claims that need to be made and evidence to be provided for this chapter.

Chapter 27 – Human Factors: As described in Section 8.1.1, the case for structural integrity is dependent to some degree at all classification levels on “effective control in operation” of the plant. Specifically, due to the reliance on reliable in-manufacture inspection, testing and in-service testing, maintenance and inspection of the components, there are obvious links to Chapter 27 and the human-based safety claims made within it. These claims are not considered explicitly within the claims tree in this chapter as they are general and high-level for GDA. They are however included within Chapter 27.

Chapter 28 – ALARP Evaluation: Chapter 28 discusses the high level approach taken for demonstrating ALARP across all aspects of the design and operation of UK ABWR.

Chapter 31 – Decommissioning: Chapter 31 discusses general requirements for decommissioning of the systems, structures and components within this chapter scope are described in PCSR Chapter 31: Decommissioning. The related claims are summarised in Chapter 31, section 31.5.2.

Environmental and security aspects of the UK ABWR design. For links to GEP, and CSA documentation, please refer to Generic PCSR Chapter 1: Introduction. For GEP, where specific references are required, for example in Radioactive Waste Management, Radiation Protection, Decommissioning, these are included in the specific sections within the Generic PCSR.

8.2 Purpose and Scope

8.2.1 Purpose

The purpose of this chapter is to justify the structural integrity of the SSCs included within the scope, and demonstrate that plant risk due to structural failures is and remains both tolerable and As Low As Reasonably Practicable (ALARP) for GDA.

Specifically, the objectives of this chapter and its supporting documents are to:

- Identify all relevant codes and standards that form the structural integrity requirements;
- Identify the structural integrity safety functions and specify the safety classifications of the SSCs that are within the scope of this chapter;
- Specify the relevant Safety Functional Claims (SFCs) related to the structural integrity topic area;
- Describe where the arguments and evidence that substantiate all relevant safety case claims are presented in the TRs or other supporting documents;
- Identify all links to other chapters of the PCSR to ensure consistency within the structural integrity topic area across the whole safety case; and

The chapter also introduces the structural integrity classifications of High Integrity (HI) and Very High Integrity (VHI) components and identifies the additional requirements for these components over and above those of "Standard" Class 1 SSCs.

8.2.2 Scope

The scope of the structural integrity case comprises all metal SSCs that are important to safety, as identified by classification, with respect to their structural integrity for all conditions that may credibly occur during a 60 year period of operation. The decommissioning period is also considered at a strategic level during GDA (see Chapter 31: Decommissioning), and this revision of this chapter presents the work done in this area.

The approach to substantiate structural integrity is summarised as follows:

- Identify Safety Functional Requirements and categorise these according to their importance to safety.
- Identify SSCs that deliver each safety function.
- Establish suitably rigorous requirements for design, construction, and operation, according to classification.

A detailed scope for each of the SSC groups or plant areas, in terms of the components considered and their physical boundaries, is defined within each of the underpinning TRs [Ref-8-6 through Ref-8-12] and is summarised in each related appendix (see Section 8.1.2). These TRs demonstrate that the pressure boundary of metal SSCs satisfies function to form reactor coolant pressure boundary and the reactor internals satisfy function to maintain core geometry.

It is noted that structural integrity concentrates on the integrity of static components and structures, like the ones described above, which are important to safety. Mechanical systems like lifting equipment, the fuel handling machine, pressure relief systems, pumps, valves and ventilation systems are in the mechanical engineering scope and not considered within the scope of this chapter. Civil structures are also out of the scope of this chapter.

8.3 Safety Functional Claims

The SFCs of any SSC determine how nuclear safety should be maintained under all design basis conditions. The SFCs have been identified for the respective SSCs during the GDA process. The high level safety claims, performance and safety design bases for the reactor coolant system are discussed in Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems, section 12.3.3.1 and 12.3.3.2, and provide the basis for identifying the specific level SFCs for the respective SSCs. These system based SFCs are to satisfy the Reactor Coolant Pressure Boundary (RCPB) function. This function is ensured by providing each one of the SSCs forming it with the reliability commensurate with their Structural Integrity classification. This is achieved by compliance with the requirements of well-established and appropriate design codes. This approach is described in the following sections in this chapter.

Claims and Link to High Level Safety Functions

The list of claims in this chapter and the linkage to corresponding High Level Safety Functions (HLSFs) is shown in Appendix A. A short description on the application of High Level Safety Functions in the development of the claims, arguments and evidence is provided in Chapter 1 of this PCSR. Fundamental Safety Functions (FSFs) and HLSFs are described in [Ref-8-15 Section 3.2.1]. For example, from the view point of structural integrity, the representative FSF of reactor coolant pressure boundary components is “Containment of radioactive materials” and the HLSF is “Function to form reactor coolant pressure boundary”. SFCs to satisfy this HLSF are described in Appendix A. Based on these SFCs, arguments and evidences are demonstrated in the relevant BSC or Topic Report for respective SSCs.

8.4 Structural Integrity Classification

The structural reliability of UK ABWR SSCs will be justified according to the consequences of their failure, as established by a system of component structural integrity classification. Section 5.6 of this PCSR describes the safety classification scheme for UK ABWR SSCs and establishes three classes based on safety significance. These are Class 1, 2 and 3.

The frequency and consequences of failure of Class 1 components vary significantly. As the risk of failure varies so will the required assurance of structural integrity. In order to identify where the very highest standards of structural integrity should apply, [Ref-8-7 Section 2] describes a refined scheme of classification to be adopted which sub-divides Class 1 into three; Very High Integrity (VHI), High Integrity (HI) and Standard Class 1.

Table 8.4-1 shows Structural Integrity Classes which illustrates the criteria in terms of consequences of failure. VHI is assigned to failure modes for which there is no protection, where failure is intolerable and where it is not reasonably practicable to provide protection. HI is assigned where failure can lead to severe core damage but where a single line of protection exists; generally this means that effective containment exists to limit the offsite consequences to a tolerable level.

Process

The process identified in [Ref-8-8] is based on consideration of the components safety function, postulated failure mode, direct consequences of failure, lines of protection and the indirect consequences of failure, such as those arising from the generation of energetic missiles or pipe whip. [Ref-8-7 Section 2] describes an approach for considering each of these, a method for recording the data and a process of expert review.

Status

Safety classification has been undertaken as an iterative process. The RPV (main parts: Top Head, Bottom Head, Shell and Large Diameter Nozzles), parts of the Main Steam Piping and inboard MSIV have been assigned a classification of VHI. Reactor Internals (Core Support Structures), outboard MSIVs, Feedwater Piping and parts of the Main Steam Piping have been assigned a classification of Standard Class 1 as described in section 12.3 (Table 12.3-3). Classification has been completed by consideration of the results of fault analyses and reference to the single failure criterion.

Table 8.4-1: Structural Integrity Classes

Class	Consequences of failure
VHI	Severe core damage and large off-site release of radiation.
HI	Severe core damage. Containment protects against large off-site release. Limited release of radioactive material.
SC 1	Localised damage to fuel. Minor off-site release. Significant release within nuclear island.
SC 2 & 3	No core damage. Fault within capability of protective systems. Contamination within nuclear island.

8.5 Topic Reports

This section summarises the approach to demonstrate suitably robust structural integrity for Class 1, 2, 3, HI, and VHI SSCs of the UK ABWR and is supplemented by the appendices described in Section 8.1.

The standards by which structural integrity is assured reflects the functional reliability requirements of the SSCs, commensurate with their safety classification. The structural integrity of all Class 1, HI and VHI SSCs will, as a minimum, be justified by evidence of compliance with the requirements of well-established and appropriate design codes. Supplementary evidence to support more exacting integrity claims will be provided for HI and VHI components. The structure of the Topic Reports, and the approach for each class, is described below.

8.5.1 Structure of Topic Reports

Seven Topic Reports have been appended to this chapter, as described in Section 8.1. The structure of the topic reports for VHI and HI components is consistent with that recommended by the UK Technical Advisory Group on Structural Integrity (TAGSI). This provides an approach for justification of high structural reliability claims by establishing diverse evidence of conceptual defence in depth against the risk of failure. At the highest level, the safety case for HI and VHI components is structured according to five distinct safety function claims as follows:

Claim 1: Structural Integrity is assured by good design [SI SFC 4-1.1]

Claim 2: Structural Integrity is assured by material selection and quality manufacturing [SI SFC 4-1.2]

Claim 3: Functional testing provides a demonstration of integrity at start of life [SI SFC 4-1.3]

Claim 4: Through life integrity is demonstrated by analysis and inspection [SI SFC 4-1.4]

Claim 5: Inspection and Monitoring regularly validate integrity through life [SI SFC 4-1.5]

Each claim will be supported by a series of arguments, which will each be substantiated by identification of robust and diverse evidence, typically compiled as a dossier of technical information, data and analyses reports.

Structural integrity of Class 1 components is founded on demonstration of compliance with the requirements of an established nuclear design code. Reactor internals and feedwater piping have been assigned as Standard Class 1, and summaries of the topic reports for each have been included in Appendices G and H.

A similar approach has been adopted for Class 2 and 3 components. The SLC Storage Tank has been assigned as Class 2, and a summary of the topic report has been included in Appendix I. The HP Turbine Casing has been assigned as Class 3, and a summary of the topic report has been included in Appendix J.

8.5.2 Class 1 Components

Substantiation of the structural reliability of the Class 1 components is based on demonstrating high quality of design and manufacture by compliance with relevant aspects of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the ASME B&PV Code). The precise edition, addenda and code cases applied is specified in the appendices. Additional, beyond code requirements are applied where appropriate to justify that the risk of compromising the structural integrity of the components is ALARP.

The ASME B&PV Code prescribes rules governing the design, fabrication, and inspection of boilers and pressure vessels that are well established for application in the nuclear industry. The following sections of the ASME B&PV Code establish requirements that address key aspects of ABWR design:

Section II	Materials
Section III	Rules for construction of nuclear components
Section V	Non-destructive examination
Section IX	Welding and brazing qualification
Section XI	Rules for in-service inspection of nuclear components

For Class 1 components, the process of structural integrity classification establishes that protection against failure exists and that the potential consequences of failure are of a limited extent. The safety argument for Class 1 components therefore concentrates on the effective prevention of failure. Four broad claims are used to structure the topic reports for Class 1 components, each founded on compliance with relevant requirements of the ASME B&PV Code which provides suitably robust assurance of structural integrity. These cover quality design and manufacture, design code assessment, hydrostatic testing and in service inspection.

The ASME B&PV Code prescribes diverse measures to control quality of design and manufacture and embodies extensive operating experience that is relevant to the ABWR components. This ensures a structurally robust design and provides effective measures to prevent failure and to minimise, monitor and control degradation by good design. Compliance with the ASME B&PV Code is therefore judged to provide a suitable means for assuring that the structural integrity of the ABWR Class 1 components can be maintained for the design lifetime.

Materials will be specified and examined to effectively resist fracture and degradation. To demonstrate good choice of materials, evidence will be provided regarding their specification and procurement in accordance with the requirements of Section II of the ASME B&PV Code. This is intended to ensure that well proven materials are chosen which are resistant to fracture and of suitable composition to effectively limit the effect of through-life degradation. The material specification requirements include limitations on manufacturing techniques, the use of weld repairs, heat treatment, chemical composition, mechanical testing, inspection, and quality assurance. Evidence will also be provided to demonstrate compliance with the mechanical testing requirements of Section III of the ASME B&PV Code. Consideration of international OPEX and best practice will also be used to inform decisions on material selection and processing.

Section III of the ASME B&PV Code includes a requirement to conduct structural analyses to support the design for a range of conditions. These include pressure, temperature, and mechanical loadings due to normal operating and test conditions, anticipated transients and postulated accident conditions that could occur during operation. The evaluation of the service and testing conditions includes an evaluation of fatigue due to cyclic stresses. The results of these analyses will be identified as evidence to deterministically justify the structural integrity of ABWR components against stress and fatigue limits established in the ASME B&PV Code and thus confirm robust design.

Controls will be applied to ensure compliance with the design specification in manufacture, installation and commissioning. The ASME B&PV Code includes measures to control quality of manufacture and installation. Relevant evidence will include controls to ensure compliance with the welding procedures, testing of weld materials and welder qualification with rules prescribed in Section III and Section IX of the code. End of manufacturing inspections will be performed to confirm the quality of the manufacturing.

Pre-Service Inspection and testing will be specified to confirm quality of design and manufacture and these will be applied to confirm robust structural integrity before the component enters service.

In addition to a programme of inspection the Class 1 components will, where appropriate, be subjected to a hydrostatic overpressurisation test and to a system hydrostatic test before entering service. The purpose of these tests is to confirm that the ability to sustain design pressure has not been compromised during manufacture and installation and that the design adequately prevents leakage. Hydrostatic tests will be specified in accordance with Section III (Subsections NB and NC) of the ASME B&PV Code.

In-Service Inspection (ISI) and monitoring will be specified to effectively reveal degradation in good time. Evidence of ISI that will effectively provide timely forewarning of failure for Class 1 components will be provided by establishing that ISI will be specified in accordance with the requirements of the ASME B&PV Code Section XI. Arrangements for leak monitoring, leak detection and environmental monitoring will be identified, providing diverse means to reveal degradation and prompt corrective action.

8.5.3 HI Components

The failure of HI components can lead to radiological consequences, but the process of structural integrity classification will identify evidence that effective containment exists to limit the offsite consequences. It is necessary that the structural integrity of the HI regions is substantiated to a higher degree of rigour than that required for the Standard Class 1 components. This is provided by evidence to demonstrate that welds will be subject to qualified manufacturing inspections, supported by an elastic-plastic fracture assessment to demonstrate tolerance to defects as described for VHI components in the following section. The principle difference between HI and VHI components is in regard to the diversity of evidence to support the arguments, which for VHI components will be greater than that provided for HI components. In addition to the structural reliability justified by compliance with the ASME B&PV Code, these additional measures are claimed to support a frequency of failure commensurate with HI classification.

8.5.4 VHI Components

VHI Topic Reports are based on five claims to identify diverse evidence to demonstrate the avoidance of significant defects; that any identified defects are tolerable and that defect growth will not compromise safety through the component life-cycle. The numerous arguments and diverse evidence to substantiate these claims will be provided in a separate TR for each component that includes regions classed as VHI. A general summary of the content is provided below.

The first two claims are intended to establish high quality through good design and manufacture, supplemented by the third claim of functional testing to demonstrate fitness for purpose at start of life. This is the foundation for demonstration of very high reliability through the avoidance of significant defects. The basis for these claims is established, in part, by compliance with relevant sections of the ASME B&PV Code and informed by relevant operating experience. As such, claims 1 and 2 share some common features with the approach for Class 1 components.

Supplemental measures will be applied to support these claims for VHI components. These will include additional fracture toughness testing to directly characterise the fracture toughness of the material in order to inform the defect tolerance assessment and for some VHI components, additional stringent control of chemical composition will be specified to minimise the effect of degradation mechanisms such as irradiation embrittlement or thermal ageing. High reliability manufacturing inspections will be applied to the VHI components, for which the inspection system, including procedure, equipment, and personnel, will be qualified according to the European Network for Inspection & Qualification (ENIQ)-based methodology for qualification of non-destructive testing. The approach for inspection qualification is described in [Ref-8-9]. These qualified inspections provide high confidence in establishing the absence of significant defects at the end of the manufacturing and at the Pre-Service Inspection (PSI) stages. The output of the PSI, completed before the start of operation, is the generation of a set of benchmark data against which future ISI results can be compared. Claim 3 details the hydrostatic testing, conducted in accordance with requirements of the ASME BP&V Code.

The fourth claim is that the VHI components are tolerant to through life degradation. This is demonstrated by the results of assessments of through life crack growth, to show that such mechanisms will not threaten integrity over a specific interval. This exceeds conventional design code requirements to provide a further demonstration of integrity, by acknowledging that defects may be present and demonstrating tolerance to them. Defect tolerance is demonstrated by fracture assessments to establish tolerance to all defects smaller than a Qualified Examination Defect Size (QEDS) by a size margin of two. For VHI components, the elastic plastic fracture mechanics methodology of the R6 [Ref-8-10] procedure will be used for defect tolerance assessment, as described in [Ref-8-11]. Evidence is provided to identify how pressure-temperature limits are prescribed and controlled to prevent rupture, particularly at low temperatures during operation.
A limited set of VHI regions are selected for R6 defect tolerance assessment in GDA, chosen by a process of weld ranking. These include VHI welds in the RPV, Main Steam piping and inboard MSIV. A nozzle crotch corner in the RPV has also been selected for assessment. These are regarded as limiting locations where defect tolerance will be at a minimum. With the exception of the MS nozzle crotch corner, parent material is not selected for assessment in GDA; weld metals are judged to be less defect tolerant for the following reasons:

- Both plate and forging parent metal will have lower incidence of manufacturing defects as compared with welds.
- The likely orientation and nature of manufacturing defects in parent material are well understood and readily detectable by manufacturing and pre-service inspection.
- Parent material will have higher fracture toughness, as compared with welds.
- Parent metals are subject to minimal residual stress, as compared with welds.

The results of the defect tolerance assessments establish a QEDS for each region subject to assessment. Inspection qualification, conducted in accordance with ENIQ methodology, will be applied to confidently establish capability of detection for defects equal to or larger than the QEDS. The topic reports for VHI components describe R6 defect tolerance assessment in detail.

The fifth claim is that systems are provided to effectively forewarn of failure. Early indication of degradation is provided to prompt corrective action before gross failure occurs. This is achieved by specification of ISI to detect any degradation in good time before defect growth could significantly compromise structural integrity. ISI is also used to periodically confirm the absence of unanticipated degradation. ISI is a particularly important provision to forewarn of failure, it will be specified for the VHI locations in accordance with the robust ENIQ-based qualification methodology, as described in [Ref-8-12]. Environmental plant surveillance, leak detection and leak testing will be identified as evidence of diversity for forewarning of structural failure. To periodically confirm that the material property values applied in design analysis remain appropriate throughout the plant lifetime, a programme of surveillance sampling will be specified. This will provide samples for testing of mechanical properties, fracture toughness and corrosion resistance properties to account for the effects of irradiation embrittlement and thermal ageing.

The ABWR design has been subject to detailed review for UK application; each topic report for VHI components identifies a design philosophy report describing how component design has developed to enhance safety and reliability.
The following key items are described in the RPV topic report:
• 'Avoidance of Failure based on Defect Tolerance Assessment' concerns the demonstration of avoidance of fracture; actions were raised in relation to material properties, fracture assessment, pre-service inspection and the need for compatibility of these aspects in a combined approach.
• 'CRD Penetration Design' demonstrates the robust design of the control rod drive penetrations in the RPV.
• 'RPV Design' demonstrates the choice of materials that will form the main pressure boundary.

8.5.5 Materials and Degradation

The topic reports identify evidence to justify the structural integrity of those UK ABWR components included within the scope of this sub-chapter for a 60 year period of operation. In order to do this, it is necessary to take account of the potential for degradation.
Materials selection reports are identified in the topic reports for VHI components as evidence that highly robust materials have been selected that are compatible with the environment in which they will operate. Other topic reports, for Class 1 components and BOP, will identify that material selection is generally to be in accordance with appropriate design codes and is made with regard to the ALARP (As Low As Reasonably Practicable) principle.

Operating experience (OPEX) provides a valuable source by which understanding of susceptibility to degradation is developed and maintained. The UK ABWR benefits from this by including design enhancements to decrease the potential for degradation, as compared with other Light Water Reactor (LWR) designs. The materials selection reports, described above, include consideration of potential degradation mechanisms that may affect components of ABWR plant, as indicated by OPEX.

Some components of earlier BWR plants have experienced degradation by various mechanisms, notably by Stress Corrosion Cracking (SCC). [Ref-8-13] describes the UK ABWR strategy for avoidance of SCC. Enhancements that have been included to minimise the potential for SCC include selecting materials resistant to corrosion and optimisation of the manufacturing processes. The UK ABWR will operate with a water chemistry regime intended to prevent degradation. Specification of reactor coolant water chemistry is discussed in the RPV topic report. In this chapter, Hydrogen Water Chemistry (HWC), On Line Noble metal Chemistry (OLNC) and Depleted Zinc Oxide (DZO) additions are the reference reactor chemistry regime for the UK ABWR.
The measures described in [Ref-8-13] are considered to effectively minimise the potential for SCC, however, it is acknowledged that the potential for such degradation cannot be eliminated for a 60-year period. For this reason, in-service inspection will form an important element for control of degradation through periodic monitoring, particularly at locations where OPEX indicates vulnerability.
ISI is discussed in each topic report and the reactor internals topic report describes an augmented ISI programme to detect degradation by SCC. For VHI regions, ISI will be qualified in accordance with ENIQ methodology. More generally, ISI will be specified in accordance with Section XI of the ASME B&PVC for Class 1, 2 and 3 components. The topic reports also identify evidence of design for effective inspection.
'Material for RPV Pressure Boundary' is another key item. It is related to the specification and control of RPV material properties and associated manufacturing processes. In particular, it is necessary to clarify the chemical composition of forgings, and the processes to be applied for forging, welding and cladding. The RPV topic report provides information regarding RPV materials specification and manufacture, and fully addresses the questions raised during GDA.

8.6 Load Condition

Plant events affect mechanical systems and components. The load conditions due to the plant events and these load combinations are considered to evaluate the structural integrity. In addition, the plant events are classified into the plant operating conditions: Operational Condition I, II, III, IV and Test Condition.

8.6.1 Plant Operating Condition

Postulated events that the plant will or might credibly experience during design life are considered, to establish the design condition for mechanical systems and components. These events are classified into five plant conditions determined in PCSR chapter 5: General Design Aspects, section 5.4.

8.6.2 Service Load

The plant events are divided into two groups; plant operating events during which thermal-hydraulic transients occur, and dynamic loading events due to accidents, earthquakes and certain operating conditions. The service loads, which are normal loads and dynamic loads induced by these events, are outlined as follows:

(1) Normal Loads

- Dead loads
- Live loads: The loads induced by movable equipment loads.
- Pressure loads: Lateral and vertical pressure of liquids.
- Thermal loads: Thermal effects and loads during plant operating events.
- Reaction loads: Piping and equipment reactions during plant operating events.

(2) Dynamic Loads [Ref-8-14]

- Safety Relief Valve (SRV) Loads: Hydrodynamic loads induced by pressure waves, due to SRV actuation.
- LOCA Loads: Hydrodynamic loads and Reactor Building Vibration (RBV) loads induced by LOCAs.
- Seismic Loads: RBV loads due to earthquakes.

8.6.3 Load Combinations

Load combinations are associated with normal operation, postulated accidents, specified earthquake and other RBV events. Therefore, the load combinations are appropriately considered to evaluate structural integrity. The principles of the load combination for mechanical systems and components are as follows:

(1) The loads induced in an event are combined.

(2) When independent events occur simultaneously in certain probabilities, the loads from each event are combined.

(3) The combination of a hazard and the hazard-dependent event is considered in the frequency of the initiating hazard.

8.7 Assumptions, Limits and Conditions for Operation

8.7.1 Purpose

One purpose of this generic PCSR is to identify constraints that must be applied by a future licensee of a UK ABWR plant to ensure safety during normal operation, fault and accident conditions. This applies to the scope of GDA, and primarily Class 1 and 2 for SSCs. The general principles are defined in Generic PCSR Chapter 4: Assumptions and Limits and Conditions for Operation, section 4.12.

This section provides a summary of the Assumptions and LCOs that apply specifically to the scope of Structural Integrity.

8.7.2 LCOs

As for LCOs, some of these constraints are maximum or minimum limits on the values of system parameters, such as pressure or temperature, whilst others are conditional, such as prohibiting certain operational states or requiring a minimum level of availability of specified equipment.

Design conditions are specified to substantiate the structural integrity of SSCs. Thermal cycles for 60 years life time are assumed for SSCs for which ‘design by analysis’ is applied. And essential parameters related temperature and pressure are defined as LCOs for RPV as described in PCSR Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems, section 12.3.4. The LCOs applying on the reactor coolant systems and the reactivity control systems are identified in the sections related to each individual system as described in Chapter 12, section 12.6.2. These LCOs mainly define the initial conditions of fault analyses in Fault Studies and other safety analyses.

8.7.3 Assumptions

Design conditions are specified to substantiate the structural integrity of SSCs. Load conditions are considered as described in section 8.6 of this Chapter. Thermal cycles for 60 years life time are assumed for SSCs for which ‘design by analysis’ is applied. LOCA and Seismic Loads as Dynamic Loads are assumed. Mechanical loads such as Dead Loads are assumed based on the design conditions of the Reference Plant as long as there is not design change. For material selection, OLNC+HWC+DZO is assumed as reactor chemistry regime as described in Chapter 23: Reactor Chemistry. And the assumptions required for safety case are described in the sections related to each individual system as described in Chapter 12, section 12.6.3.

8.8 Summary of ALARP Justification

This section presents a high level overview of how the ALARP principle has been applied for the structural integrity of metal components for a 60 year period of operation.
Generic PCSR Chapter 28: ALARP Evaluation presents a high level approach taken for demonstrating ALARP across all aspects of the design and operation. It presents an overview of how the UK ABWR design has evolved, the further options that have been considered across all technical areas resulting in a number of design changes, and how these contribute to the overall ALARP case. The approach to undertaking ALARP Assessment during GDA is described in the GDA ALARP Methodology [Ref-8-23] and GDA Safety Case Development Manual [Ref-8-24].
In this chapter, the structural integrity classification is performed and UK safety class 1 components are sub-divided into VHI, HI and Standard Class 1. The safety case for the UK ABWR design reference of each of the VHI components is based on the five safety claims listed in section 8.5.1 and is justified in a Topic Report for each VHI component [Ref-8-16], [Ref-8-17], [Ref-8-18]. This multi-legged approach, as explained in Section 8.5 of this Chapter, is in line with UK good practice, being based upon that advocated by the UK Technical Advisory Group on Structural Integrity (TAGSI) for Incredibility of Failure components.

Each of the three VHI Topic Reports describes an evaluation against the principle of ALARP of the tolerability of risks associated with the structural integrity of that VHI component. In addition, Topic Reports for the Class 1 feedwater piping and reactor internals have been produced. These TRs describe the corresponding structural integrity assessments against the ALARP principle.

The RPV design optimisation investigation into the latest design of existing operating plants was performed in accordance with the ALARP principle [Ref-8-16]. This ALARP evaluation methodology consisted of five stages below including quantified evaluation using a ranking methodology. This investigation was reviewed and assessed through expert panel meetings, whose member's expertise covered a wide range of subject areas which would be affected by the RPV design. These expert panel meetings concluded that the current RPV design is the best viable choice for the UK ABWR due to the well-established manufacturing and fabrication methods, and good track record (OPEX). Maintaining the current design also avoids the increase in risk which would come from using a novel design, mitigating the proposed design optimisation benefits. This was supported by a quantified ranking method in a feasibility study.

- Stage 1: Good practice
- Stage 2: Prioritisation of activities to be assessed
- Stage 3: Systematic search for risk-reduction measures
- Stage 4: Screening out of options that are not reasonably practicable
- Stage 5: Detailed assessment of reasonably practicable options

Another key example is ALARP evaluation of RPV Top Head. It is necessary to minimise the number and length of welds in the component in principle to reduce risk of gross failure of welding. The one body forging closure head of PWR is considered as Relevant Good Practice (RGP). However the diameter of ABWR RPV Top Head is approximately 1.5 times that of PWR's. The forging manufacturer capability becomes essential in the ALARP evaluation. The reference design of the Top Head comprises of a plate dome, multi-plates petals and a forged flange. Factors of material manufacturing, RPV fabrication and In-Service are considered in the ALARP evaluation. Manufacturing aspects were also considered in the ALARP evaluation. The BWR RPV diameter is larger than PWR's and it is beyond the capability of forging manufacturers to provide one-body forged Top Head for ABWR. As a result of the detailed assessment, the RPV Top Head design comprising of a plate dome, multi-plates petals and a forged flange, is concluded as the most appropriate for use in the UK ABWR.

The justification of the use of Set-In Nozzles for RPV was also performed. Optioneering of the Set-On Nozzle, Set-In Nozzle and Integrally-Forged Nozzle types are considered in the ALARP evaluation. BWR RPV diameter is larger than PWR's and the Integrally-Forged Nozzle type has a detriment in manufacturability. The Set-In Nozzle type has a benefit that welding location is apart from high stress region of nozzle corner. As a result of the detailed assessment, the Set-In Nozzle type is concluded as the most appropriate for use in the UK ABWR.

In the RPV design optimisation investigation, parameters that are important in the specification of material properties and manufacturing processes are discussed. Many of the same principles are applicable to the MS Piping and MSIV to provide an ALARP argument [Ref-8-17], [Ref-8-18].

It is noted that the structural integrity aspects of metal components important to safety on the Japanese ABWR reference design are well understood, using proven technology. Hence a significant aspect of demonstrating the application of relevant good practice in the design of the UK ABWR is to generally adopt the Japanese reference design, with necessary modifications to enhance safety margins even further where it is reasonably practicable to do so. Thus, the designs of the metal components of the UK ABWR within the scope of this chapter are mainly the same as for the four operating ABWRs in Japan, which together have provided many years of operating experience.
As for material selection, the most significant nuclear safety risks associated specifically with the Structural Integrity of metal components in the UK ABWR design over the plant lifetime are:

Significant material degradation due to:

- Flow-Accelerated Corrosion (FAC) and Erosion Corrosion (EC)
- Stress Corrosion Cracking (SCC)
- Irradiation-Assisted Stress Corrosion Cracking (IASCC)
- Neutron Irradiation Embrittlement
- Fatigue
- Other mechanisms (including general corrosion, pitting, etc.)

Based on RGP and OPEX, the material selection optioneering is performed and weighting and the scoring ALARP approach is applied considering factors below:

- Design requirements/Mechanical properties
- Procurement availability
- Manufacturability/Weldability
- Inspectability
- OPEX
- SCC/IASCC
- FAC/Corrosion
- Fatigue/Environmental fatigue
- Thermal aging/Irradiation embrittlement
- Release of detrimental materials/Source term
- Radiological dose

Materials for UK ABWR metal components are appropriately selected to reduce risks caused by material degradations ALARP. It is recognised that material selection affects not only the potential for material degradation, but must also consider the interactions with the chosen UK ABWR water chemistry (Chapter 23: Reactor Chemistry) and the need to minimise operating doses (Chapter 20: Radiation Protection). For example, use of Low Cobalt Material reduces Co corrosion products and minimises the operating doses (Chapter 20) and the decommissioning Source Term (Chapter 31: Decommissioning).

8.9 Conclusions

This chapter describes how the structural integrity of metal SSCs that are significant to safety is assured for the UK ABWR. The process commences with the establishment of the safety functions required of a particular structural component, following which a system of safety structural classification is applied to determine the measures warranted to provide suitably robust assurance of structural integrity. The method of safety structural classification is based on postulated structural failure of the component with the associated loss of its safety function(s), taking account both of the direct and indirect unmitigated consequences of the failure.

This chapter describes the methods to establish SFCs and structural integrity classification in sections 8.3 and 8.4 respectively. It also provides references to Topic Reports (TR) that provide the detailed substantiation of structural integrity claims for ABWR components. The structure and content of these varies according to classification, as described in section 8.5 of this chapter. The nature and extent of evidence necessary to justify structural reliability is summarised for all classes of component that are significant to nuclear safety. This is based on compliance with appropriate design codes and standards, with supplementary measures identified to provide additional evidence of both defect avoidance and defect tolerance for components with the highest safety significance. The approach to specify load conditions and their combination for input to assessments that will support the structural integrity safety case is described in section 8.6.

Summaries of seven topic reports, which cover 3 VHI components, 2 Class 1 components, 1 Class 2 components and 1 Class 3 components, are appended to this chapter. The summaries explain how, for GDA, the structural integrity of the varying Safety Classes is substantiated.

A key element of this chapter has been to show that risks of accidents from all safety significant metal structural failures are very low; this is especially important for VHI components as the claim is that the failure of a VHI component can be discounted. This discounting of all VHI component failures is equivalent to the international concept of ‘practical elimination’ as defined in part 7 of a recent IAEA Technical Document [Ref- 8- 25] and section 2 of [Ref- 8- 26]. A demonstration of ‘practical elimination’ is provided in section 8.5.4 of this chapter. Section 8.5.4 provides a high level summary of the claims, arguments and evidence covering the multiplicity of independent deterministic evidence to demonstrate a high confidence of low frequency of failure required of all UK ABWR VHI components. Concepts such as excellence of design, excellence of manufacture, rigorous testing etc. are also the basis for other structural metal components important to safety such as HI and Standard Class 1. The integrated structural integrity safety analysis of metal components in this chapter has not only demonstrated the risks of accidents from metal structural components are very low, but are also as low as is reasonably practicable to achieve.

8.10 References

[Ref-8-1] Hitachi-GE, "Genesis of ABWR design", GA91-9901-0011-00001 (XE-GD-0083) Rev. A, October 2013.

[Ref-8-2] Hitachi-GE, "Summary of the Design of Reactor Pressure Vessel for UK ABWR", GA91-9201-0003-00035 (RE-GD-2010) Rev.0, March 2014.

[Ref-8-3] Hitachi-GE, "Summary of the Design of Main Steam Piping for UK ABWR", GA91-9201-0003-00036 (PD-GD-00010) Rev.0, March 2014.

[Ref-8-4] Hitachi-GE, "Summary of the Design of Feedwater Piping for UK ABWR", GA91-9201-0003-00037 (PD-GD-0012) Rev.0, March 2014.

[Ref-8-5] Hitachi-GE, "Summary of the Design of Main Steam Isolation Valve for UK ABWR", GA91-9201-0003-00038 (PVD-GD-0004) Rev.0, March 2014.

[Ref-8-6] Hitachi-GE, "Preliminary Safety Report on Structural Integrity", GA91-9901-0005-00001 (XE-GD-0113) Rev. C, March 2014.

[Ref-8-7] Hitachi-GE, "Structural Integrity Classification Procedure", GA91-9201-0003-00054 (RD-GD-0001) Rev.0, March 2014.

[Ref-8-8] Hitachi-GE, "Categorisation and Classification of Structures, Systems and Components" GA91-9901-0007-00001 (XE-GD-0104) Rev. B, March 2014.

[Ref-8-9] Hitachi-GE, "Inspection Qualification Strategy", GA 91-9201-0003-00057 (G-TY-53082) Rev.0, March 2015.

[Ref-8-10] EDF Energy, R6 Revision 4, Assessment of the Integrity of Structures Containing Defects, 2013.

[Ref-8-11] Hitachi-GE, "Defect Tolerance Assessment Plan", GA91-9201-0003-00056 (RD-GD-0002) Rev.0, March 2014.

[Ref-8-12] Hitachi-GE, "Outline of the PSI and ISI Plan for UK ABWR", GA 91-9201-0003-00039 (PD-GD-0009) Rev.0, March 2014.

[Ref-8-13] Hitachi-GE, "UK ABWR - Approach for the Avoidance of SCC" GA11-1001-0003-00001 (1D-GD-0003) Rev.0, March 2014.

[Ref-8-14] Hitachi-GE, "Preliminary Safety Report on Civil Engineering and External Hazards", GA91-9901-0004-00001 (XE-GD-0112) Rev. B, March 2014.

[Ref-8-15] Hitachi-GE, "List of Safety Category and Class for UK ABWR", GA91-9201-0003-00266 (AE-GD-0224) Rev.4, August 2017.

[Ref-8-16] Hitachi-GE, "Topic Report on RPV Structural Integrity", GA91-9201-0001-00076 (RD-GD-0020) Rev.2, June 2017.

[Ref-8-17] Hitachi-GE, "Topic Report on MS Piping Structural Integrity", GA91-9201-0001-00077 (ZD-GD-0013) Rev.2, June 2017.

[Ref-8-18] Hitachi-GE, “Topic Report on MSIV Structural Integrity”, GA91-9201-0001-00078 (PVD-GD-0016) Rev.2, June 2017.

[Ref-8-19] Hitachi-GE, “Topic Report on Reactor Internals”, GA91-9201-0001-00080 (CD-GD-0001) Rev.1, December 2016.

[Ref-8-20] Hitachi-GE, “Topic Report on FDW Piping Structural Integrity”, GA91-9201-0001-00079 (ZD-GD-0014) Rev.2, June 2017.

[Ref-8-21] Hitachi-GE, “Topic Report on SLC Storage Tank Structural Integrity”, GA91-9201-0001-00268 (RWE-GD-9003) Rev. 0, June 2017.

[Ref-8-22] Hitachi-GE, “Topic Report on HP Turbine Casing Structural Integrity”, GA91-9201-0001-00278 (CXJ-GD-1013) Rev. 0, June 2017.

[Ref-8-23] Hitachi-GE, “GDA ALARP Methodology”, GA10-0511-0004-00001 (XD-GD-0037) Rev.1, November 2015.

[Ref-8-24] Hitachi-GE, “GDA Safety Case Development Manual”, GA10-0511-0006-00001 (XD-GD-0036) Rev.3, June 2017.

[Ref-8-25] Consideration on the Application of the International Atomic Energy Agency Safety (IAEA) Safety Requirements for the Design of Nuclear Power Plants (IAEATECDOC-1791) 2016.

[Ref-8-26] Hitachi-GE, Demonstration of Practical Elimination of or Large Fission Product Release for UK ABWR, GA91-9201-0003-22179 (AE-GD-0992) Rev. 0, June 2017.

Appendix A: Safety Functional Claims Table

Claim Tree for Ch. 8 (Structural Integrity)

	Top Claim for Structural Integrity						Safety Functional Claim (SFC) for Structural Integrity		
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)				
	PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.2 Identification of ABWR Safety Functions)		PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.6 Summary of Safety Category and Classification)		Topic Report on Fault Assessment (UE-GD-0071) Table.4.2-1 Fault Schedule		State	Claim ID	Claim Contents
1	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.1	For Very High Integrity (VHI) Structural Integrity is assured by good design.
2	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.2	For Very High Integrity (VHI) Structural Integrity is assured by material selection and quality.
3	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.3	For Very High Integrity (VHI) Functional testing provides a demonstration of integrity at start of life.
4	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.4	For Very High Integrity (VHI) Through life integrity is demonstrated by analysis and inspection.
5	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.5	For Very High Integrity (VHI) Inspection and Monitoring regularly validate integrity through life.

Claim Tree for Ch. 8 (Structural Integrity)

	Top Claim for Structural Integrity						Safety Functional Claim (SFC) for Structural Integrity		
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)				
	PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.2 Identification of ABWR Safety Functions)		PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.6 Summary of Safety Category and Classification)		Topic Report on Fault Assessment (UE-GD-0071) Table.4.2-1 Fault Schedule		State	Claim ID	Claim Contents
6	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.6	For Standard Class 1 High Quality Is Achieved Through Good Design And Manufacture.
7	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.7	For Standard Class 1 Design Code Assessment Provides Assurance Of Integrity.
8	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.8	For Standard Class 1 Hydrostatic pressure tests confirm integrity.
9	4	Containment of radioactive materials	4-1	Functions to form reactor coolant pressure boundary	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 4-1.9	For Standard Class 1 In-service inspection & monitoring forewarns of failure.

Claim Tree for Ch. 8 (Structural Integrity)

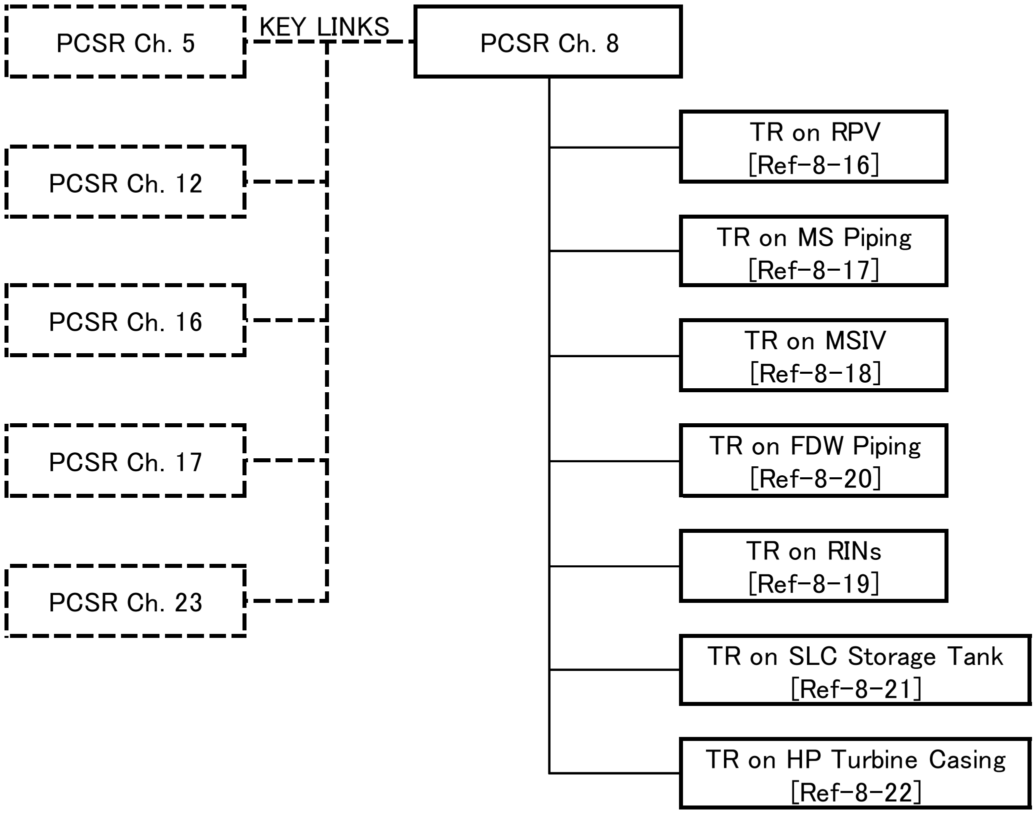
	Top Claim for Structural Integrity						Safety Functional Claim (SFC) for Structural Integrity		
	Fundamental Safety Function (FSF)		High Level Safety Function (HLSF)		Fault Schedule (Bounding Fault)				
	PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.2 Identification of ABWR Safety Functions)		PCSR Ch.5.4 (List of Safety Category and Class for UK ABWR (AE-GD-0224) 3.6 Summary of Safety Category and Classification)		Topic Report on Fault Assessment (UE-GD-0071) Table.4.2-1 Fault Schedule		State	Claim ID	Claim Contents
10	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 1-2.1	For Standard Class 1 High Quality Is Achieved Through Good Design And Manufacture.
11	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 1-2.2	For Standard Class 1 Design Code Assessment Provides Assurance Of Integrity.
12	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 1-2.3	For Standard Class 1 Hydrostatic pressure tests confirm integrity.
13	1	Control of Reactivity	1-2	Functions to maintain core geometry	-	No Claim	Operational Condition I, II, III, IV and Test Condition	SI SFC 1-2.4	For Standard Class 1 In-service inspection & monitoring forewarns of failure.

Appendix B: Safety Properties Claims Table

Safety Property Claims Table is not prepared for Structural Integrity.

NOT PROTECTIVELY MARKED

Appendix C: Document Map



Appendix D: Topic Report Summary – Reactor Pressure Vessel (RPV)

This appendix presents the summary of the TR on RPV [Ref-8-16]. This TR describes the structural integrity of RPV. Its purpose is to demonstrate that the probability of gross disruptive failure of the RPV main pressure boundary, without forewarning, is so low that it can be discounted. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design promotes very high integrity

Claim 2: High quality of manufacture is specified to achieve very high integrity

Claim 3: Functional testing confirms integrity at start of life

Claim 4: Lifelong integrity is demonstrated by failure analysis

Claim 5: Inspection, monitoring and surveillance confirm very high integrity through life

Each claim is substantiated by a series of arguments which, in turn, are validated by the identification of evidence. The approach has been chosen to conform to UK good practice for components classified as Very High Integrity (VHI). Existing evidence has been identified to substantiate, either in whole or part, the safety arguments. All key technical reports are complete, and consequently this revision of the topic report is essentially the final safety case. It is presented to demonstrate that the approach to justify RPV integrity remains satisfactory in accordance with regulatory expectations. Important topics where detailed work continues include material specification and testing, thermal, stress and fatigue analyses in accordance with ASME Code, R6 defect tolerance assessment and ENIQ qualification of inspection. The TR on RPV [Ref-8-16] reflects the current status and maturity of the available evidence and it is presented to justify the structural integrity of the VHI components of the RPV.

Appendix E: Topic Report Summary – Main Steam Piping

This appendix presents the summary of the TR on MS Piping [Ref-8-17]. This TR describes the structural integrity of MS Piping. The TR on MS Piping [Ref-8-17] presents the safety justification for Main Steam (MS) Piping structural integrity. Its main purpose is to demonstrate that the probability of gross disruptive failure of the MS Piping from Reactor Pressure Vessel to inboard Main Steam Isolation Valves, without forewarning, is so low that it can be discounted. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design promotes very high integrity

Claim 2: High quality of manufacture is specified to achieve very high integrity

Claim 3: Functional testing confirms integrity at start of life

Claim 4: Lifelong integrity is demonstrated by failure analysis

Claim 5: Inspection, monitoring and surveillance confirm very high integrity through life

The approach has been chosen to conform to UK good practice for components of Very High Integrity (VHI) classification. Existing evidence has been identified to substantiate, either in whole or part, the safety arguments presented. All key technical reports are complete, and consequently this revision of the topic report is essentially the final safety case. It is presented to demonstrate that the approach to justify MS Piping integrity remains satisfactory in accordance with regulatory expectations. Important topics where detailed work continues include material specification and testing, thermal, stress and fatigue analyses in accordance with ASME Code, R6 defect tolerance assessment and ENIQ qualification of inspection. The TR on MS Piping [Ref-8-17] reflects the current status and maturity of the available evidence and it is presented to justify the structural integrity of the VHI components of the MS Piping.

Appendix F: Topic Report Summary – Main Steam Isolation Valve (MSIV)

This appendix presents the summary of the TR on MSIV [Ref-8-18]. This TR describes the structural integrity of the Main Steam Isolation Valve (MSIV). Its main purpose is to demonstrate that the probability of gross disruptive failure of the MSIV main pressure boundary, without forewarning, is so low that it can be discounted. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design promotes very high integrity

Claim 2: High quality of manufacture is specified to achieve very high integrity

Claim 3: Functional testing confirms integrity at start of life

Claim 4: Lifelong integrity is demonstrated by failure analysis

Claim 5: Inspection, monitoring and surveillance confirm very high integrity through life

Each claim is substantiated by a series of arguments which, in turn, are validated by the identification of evidence. The approach has been chosen to conform to UK good practice for components classified as Very High Integrity (VHI). Existing evidence has been identified to substantiate, either in whole or part, the safety arguments. All key technical reports are complete, and consequently this revision of the topic report is essentially the final safety case. It is presented to demonstrate that the approach to justify MSIV integrity remains satisfactory in accordance with regulatory expectations. Important topics where detailed work continues include material specification and testing, thermal, stress and fatigue analyses in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, R6 defect tolerance assessment and European Network for Inspection Qualification (ENIQ) qualification of inspection. The TR on MSIV [Ref-8-18] reflects the current status and maturity of the available evidence and it is presented to justify the structural integrity of the VHI components of the MSIV.

Appendix G: Topic Report Summary – Feedwater Piping

This appendix presents the summary of the TR on FDW Piping [Ref-8-20]. The TR on FDW Piping [Ref-8-20] is a supporting reference to this chapter, which presents the safety justification for the structural integrity of the Standard Class 1 regions of the Feedwater (FDW) pipework, is presented in support of the Generic Design Assessment (GDA). Its main purpose is to demonstrate that structural integrity is assured to a degree commensurate with the classification. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design and design code assessment provides assurance of integrity

Claim 2: High quality of manufacture is specified to ensure integrity is maintained throughout service

Claim 3: Functional testing confirms integrity at start of life

Claim 4: In-service inspection and monitoring forewarns of failure

The approach has been chosen to conform to UK good practice for Class 1 components. Existing evidence has been identified to substantiate, either in whole or part, the safety arguments presented. All key technical reports are complete, and consequently this revision of the topic report is essentially the final safety case. It is presented to demonstrate that the approach to justify Feedwater system integrity remains satisfactory in accordance with regulatory expectations. Important topics where detailed work continues include material specification and testing, thermal, stress and fatigue analyses in accordance with the design code, Pre-Service Inspection and In-Service Inspection.
The TR on FDW Piping [Ref-8-20] reflects the current status and maturity of the available evidence and it is presented to justify the structural integrity of the FDW Piping.

Appendix H: Topic Report Summary – Reactor Internals

This appendix presents the summary of the TR on Reactor Internals [Ref-8-19]. The TR on Reactor Internals [Ref-8-19] describes the structural integrity of the Standard Class 1 Reactor Internals (RINs). The RINs perform a number of functions but can broadly be considered as those which provide support to the core structure and those which do not.
The main purpose of this report is to demonstrate that the probability of structural failure of the RINs is tolerably low and As Low As Reasonably Practicable (ALARP) in keeping with the identified safety classification of Class 1. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: High Quality Is Achieved Through Good Design and Manufacture

Claim 2: Design Code Assessment Provides Assurance of Integrity

Claim 3: Functional Tests Confirm Integrity

Claim 4: In-Service Inspection & Monitoring Forewarns of Failure

Each claim is substantiated by a series of arguments which, in turn, are validated by the identification of evidence. The approach has been chosen to conform to UK good practice. Existing evidence has been identified to substantiate, either in whole or part, the safety arguments. All key technical reports are complete, and consequently this revision of the topic report is essentially the final safety case. It is presented to demonstrate that the approach to justify RINs structural integrity remains satisfactory in accordance with regulatory expectations. Important topics where detailed work continues include materials and chemistry specification, thermal, stress and fatigue analyses in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, identification of In-Service Inspection requirements and qualification of Pre-Service Inspection.
The TR on Reactor Internals [Ref-8-19] reflects the current status and maturity of the available evidence and it is presented to justify the structural integrity of the Reactor Internals.

Appendix I: Topic Report Summary – SLC Storage Tank

This appendix presents the summary of the TR on SLC Storage Tank [Ref-8-21]. This TR describes the structural integrity of SLC Storage Tank. The TR on SLC Storage Tank [Ref-8-21] presents the safety justification for the structural integrity of the Class 2 SLC Storage Tank part of the Standby Liquid Control (SLC) system. Its main purpose is to demonstrate that structural integrity is assured to a degree commensurate with the classification. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design and design code assessment provides assurance of integrity

Claim 2: High quality manufacture is specified to ensure integrity is maintained throughout service

Claim 3: Functional testing confirms integrity at start of life

Claim 4: In service inspection and monitoring forewarns of failure

The approach has been chosen to conform to UK good practice for Class 2 components. Existing evidence has been identified to substantiate, either in whole or part, the safety arguments presented. It is presented to demonstrate that the approach to justify SLC Storage Tank integrity remains satisfactory in accordance with regulatory expectations.

Appendix J: Topic Report Summary – HP Turbine Casing

This appendix presents the summary of the TR on HP Turbine Casing [Ref-8-22]. This TR describes the structural integrity of the HP Turbine Casing. The TR on HP Turbine Casing [Ref-8-22] presents the safety justification for the structural integrity of the Class 3 HP Turbine Casing part of the Main Turbine system. Its main purpose is to demonstrate that structural integrity is assured to a degree commensurate with the classification. Demonstration is presented in the form of a multi-legged safety justification based on the following claims:

Claim 1: Sound design

Claim 2: High quality of manufacture

Claim 3: Functional testing and inspection

Claim 4: Maintenance plan

The approach has been chosen to conform to UK good practice for Class 3 components. Existing evidence has been identified to substantiate, either in whole or part, the safety arguments presented. It is presented to demonstrate that the approach to justify HP Turbine Casing integrity remains satisfactory in accordance with regulatory expectations.