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General Nuclear System

# General Nuclear System Ltd.

## UK HPR1000 GDA Project

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**Title:**

# Pre-Construction Safety Report

# Chapter 17

# Structural Integrity

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## 17.1 List of Abbreviations and Acronyms

| | |
|---|---|
| ACC | Accumulator |
| ALARP | As Low As Reasonably Practicable |
| CAE | Claims, Arguments, Evidence |
| CGN | China General Nuclear Power Corporation |
| CPR1000 | Chinese Pressurised Reactor |
| CRDM | Control Rod Drive Mechanism |
| CSR | Component Safety Report |
| DSM | Defect Size Margin |
| DTA | Defect Tolerance Assessment |
| ELLDS | End of Life Limiting Defect Size |
| ENIQ | European Network for Inspection and Qualification |
| EOMR | End of Manufacturing Report |
| GDA | Generic Design Assessment |
| HPR1000 (FCG3) | Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 |
| IAEA | International Atomic Energy Agency |
| ISI | In-Service Inspection |
| KIF | Fatigue Monitoring System [FMS] |
| KIL | Leakage Monitoring System [LMS] |
| KIR | Loose Parts and Vibration Monitoring system [LPVM] |
| LBB | Leak Before Break |
| LFCG | Lifetime Fatigue Crack Growth |
| MCL | Main Coolant Line |
| MSL | Main Steam Line |
| MSIV | Main Steam Isolation Valve |
| MSQA | Management of Safety and Quality Assurance |
| NDT | Non-Destructive Testing |

| | |
|---|---|
| OPEX | Operating Experience |
| PCSR | Pre-Construction Safety Report |
| PSI | Pre-Service Inspection |
| PSR | Preliminary Safety Report |
| PWR | Pressurised Water Reactor |
| PZR | Pressuriser |
| QEDS | Qualified Examination Defect Size |
| RCP | Reactor Coolant Pump |
| RCPB | Reactor Coolant Pressure Boundary |
| RGP | Relevant Good Practice |
| RPV | Reactor Pressure Vessel |
| $RT_{NDT}$ | Reference Nil Ductility Transition Temperature |
| RVI | Reactor Vessel Internals |
| SCC | Stress Corrosion Cracking |
| SFC | Single Failure Criterion |
| SG | Steam Generator |
| SL | Surge Line |
| SSC | Structures, Systems and Components |
| SSE | Safe Shutdown Earthquake |
| TAGSI | Technical Advisory Group on Structural Integrity |
| TJ | Technical Justification |
| UK HPR1000 | UK version of the Hua-long Pressurised Reactor |
| UT | Ultrasonic Testing |

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Fatigue Monitoring System (KIF [FMS]).

## 17.2 Introduction

This Chapter represents the top level safety case that describes the demonstration of structural integrity for the UK version of the Hua-long Pressurised Reactor (UK HPR1000). It presents the safety arguments and evidence for nuclear safety-related metal Structures, Systems and Components (SSC). The chapter presents how they are organised and developed to demonstrate an adequate level of structural integrity commensurate with the required level of structural reliability and the consequence of gross failure.

Firstly, this chapter describes the applicable codes and standards applied to the structural integrity area, it then presents the methodology for structural integrity classification and current completed results of structural integrity classification. Subsequently, the structured arguments and associated evidence are presented for High Integrity Components (HIC) and components of other classes.

For HICs, where gross failure is intolerable and for which no physical protection is provided or protection provision is not reasonably practicable, the safety arguments and evidence are presented in the approach of Technical Advisory Group on Structural Integrity (TAGSI), in line with UK good practice. For such components, the arguments enhanced by additional measures for defect tolerance and the application of qualified manufacturing inspections (based on European Network for Inspection and Qualification (ENIQ) methodologies) are shown to provide a robust demonstration that the component is free from structural defects of concern before entering service or is tolerant of defects to through-life degradation. For components where the postulated failure consequences are less severe than those for HICs, the safety arguments are provided in compliance with the appropriate codes and standards.

In this chapter, the safety arguments of each structural integrity class component are presented in the form of a series of Claims, Arguments, Evidence (CAE), and further supported by a suite of logically-referenced documentary evidence. This version of Pre-Construction Safety Report (PCSR) covers a section of the Component Safety Report (CSR) for HICs to present specific details of design features and requirements, code compliance assessments, material selection, quality manufacture, manufacturing inspection qualification, defect tolerance assessments and In-Service Inspection (ISI) requirements, but also contains the several other sections to demonstrate the main contents of CSR for non-HICs.

Finally, in addition to the high-level description of the loading conditions under which structural integrity should be evaluated, this chapter presents how the As Low As Reasonably Practicable (ALARP) principle is applied in the structural integrity area.

The present safety case of structural integrity is produced based on the design reference version 2.1, as described in UK HPR1000 Design Reference Report (Reference [1], Rev. E). The safety assessment results are documented in this chapter and corresponding safety assessment reports.

### 17.2.1 Objective

The ***Fundamental Objective*** of the UK HPR1000 is that:

*The Generic UK HPR1000 could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment.*

To underpin this objective**,** five level 1 claims and a number of level 2 claims are developed and presented in PCSR Chapter 1. This Chapter supports ***Claim 3.3*** and ***Claim 3.4*** derived from the level 1 ***Claim 3.***

***Claim 3*** *(Level 1 Claim): Nuclear safety*

*The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is as low as reasonably practicable.*

***Claim 3.3*** *(Level 2 Claim): The design of the processes and systems has been substantiated and the safety aspects of operation and management have been substantiated.*

***Claim 3.4*** *(Level 2 Claim): The safety assessment shows that the nuclear safety risks are ALARP.*

***Claim 3.3*** and ***Claim 3.4*** can be split further into level 3 claims. According to the study and assessment scope of the structural integrity area, ***Claim 3.3.2, Claim 3.3.3, Claim 3.3.6, Claim 3.3.7*** and ***Claim 3.4.8*** are relevant to this chapter.

***Claim 3.3.2*** *(Level 3 Claim): The design of the Reactor Coolant System has been substantiated.*

***Claim 3.3.3*** *(Level 3 Claim): The design of the Safety Systems has been substantiated.*

***Claim 3.3.6*** *(Level 3 Claim): The design of the Auxiliary Systems has been substantiated.*

***Claim 3.3.7*** *(Level 3 Claim): The design of the Steam & Power Conversion System has been substantiated.*

***Claim 3.4.8*** *(Level 3 Claim): All reasonably practicable options to improve nuclear safety have been adopted, demonstrating that the risk is ALARP.*

To support the above claims, PCSR Chapter 17 develops a ***Chapter Claim***, which is consistent with Preliminary Safety Report (PSR) Chapter 17:

*The structural integrity of SSC is justified by adopting appropriate methods and it is demonstrated that plant risk due to structural failures remains both tolerable and as low as reasonably practicable (ALARP).*

The objective of this chapter is to support the chapter claim for justifying the structural integrity of nuclear safety-related metal SSCs within the scope, and demonstrating the plant risk to the UK HPR1000 as a result of structural failures is and remains both tolerable and ALARP throughout design lifetime.

### 17.2.2 Scope

17.2.2.1 SSCs Scope

The scope of this chapter is to substantiate the structural integrity of all metal components and structures that are related to nuclear safety during design lifetime for all conditions within the design basis.

According to the level 3 claims presented in Sub-chapter 17.2.1, the following systems are key to nuclear safety:

a) Reactor Coolant System.

b) Safety Systems.

c) Auxiliary Systems.

d) Steam & Power Conversion System.

Therefore, the scope of the structural integrity case will focus on the components and structures which form part of these systems, with a focus on the integrity of static components and structures related to nuclear safety. Other SSCs, such as lifting equipment, fuel handling and storage equipment, and ventilation systems are covered under the mechanical engineering area. Concrete structures are also omitted from this chapter, with the exception of the concrete containment steel liners. The more detailed system and component list for structural integrity area will be covered in Reference [2], *Equipment Structural Integrity Classification List*.

17.2.2.2 GDA Scope and Extent

For the UK HPR1000, the GDA scope and extent with respect to structural integrity is specified in Reference [3], *Scope for UK HPR1000 GDA Project*. It presents the main activities relevant to the demonstration of structural integrity during GDA and some items out of GDA.

During GDA, a series of individual CSRs are developed to provide the safety arguments and evidence to support structural integrity claims for the following SSCs in the CAE format:

a) Reactor Pressure Vessel (RPV).

b) Pressuriser (PZR).

c) Steam Generator (SG).

d) Main Coolant Lines (MCL).

e) Reactor Coolant Pump (RCP).

f) Main Steam Line (including Main Steam Isolation Valve (MSIV)).

g) Reactor Vessel Internals (RVI).

h) Surge Line (SL).

i) Accumulator (ACC).

j) Surge Tank.

The rationale for the selection of SSCs included for detailed evaluation will be presented in *Equipment Structural Integrity Classification List*. The selected SSCs include the major components that constitute the primary and secondary system pressure boundaries within the containment.

The detailed scope for each of the SSC groups is presented in their respective CSRs, and is summarised in each related appendix to this chapter. It is recognised that the extent and depth of these CSRs will be enhanced in line with the progress of the overall structural integrity demonstration.

### 17.2.3 Chapter Route Map

For the structural integrity area, the Fundamental Objective, Level 1 Claim 3, Level 2 Claims 3.3&3.4, Level 3 Claims 3.3.2, 3.3.3, 3.3.6, 3.3.7, 3.4.8 and Chapter Claim have been presented in Sub-chapter 17.2.1.

In order to support the above Chapter Claim, combined with the safety design basis and performance requirements of a nuclear facility, and taking the approach of TAGSI into account, the structural integrity demonstration will be developed to meet the following three Sub-claims:

a) Sub-Claim 1: High quality is achieved through good design and manufacture and functional testing.

b) Sub-Claim 2: The avoidance of fracture demonstrates that the High Integrity Components are tolerant of defects to through-life degradation.

c) Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

The chapter route map related to claims and arguments for HICs and SIC-1/SIC-2/SIC-3 (Structural Integrity Class-1/2/3) components is presented in Appendix 17A. The detailed CAE for HICs and SIC-1, SIC-2 and SIC-3 components are presented in Reference [4], *Safety Case Methodology for HIC and SIC Components*.

It should be noted that the structural integrity requirements of SSCs are based on assessing the consequences of gross failure and assigning an appropriate safety classification. Before implementing structural integrity assessments for SSCs in accordance with the above sub-claims, a systematic approach to establishing the levels of integrity claims with appropriate levels of reliability should be carried out. This is a fundamental part of the process and key prerequisite, with the aim of establishing a systematic method to determining structural integrity classification for each SSC and a reasonable procedure to produce the specific sub-claims, arguments and evidence for each level of SSC.

For the UK HPR1000, the structural integrity classification methodology and its current results, and the safety case methodology for HIC and SIC-1/SIC-2/SIC-3 components are established and can be found in References [5], [2] and [4].

### 17.2.4 Chapter Structure

This chapter is comprised of the following sections:

a) Sub-chapter 17.1 lists all the abbreviations and acronyms quoted in this chapter.

b) Sub-chapter 17.2 introduces the objective of this chapter, scope, chapter route map, the proposed chapter structure, supporting documents and the interfaces with other PCSR chapters.

c) Sub-chapter 17.3 presents the principal code selection principles, applicable codes and standards in structural integrity area and relevant applicability descriptions.

d) Sub-chapter 17.4 describes the safety functional requirements of metal components and structures which are important to nuclear safety for the UK HPR1000.

e) Sub-chapter 17.5 describes the structural integrity classification methodology and its current application to the UK HPR1000.

f) Sub-chapter 17.6 introduces the methodology for constructing safety cases for different classes of components and structures and the associated main CAE contents.

g) Sub-chapter 17.7 introduces the loading conditions to be considered in the design for integrity evaluation of components and structures.

h) Sub-chapter 17.8 introduces considerations and activities to present how the ALARP principle is applied in the structural integrity demonstration.

i) Sub-chapter 17.9 summarises the main aspects of this chapter.

j) Sub-chapter 17.10 presents the reference documents and standards within this chapter.

k) Appendix 17A presents the chapter route map for the structural integrity area.

l) Appendices 17B to 17F present a summary of the CSRs for this chapter. This version only covers the CSRs of RPV, PZR, SG, MCL and RVI, the rest will be added in the next version.

### 17.2.5 Supporting Documents

PCSR Chapter 17 is considered as a summary of safety cases for demonstrating structural integrity reliability of SSCs. It presents the chapter claim and sub-claims of the different structural integrity class components and is considered as a Tier 1 document. The basis of safety case reports, methodology reports, and the generic design scheme report are considered as Tier 2 documents which present the relevant arguments. CSRs are also considered as Tier 2 documents, as they summarise the safety arguments and present sufficient diversity of evidence to substantiate the structural reliability of HICs and significant non-HIC components in the CAE form. Hence, the references of CSRs and their supporting documents are regarded as Tier 3 documents as evidence for providing the detailed design or assessment information for a particular component or group of component. Herein, the supporting documents mainly refer to CSRs and Tier 3 documents.

For the UK HPR1000, the following CSRs are provided and present detailed arguments and evidence to demonstrate the structural integrity reliability of all HIC components and significant SIC-1/SIC-2/SIC-3 components.

a) For HIC:

   1) Reactor Pressure Vessel Component Safety Report, in Reference [6].

   2) Pressuriser Component Safety Report, in Reference [7].

   3) Steam Generator Component Safety Report, in Reference [8].

   4) Main Coolant Lines Component Safety Report, in Reference [9].

   5) Reactor Coolant Pump Component Safety Report.

   6) Main Steam Lines Component Safety Report.

b) For SIC-1

   1) Reactor Vessel Internals Component Safety Report, in Reference [10].

   2) Surge Line Component Safety Report.

c) For SIC-2

   1) Accumulator Component Safety Report.

d) For SIC-3

   1) Surge Tank Component Safety Report.

In GDA Step 3, five CSRs (for RPV/PZR/SG/MCL/RVI) have been issued and submitted. In GDA Step 4, the above five CSRs will be updated according to the design assessment progress and the relevant review comments. The new CSRs of HIC and SIC-1/2/3 components are presented as well.

In addition to the above CSRs, the relevant supporting documents of these CSRs also provide useful information to support the structural integrity demonstration of a particular component or group of components. These supporting documents mainly cover design specification, sizing calculation report, equipment drawing, main material specification, general mechanical analysis reports, defect tolerance assessments, Non-Destructive Testing (NDT) inspection specifications, GDA technical justifications of limiting locations, requirements of PSI and ISI and ageing and degradation of material etc.

In general, the extent and scope of CSRs and their supporting documents are proportional to the structural integrity class of the SSCs.

### 17.2.6 Interfaces with Other Chapters

The interfaces with other chapters are listed in table T-17.2-1.

T-17.2-1 Interfaces between Chapter 17 and Other Chapters

| PCSR Chapter | Interface |
|---|---|
| Chapter 1 Introduction | Chapter 1 provides the Fundamental Objective, Level 1 Claims and Level 2 Claims.<br>Chapter 17 provides chapter claims and arguments to support relevant claims in Chapter 1. |
| Chapter 4 General Safety and Design Principles | Chapter 4 provides the general safety and design principles.<br>Chapter 17 demonstrates structural integrity of metal SSCs based on relevant general safety and design principles. |
| Chapter 5 Reactor Core | Chapter 5 describes the fuel system design, nuclear design and thermal and hydraulic design.<br>The relevant descriptions of irradiation surveillance requirements for the RPV core shell and its radiation damage mechanism are discussed in Chapter 17. |

| PCSR Chapter | Interface |
|---|---|
| Chapter 6 Reactor Coolant System | Chapter 6 provides general design information relevant to the main components of the Reactor Coolant System.<br>The structural integrity demonstration of these components is presented in Chapter 17. |
| Chapter 7 Safety Systems | Chapter 7 provides the system description of the safety systems which includes Containment and Related Safety Systems and Engineered Safety Features.<br>The structural integrity classification and demonstration of relevant components are presented in Chapter 17. |
| Chapter 10 Auxiliary Systems | Chapter 10 provides detailed design information for the auxiliary systems and equipment.<br>The structural integrity classification and demonstration of relevant components are presented in Chapter 17. |
| Chapter 11 Steam & Power Conversion System. | Chapter 11 provides the system description of Steam and Power Conversion Systems.<br>The structural integrity classification and demonstration of relevant components are presented in Chapter 17. |
| Chapter 15 Human Factor | Chapter 15 provides the principles and methodology of human factors that shall be considered in the system and component design.<br>Chapter 17 provides the substantiation of relevant human factors principles integrated into structural integrity related equipment design. |
| Chapter 16 Civil Works & Structures | Chapter 16 is linked to relevant information for containment steel liners.<br>The structural integrity classification and demonstration of containment steel liners is presented in Chapter 17. |
| Chapter 18 External Hazards | Chapter 18 provides a list of external hazards, relevant design principles, design basis and safety assessment to identify potential risk information, and the ALARP demonstration from the external |

| PCSR Chapter | Interface |
|---|---|
| | hazards point of view.<br>Chapter 17 considers these external hazards and demonstrates that protection measures against external hazards have been implemented in the design. |
| Chapter 19 Internal Hazards | Chapter 19 provides a consequence analysis related to internal hazards caused by the failure of SSCs relevant to structural integrity，which is used for the structural integrity classification.<br>The analysis results are presented in Chapter 17. Chapter 17 also presents the list of HICs, which is an input used in bounding case selection of internal hazards. |
| Chapter 21 Reactor Chemistry | Chapter 21 presents the chemistry effects on the assessment of ageing and degradation of materials.<br>Chapter 17 provides the construction materials selection and degradation demonstration. |
| Chapter 22 Radiological Protection | Chapter 22 provides radiological protection design considerations relevant to material selection.<br>Chapter 17 provides optimum material selection for the minimisation of source term, and appropriate structure design to minimise the radioactive dose to workers and the public. |
| Chapter 23 Radioactive Waste Management | Chapter 23 provides the description of management of radioactive waste generated from operation of the UK HPR1000.<br>Chapter 17 provides optimum material selection for RCPB components which contribute to minimising radioactive waste at source. |
| Chapter 24 Decommissioning | Chapter 24 presents the design principles of material selection that facilitate decommissioning and dismantling consideration of structures.<br>Chapter 17 covers material selection and structure design to minimise waste generation and facilitate decontamination. |

| PCSR Chapter | Interface |
|---|---|
| Chapter 25 Conventional safety and Fire Safety | Chapter 25 provides the conventional health and safety risk management techniques and general prevention principles related to Structural Integrity.<br>Chapter 17 provides the design information to demonstrate that conventional health and safety risk management techniques and general prevention principles are applied in the design process. |
| Chapter 31 Operational Management | Chapter 31 presents the arrangements for operating limits and conditions.<br>Chapter 17 demonstrates the structural integrity of metal SSCs by taking into account plant operational management. |
| Chapter 33 ALARP Evaluation | ALARP analysis for structural integrity by applying the ALARP methodology is provided by Chapter 17 to support overall ALARP demonstration addressed in Chapter 33. |

## 17.3 Applicable Codes and Standards

For the UK HPR1000, the applicable codes and standards for structural integrity are selected and determined based on practices at the existing Hua-long Pressurised Reactor under construction at Fangchenggang nuclear power plant unit 3 (HPR1000 (FCG3)), based on the document *General Principles for Application of Laws, Regulations, Codes and Standards* in Reference [11] and the requirements presented in PCSR Chapter 4.4.7. The aim is to ensure the applied codes and standards comply with UK context, including applicable acts and regulations, in addition to taking into account international good practice or Relevant Good Practice (RGP).

The current identified main applicable codes and standards for the structural integrity demonstration are listed in T-17.3-1. The *Suitability Analysis of Codes and Standards in Structural Integrity* in Reference [12] has been submitted during GDA Step 3.

In GDA Step 4, the differences between current versions of these codes and latest versions (where applicable), which may have an effect on the existing design of SSCs will be considered and justified. The *Compliance Analysis of Codes and Standards in Structural Integrity Area* is developed and submitted during GDA Step 4. This comparison work will be continued as and when the relevant codes are updated.

T-17.3-1 Main Applicable Codes and Standards

| Codes and Standards | Title |
|---|---|
| RCC-M 2007 Edition, in Reference [13] | Design and Construction Rules for Mechanical Components of PWR Nuclear Islands |
| RSE-M 2010+2012 Addendum, in Reference [14] | In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands |
| IAEA SSG-30, in Reference [15] | Safety Classification of Structures, Systems and Components in Nuclear Power Plants |
| ASME 2007+ 2008 Addendum Section II, III, V, IX and XI, in Reference [16] | ASME Boiler & Pressure Vessel Code |
| R6 Revision 4, in Reference [17] | Assessment of the Integrity of Structures Containing Defects |
| TEMA Revision 9, in Reference [18] | Standards of the Tubular Exchanger Manufacturers Association |
| RCC-MR 2007, in Reference[19] | Design and Construction Rules for Mechanical Components of Nuclear Installations |
| ASME B&PVC 2017, in Reference[20] | Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC: Code for Concrete Containments |

The applicable codes and standards for respective components and structures can be found in T-17.3-2 and T-17.3-3.

T-17.3-2 Applicable Codes and Standards for Standard Class 1 Components and Relevant Supports

| Structures, Systems and Components (SSC) Type | Applicable Codes and Standards |
|---|---|
| Reactor Pressure Vessel | RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection B<br>RCC-M-2007 Section I Subsection Z |
| Main Coolant Line | |

| Structures, Systems and Components (SSC) Type | Applicable Codes and Standards |
|---|---|
| Surge Line<br>Pressuriser<br>Pump<br>Valve<br>Piping (within RCPB) | RCC-M-2007 Section II<br>RCC-M-2007 Section III<br>RCC-M-2007 Section IV<br>RCC-M-2007 Section V<br>RSE-M-2010+2012 Addendum Volume I Subsection A<br>RSE-M-2010+2012 Addendum Volume I Subsection B<br>RSE-M-2010+2012 Addendum Volume II<br>RSE-M-2010+2012 Addendum Volume III Appendix 3.1.I<br>RCC-MR-2007 Section 2 |
| Support | RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection Z<br>RCC-M-2007 Section I Subsection H<br>RCC-M-2007 Section II<br>RCC-M-2007 Section III<br>RCC-M-2007 Section IV<br>RCC-M-2007 Section V |
| Reactor Vessel Internals | RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection Z<br>RCC-M-2007 Section I Subsection G<br>RCC-M-2007 Section II<br>RCC-M-2007 Section III<br>RCC-M-2007 Section IV<br>RCC-M-2007 Section V |
| Steam Generator | ASME-2007, 2008a Addenda BPVC Section II<br>ASME-2007, 2008a Addenda BPVC Section III Division 1<br>ASME-2007, 2008a Addenda BPVC Section V<br>ASME-2007, 2008a Addenda BPVC Section IX<br>ASME-2007, 2008a Addenda BPVC Section XI<br>RSE-M-2010+2012 Addendum Volume I Subsection A<br>RSE-M-2010+2012 Addendum Volume I Subsection B<br>RSE-M-2010+2012 Addendum Volume II<br>RSE-M-2010+2012 Addendum Volume III Appendix 3.1.I |

T-17.3-3 Applicable Codes and Standards for Standard Class 2 and Standard Class 3 Components and Relevant Supports

<table>
<tr><th>Structures, Systems and Components (SSC) Type</th><th>Applicable Codes and Standards</th></tr>
<tr><td>Pressure Vessel</td><td rowspan="5">RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection Z<br>RCC-M-2007 Section I Subsection C (for class 2)<br>RCC-M-2007 Section I Subsection D (for class 3)<br>RCC-M-2007 Section I Subsection H<br>RCC-M-2007 Section II<br>RCC-M-2007 Section III<br>RCC-M-2007 Section IV<br>RCC-M-2007 Section V<br>RSE-M-2010+2012 Addendum Volume I Section A<br>RSE-M-2010+2012 Addendum Volume I Section C (for class 2 and 3)<br>RSE-M-2010+2012 Addendum Volume II<br>RSE-M-2010+2012 Addendum Volume III Appendix 3.1.II</td></tr>
<tr><td>Piping</td></tr>
<tr><td>Pump</td></tr>
<tr><td>Valve</td></tr>
<tr><td>Support</td></tr>
<tr><td>Heat Exchanger</td><td>RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection Z<br>RCC-M-2007 Section I Subsection C (for class 2)<br>RCC-M-2007 Section I Subsection D (for class 3)<br>RCC-M-2007 Section II<br>RCC-M-2007 Section III<br>RCC-M-2007 Section IV<br>RCC-M-2007 Section V<br>RSE-M-2010+2012 Addendum Volume I Section A<br>RSE-M-2010+2012 Addendum Volume I Section C (for class 2 and 3)<br>RSE-M-2010+2012 Addendum Volume II<br>RSE-M-2010+2012 Addendum Volume III Appendix 3.1.II<br>TEMA, Revision 9</td></tr>
<tr><td>Steel Liner</td><td>ASME B&PVC-2017 Section III, Division 2, Subsection CC: Code for Concrete Containments</td></tr>
</table>

According to information available from codes in the above tables, for standard Class 1, Class 2 and Class 3 components and structures within structural integrity areas of the UK HPR1000, the design, manufacture, manufacturing inspection and testing activities are predominantly based on RCC-M in Reference [13], and the Pre-Service Inspection (PSI) and ISI shall comply with RSE-M in Reference [14].

For HIC components, in addition to the structural reliability being justified by compliance with RCC-M and RSE-M:

a) The R6 procedure is applied to defect tolerance assessments.

b) The relevant ASTM code is applied for additional fracture toughness testing to characterise the fracture toughness of the material in order to inform the defect tolerance assessment, which is presented in *Supplementary Fracture Toughness Test Requirements of Materials for HIC Components.*

c) The inspection system, including procedure, equipment and personnel, will be qualified according to the European Network for Inspection & Qualification (ENIQ) for qualification of NDT.

The details of exceeding the requirements stipulated in the above codes are identified in Sub-chapter 17.6.1.

For the SG, the design and manufacture complies with ASME III in Reference [16] with additional enhancements drawn from RCC-M, with the PSI and ISI being based on RSE-M. A review of the codes and standards for the SG and the rationale are detailed in *High Level ALARP Assessment for SG Code* in Reference [21].

To mitigate the potential risk due to the use of different codes, the following main supplementary requirements have been added, the details of which can be found in Reference [21]:

a) For the design of the SG pressure boundary, the design rules of mechanical analysis are based on the ASME Code, Section III and Subsection NB are supplemented with the some additional requirements from the RCC-M Code Subsection B rules.

b) The main forging requirements are be qualified according to RCC-M M140.

c) The requirements for the mean dimension of the throat (“minimum leak path”) are defined according to RCCC-M S 3800.

d) The hydraulic expansion gap leak tightness and mechanical strength tests are based on to RCC-M F4400.

e) The production weld test coupons requirements are based on RCC-M S7800.

f) Supplementary inspection requirements that cannot be found in ASME section III and XI are based on RSE-M.

In Step 3 of GDA, ONR has raised a draft RO on the SG code provisions and mitigation of relevant risks. CGN has committed to address systematic risk identification and provisions for mitigation.

## 17.4 Safety Functional Requirements

Satisfying the safety design basis for each SSC and the relevant structural integrity safety functional requirements are essential to ensuring the plant risk from structural failures is both tolerable and ALARP. In general, the safety functional requirements are derived from the performance and safety design basis of nuclear-safety systems. For example, the performance and safety design basis for the Reactor Coolant System are presented in PCSR Chapter 6, which provides the basis for identifying the specific safety functional requirements of respective SSCs. The specific safety functional requirements originate from the potential radiological consequences of failure and the requirements to meet the functional requirement of nuclear facility throughout design lifetime.

Each safety functional requirement for SSCs in the UK HPR1000 establishes a specific role to maintain nuclear safety under all design basis conditions. The safety functional requirements for significant SSCs are presented in the scheme description of reactor components and reactor main loop equipment in References [22] and [23] and the relevant CSRs, to specify the required safety functional performance in each case.

Postulated failure modes (during normal operations or in response to faulted conditions or hazards) which result in the loss of a safety functional requirement, lead to the identification of structural classification that will be commensurate with the consequences of gross failure. This process of structural integrity classification is presented in sub-chapter 17.5 and Reference [5].

## 17.5 Structural Integrity Classification

### 17.5.1 Structural Integrity Classification Process

In general, the reliability and robust quality of design and construction of components or structures is ensured through meeting the requirements of codes and standards which are commensurate with the safety class of the components or structures. However, the safety classification methodology is not fully comprehensive and so lack of complete criteria means component or structure structural integrity requirements cannot be fully determined. Hence, by taking accepted UK practice into account, the supplementary classification methodology for determining structural integrity requirements for metal SSCs has been established in Reference [5].

The method of safety categorisation and classification discussed in PCSR Chapter 4.4.5 is based on IAEA SSG 30 in Reference [15], and UK expectations are also taken into account. The UK HPR1000 follows a systematic method for classifying components or structures and selecting applicable engineering design codes and standards in line with their safety class. The structural integrity classification starts with the results of overall safety classification of plant, beginning with standard class 1, 2 and 3 components determined. The structural integrity classification assigned to these components is based on the severity and tolerance of the direct and indirect consequences of postulated gross failure.

The structural integrity classification method, process, definition of standards and corresponding relationship between safety class and design standard class are presented in detail in Reference [5].

Following the assignment of a standard class to each component, for components where the severity of consequences (direct or indirect) of failure are high (no effective protection is provided) then an enhanced designation of HIC is required, which sits above standard class 1. A further review is then performed to develop a list of the standard class 1 components for which postulated gross failure would be intolerable. These components are classified as HICs. Summary information on each of the classifications is provided below.

a) High Integrity Component (HIC)

HIC classification is assigned to the components and structures whose failure is intolerable and for which no physical protection is provided or protection provision is not reasonably practicable.

b) Structural Integrity Class 1 (SIC-1)

SIC-1 is assigned to the components and structures whose failure could cause limited core damage. There should be at least one line of protection with redundancy. The integrity claims for such components will be predominantly based on compliance with recognised nuclear design code requirements.

c) Structural Integrity Class 2 and 3 (SIC-2, SIC-3)

SIC-2 or SIC-3 are assigned to the components and structures whose failure does not result in core damage. There should be at least two lines of protection with diversity. The integrity claims for such components will be predominantly based on compliance with appropriate design codes and standards.

The criteria for each structural integrity class of metal components and structures are presented in T-17.5-1. The relationship among standard class determined by safety classification, structural integrity class, protection measures and potential failure consequences are also described.

T-17.5-1 Structural Integrity Classes

| **Standard Class Determined by Safety Classification** | **Protection Measures** | **Protected Potential Failure Consequence of Components** | **Structural Integrity Class** |
|---|---|---|---|
| Standard Class 1 | Unable to provide effective protection | 1.Severe core melt<br>2. Offsite large radioactive release | HIC [a)] |
| | At least one line of protection meeting | 1. Limited core damage<br>2. Offsite minor radioactive release | SIC-1 |

| Standard Class Determined by Safety Classification | Protection Measures | Protected Potential Failure Consequence of Components | Structural Integrity Class |
|---|---|---|---|
| | Single Failure Criterion (SFC) in analysis of Fault and Protection Schedule | 3.Significant mass and energy release within nuclear island | |
| Standard Class 2 | Two diverse lines of protection together meeting SFC in Fault and Protection Schedule | 1. No core damage<br>2. Offsite minor radioactive release<br>3. Limited contamination within nuclear island | SIC-2 |
| Standard Class 3 | Two diverse line of protection together meeting SFC in Fault and Protection Schedule | 1. No core damage<br>2. No significant radioactive release<br>3. Limited contamination within nuclear island | SIC-3 |

Note a): In addition to standard class 1 components, the determination of HIC components could be derived from standard class 2 or 3, taking into consideration the design differences between FCG3 practice and UK context and previous GDA experience.

### 17.5.2 Results of Structural Integrity Classification

According to the structural integrity classification method and process in Reference [5], the candidate HIC components are firstly identified based on the following principles:

1) A high-level review to identify a short list of candidate HIC for which postulated gross failure could conceivably lead to intolerable consequences, through direct or indirect means.

2) Acknowledgment design differences between FCG3 practice and UK context, which cover standard class 1, 2 and 3.

3) Comparison to previous GDA feedback to finalise the candidate HIC list.

When the candidate HIC list is determined, a more detailed evaluation of the consequence analysis and design modification demonstration is performed to justify whether these components can be classified as HIC or non-HIC.

After rigorous review of the direct and indirect consequence analysis, the candidate HIC components are identified as follows:

1) RPV
2) PZR
3) SG
4) MCL
5) RCP casing
6) RCP flywheel
7) MSL
8) Surge line
9) RVI
10) SG blowdown line
11) ACC

According to current progress and analysis results, the following components are HIC for the UK HPR1000. And the relevant evidence will be provided in Step 4.

1) RPV
2) PZR
3) SG
4) MCL
5) RCP casing
6) RCP Flywheel
7) MSL

The following candidate HIC components are identified as non-HIC.

1) Surge line
2) ACC
3) RVI
4) SG Blowdown Line

The HIC boundary of each HIC component will be indicated in the each CSR.

The relevant classification processes, results and supporting documents of the components are described or quoted in Reference [2], *Equipment Structural Integrity Classification List*. For SSCs within the scope of Sub-chapter 17.2.2, the structural integrity class will be covered and added into Reference [2] as well in the next stage of Step 4.

## 17.6 Component Safety Cases

This sub-chapter presents the methodology for developing structural integrity safety cases for each class of metal components and structures, which are presented below.

### 17.6.1 High Integrity Component

Structural integrity safety cases for HIC components will be developed in accordance with the methodology presented in Reference [4], which has been developed to be consistent with the recommendations of TAGSI.

Compared with SIC-1/ SIC-2/ SIC-3 components, HIC components require a more rigorous integrity demonstration to ensure that the component remains as defect free as possible and is defect tolerant.

For each HIC component, justifications will be presented in the relevant CSR according to the following safety claims and the supporting arguments. More detailed information of CAE can be found in Reference [4]. The sub-claims for HICs to support the chapter claim are as follows:

a) Sub-Claim 1: High quality is achieved through good design and manufacture and functional testing.

b) Sub-Claim 2: The avoidance of fracture demonstrates that the High Integrity Components are tolerant of defects to through-life degradation.

c) Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

The above sub-claims implicitly include conceptual defence-in-depth, which is derived from TAGSI concepts. The relevant information is presented in Reference [4].

Within the CSRs, relevant arguments will be produced from the above three Sub-claims to substantiate structural integrity through the provision of robust evidence. Further information on the development of the safety arguments and evidence is outlined in the sub-chapters below.

#### 17.6.1.1 Good Design and Manufacture and Functional Testing

In order to support Sub-Claim 1, there are 8 arguments identified for demonstrating the highest reliability of components as follows. Further information is presented in Reference [4].

a) Design will be well developed in accordance with a recognised and appropriate international code and taking into account relevant OPEX.

b) Design analysis will confirm structural integrity based on conservative assessment.

c) Components are manufactured through judicious material selection.

d) High manufacturing quality will be achieved in line with proven codes supported by OPEX.

e) High reliability (qualified) manufacturing Non-Destructive Testing (NDT) will be carried out to provide assurance of no structural defects of concern.

f) Stringent quality assurance controls are implemented to ensure compliance with the design and construction specifications.

g) Good operation and maintenance will be considered to demonstrate that the plant will be properly operated and maintained.

h) Hydrostatic pressure tests will verify the pressure boundary integrity at start of life.

17.6.1.1.1 Design

The relevant codes and standards for HIC components are presented in Sub-chapter 17.3. To achieve a high quality of build, HIC components comply with the requirements of relevant, mature and widely-used nuclear codes and standards as a solid foundation, and additional measures, supplemental to the requirements of codes, are identified and implemented as follows:

a) The appropriate supplementary fracture toughness test for HIC component material will be determined and implemented at the stage of product manufacture.

b) Inspection qualification according to ENIQ methodology, as described in Reference [24], will be used to achieve the reliability of objective-based manufacturing NDT.

c) Defect tolerance is to be substantiated by defect tolerance assessment.

d) Independent third party inspection.

During the design stage, in addition to complying with the recognised and appropriate international codes, the potential in-service ageing and degradation of the components and previous OPEX are considered, and novel design is avoided or adequately justified. Therefore, the relevant evidence to support this argument will be provided in CSRs. the main elements are presented below:

a) Extensive experience in design and construction of Pressurised Water Reactors (PWR).

b) Proven design codes and standards are applied.

c) Known degradation and OPEX are considered during design stage.

d) Novel design is avoided or adequately justified.

e) Where produced in multiple parts, HIC component sections will be designed to be as large as practicable with respect to manufacturing capability, in order to reduce the number of welds present in the pressure boundary.

17.6.1.1.2 Design analysis

HIC components will be assessed against code specified requirements for the design life of the plant. In order to demonstrate that HIC components are designed in compliance with allowable stress limits, fatigue usage factors and fast fracture limits as specified in the design codes, the following types of failure modes will be assessed:

a) Excessive deformation and plastic instability.

b) Elastic or elastoplastic instability (buckling).

c) Progressive deformation induced by repeated loads.

d) Fatigue (progressive cracking).

e) Fast fracture.

Finite element models for specific components will be generated and representative boundary conditions and loads imposed according to the components analysed. The loading conditions are described in Sub-chapter 17.7, and the appropriate load combinations are to be applied during structural integrity assessments. The computer software is then used to calculate the stress distribution. According to the distribution of the stress field, the analytical path will be established for areas with higher levels of stress or areas in which the designers may wish to focus. The stress on the path is linearised for the assessment of membrane stress, bending stress, total stress, etc. For each category condition, a limit is imposed on the stress intensities, corresponding to each of these stress categories.

Fatigue analysis is used to demonstrate that there is no fatigue failure during the design life of components. According to the results of the stress analysis, the linear stress values of the analytical paths under different conditions are obtained (the positions at the ends of the path), and the fatigue analysis for the region is accomplished.

The fast fracture mechanical analyses are based on design code present acceptable provisions for fast fracture prevention. Fast fracture damage is considered to include brittle fracture and ductile tearing. Except in faulted condition, tearing analysis is not permissible for the UK HPR1000, even if the design code allows. Allowable pressure temperature (P-T) curves may be established according to these provisions, and used for operating and test conditions. The loading conditions are described in Sub-chapter

17.7, and appropriate load combinations are also presented.

Transients of pressure and temperature are both considered in fracture analysis. The stress intensity factor of each transient is calculated, considering thermal and mechanical stresses, while temperature is considered for judging impact on toughness of material. Fast fracture analyses cover second, third, fourth category conditions and test conditions.

Conservative methods will be used for the stress analysis, for example in the parameter values for loading combination, structural dimensions, material property and cladding.

Detailed approaches to code-based design analysis for individual locations and the associated finite element models are included in the stress analysis reports. The outline of the analysis approach and their results will be summarised in the general reports of mechanical analysis, which support the CSRs for HIC components.

17.6.1.1.3 Material selection

Material selection for SSCs is one of the most significant factors contributing to the safety and economy of the nuclear power plant. For this reason, special attention is paid to the material selection for SSCs of the UK HPR1000 at the design stage, in order to guarantee that their Safety, Security or Environmental (SSoE) duties can be carried out with high reliability throughout the design life of the plant.

Material selection is a multi-discipline area, considering all the disciplines from related topic areas (including Structural Integrity (SI), Reactor Chemistry (RC), Mechanical Engineering (ME), Radiological Protection (RP), Radwaste, Decommissioning, Environmental, Civil Engineering and so on). SI and RC are the most important disciplines contributing to material selection which is conducted by a cross-discipline work team with qualified members from each area.

Material selection for the UK HPR1000 components is performed according to CGN’s experience from construction and operation of Chinese Pressurised Reactor (CPR1000) and HPR1000 (FCG3), as well as the UK context requirements and international PWR OPEX. A *Material Selection Methodology* report has been prepared for the UK HPR1000 in Reference [25], which provides guidance for the material selection of the UK HPR1000, to guarantee that material selection for all SSCs is carried out in a consistent manner.

The selected materials are proven materials used successfully in existing PWRs, and use of novel material is prohibited. Mechanical property tests and chemical analysis will be performed in accordance with the requirements of the related material procurement specifications, to verify the properties and chemical composition of the materials.

Furthermore, due consideration of international OPEX from other PWRs will be used to reduce risk associated with component material selection:

a) Homogeneity of large forgings for HIC components is taken into account, rational requirements on steel making technology, chemical analysis and mechanical testing are specified for large forgings associated with the UK HPR1000. Sufficient representative test specimens from different locations, depths and orientations are required to verify the homogeneous nature of large forgings to avoid abnormal macrosegregation.

b) Hydrogen content in ferrite steel (both in base metal and weld metal) is strictly limited, in combination with the appropriate requirements on post heating and post weld heat treatment, to mitigate the risk of hydrogen-induced cracking.

c) Stringent requirements above the code minimum requirements are specified on impurity elements such as S, P, As, Sn and Sb. Content of these elements is strictly limited to mitigate the risk of hot cracking and reheat cracking. Other impurity elements (such as C, H, Cu, V, Nb, Al and Ti) are strictly limited to a low level to satisfy the code minimum requirements.

d) Non-metal inclusion requirements beyond code scope are specified to guarantee the quality of the forgings.

Detailed information on materials used for the HIC components is to be presented in the related CSRs.

In GDA Step 3, material selection demonstration is mainly focused on HIC components, which play a significant role in safety and environmental duties throughout the design life time of the plant. A number of *Material Selection Reports* have been established for the RPV, SG, PZR and MCL, in References [26]-[29]. In Step 4 of GDA, the material selection reports for the MSL and RCP will be issued.

In addition, the representative SSCs for non-HICs (Main Feedwater Line) will be sampled for judicious material selection demonstration. Material Selection Reports for the Main Feedwater Line System Piping will be established and be submitted during GDA Step 4.

The potential in-service aging and degradation of materials affecting the design life of the HIC components have been considered in the design stage. Appropriate materials selection ensures compatibility with the operating environment against potential risks associated with identified degradation mechanisms. These are comprised of Stress Corrosion Cracking (SCC), irradiation embrittlement of ferritic steel in the core area, thermal aging of the cast austenitic stainless steel and intergranular corrosion of austenitic stainless steel in contact with the reactor coolant. The following measures are considered below:

a) For some reactor coolant pressure boundary (RCPB) components, such as the Control Rod Drive Mechanism (CRDM) penetrations and SG heat transfer tube, NC30Fe is selected as the preferred material over NC15Fe, mainly due to its relatively high chromium content and superior resistance to SCC.

b) For the ferritic steel in the RPV core area, the effect of irradiation embrittlement is considered during material selection. Therefore, chemical composition of sensitive elements such as Cu, Ni and P are stringently controlled, and high initial toughness is required to mitigate the effects of irradiation embrittlement degradation during the whole lifetime of the plant.

c) Austenitic stainless steel forgings are used for the MCL instead of castings which are more susceptible to embrittlement due to thermal aging during operation. The grain size number of the forgings, determined in accordance with RCC-M MC1000, is required to be greater than 2 (1 in RCC-M M3321) for microstructure control and to enable effective Ultrasonic Testing (UT).

d) With regards to the intergranular corrosion of austenitic stainless steel, several measures have been taken into account. Firstly, austenitic stainless steels with low carbon content (less than or equal to 0.035% in most cases) are selected. Secondly, all the austenitic stainless steels are delivered in a solid solution heat treatment state and all the processes that may heat the austenitic stainless steel to 425℃ above are practicably minimised to avoid sensitisation. However, austenitic stainless steel may be inevitably subjected to the sensitisation temperature range during welding or final stress relieving heat treatment of the components, and so subsequent intergranular corrosion testing is required in accordance with RCC-M MC1300. In addition, the water chemistry of the reactor coolant system is to be monitored and maintained within specific limits.

Ageing and degradation cannot be completely eliminated for a 60 year period, in-service inspection and monitoring will be in place for detecting and monitoring of degradation before structural integrity is compromised. In addition, RPV core region specimens (low alloy steel base metal and weld metal) are to undergo periodic irradiation surveillance assessment to ensure that an adequate safety margin is maintained.

Detailed information is presented in a series of *Ageing and Degradation Reports* established for the RPV, SG, PZR and MCL, in References [30]-[33], which are used to support the relevant material selection.

In step 4 of GDA, the ageing and degradation reports of other HICs (MSL and RCP) will be issued. The ageing and degradation justification for the Main Feedwater Line will be included in those material selection reports.

At the end of GDA Step 4, a summary report of material selections will collect the outcome of all the material selection reports and ageing and degradation reports and will be issued to support the ALARP demonstration of material selection in the SI area.

17.6.1.1.4 High quality of Manufacture

Based on the code requirements, the manufacture of HIC components will adopt proven techniques and approved procedures. Relevant OPEX and good practice will be considered and integrated into the appropriate evidence.

The manufacturing is to be carried out in accordance with approved procedures. Qualified and proven melting technology and heat treatment procedures are required for forging manufacture, archive material coupons will be set for base metal. Qualified welding procedures will be developed in line with good practice and qualified welders will be used to perform the production welds. Production test coupons are required for significant welds to ensure conformity with the requirements determined by the welding procedure qualification test. Suitable requirements on surface treatments and other manufacture sequences of HIC components will also be well specified. Objective based manufacturing NDT defined in Sub-section 17.6.1.1.5 of this report will be performed on the HIC components. In addition, the independent third-party inspection will also be carried out by an approved notified body. The relevant information is presented in Sub-section 17.6.1.1.6.

Furthermore, suppliers of HICs will be selected based on good practice and relevant engineering experience. The production workshop will be technically qualified to evaluate the capacity and technical resources for carrying out required manufacturing. Deviations and repairs from design intent will be recorded and justified.

Through use of the above measures, the occurrence of defects that could impact the integrity of components and structures will be minimised to guarantee the production of high quality products.

Evidence related to the aspects above will be provided to demonstrate compliance with the appropriate manufacturing requirements of RCC-M or ASME codes. Detailed information can be found in the related CSRs and their supporting documents.

17.6.1.1.5 Manufacturing NDT

For HIC components, NDT requirements prescribed in the design codes will be performed to confirm the general quality of manufacture. Objective-based high reliability (qualified) manufacturing NDT will be applied to HIC components at the end of the manufacture, which provide high confidence in establishing the absence of structural defects of concern.

The high reliability demonstration strategy and approach of objective-based manufacturing NDT for HIC components is presented in Reference [34]. It establishes several measures which will be taken to achieve high reliability of objective-based manufacturing NDT for HIC components, which are as follows:

a) Providing the inspection access assurance and applying the concept of ‘design for inspectability’ to maximise the effectiveness of NDT. During the design and manufacturing stage, access and inspectability for the implementation of manufacturing NDT and PSI/ISI should be provided for HIC components. This is shown in *Access and Inspectability Assessment* in Reference [35] and *Application of Weld Ranking Procedure* in Reference [36]. Specially, the design optimisation (if needed) for NDT should be based on plant risk and reduce risk to ALARP.

b) Applying objective-based NDT methods and techniques that are suitable for reliable detection and rejection of planar defects of concern.

c) Applying well-established NDT techniques that are based on sound physical principles.

d) Qualifying the objective-based end-of-manufacturing NDT system (including NDT procedure, equipment and personnel) according to ENIQ methodology, to demonstrate the NDT system can deliver the required performance and reliably detect planar defects equal to or larger than the Qualified Examination Defect Size (QEDS), which is presented in Sub-chapter 17.6.1.2.

e) Adopting the redundant and diverse measures during implementation of manufacturing NDT to assure further reliability of NDT.

f) Appropriate repeat inspection will be considered for the manual UT of parts of HIC welds/forgings during manufacturing.

17.6.1.1.6 Quality Assurance

The general requirements of quality assurance for the UK HPR1000 are presented in PCSR Chapter 20. For control and surveillance of the design, manufacture, inspection, and/or testing of HICs, the relevant quality assurance requirements will be applied to ensure compliance with the design and construction specification. The *Quality Assurance Grading Method and associated Management Requirements* are presented in Reference [37], to determine the quality assurance classification of each SSC and specify what measures and controls should be applied, commensurate with their QA class.

For HIC components, the highest QA class and relevant requirements will be applied through design and construction. Associated evidence will be provided, which includes specific quality assurance management requirements, non-conformance activities control requirements, approved records of quality history and QA audit. For example, the End of Manufacturing Report (EOMR) will contain:

a) Conformity declaration.

b) Quality plan list.

c) Non-conformance reports.

d) Specifications and certified material testing report of base metal and weld metal.

e) Welder qualification report.

f) Welding procedures qualification report and welding data package.

g) Procedure of surface treatment.

h) Description of the NDT.

i) Dimensioning check.

j) Hydro test result.

k) Painting details.

l) Cleaning and package information.

m) Detailed manufacturing records, manufacturing inspection records and acceptance record.

For HIC components, independent inspections in manufacture and installation processes will be required according to the requirements of independent third-party inspection based on the previous GDA practice, and the implementation will be performed during licencing phase.

The inspection qualification is required by the regulatory authority in UK to confirm that processes and procedures are being followed. The detailed requirements for inspection qualification are presented in *Inspection Qualification Strategy for High Integrity Component*, established in Reference [38].

The above relevant information is presented in individual CSRs or their supporting documents.

17.6.1.1.7 Operation and Maintenance

According to the requirements defined in PCSR Chapter 4.4.6.2.3, the design should be such that activities for examination, maintenance, inspection and testing are carried out to maintain the condition of SSCs important to safety, to ensure they perform essential safety functions and satisfy the reliability requirements.

Under the structural integrity area, the following activities will be implemented in order to demonstrate the components will be properly operated and maintained:

a) Operational pressure-temperature limits are clearly determined based on the fast fracture analysis method defined in RCC-M appendix ZG and will take into account material degradation.

b) The chemistry parameter limits and conditions of operation are to be provided in order to ensure good compatibility between RCPB materials and reactor coolant.

c) Periodic examinations ensure the integrity and leak tightness of components.

d) ISI will be performed periodically to detect structural defects of concern before they compromise structural integrity during service.

e) In-service maintenance will be carefully controlled through a formal procedure. To ensure that operators can safely operate and implement maintenance activities during operation, the operating and maintenance requirements will be presented in CSRs or design specifications.

The above relevant information is presented in individual CSRs or their supporting documents.

17.6.1.1.8 Hydrostatic Pressure Testing

Pressure testing is typically conducted on pressure vessels, pipework and systems after completion of manufacture and before service according to code requirements, to confirm integrity at start of life through verifying strength and leak-tightness of relevant components. Where relevant, this will be considered and presented in the CSRs.

Detailed records are attainable to allow review at any time during subsequent operation. The test procedures, test records and results will be considered as evidence to demonstrate compliance with appropriate requirements of design and construction codes. The relevant information is also presented in the CSRs.

17.6.1.2 Avoidance of Fracture

The Avoidance of Fracture is intended to demonstrate that the HIC components are tolerant of defects during life period. In order to support Sub-Claim 2, there is one argument identified for demonstrating the highest reliability of components:

The integration of Defect Tolerance Assessment (DTA), high reliability NDT and lower bound material properties support the avoidance of fracture demonstration.

To demonstrate avoidance of fracture, these three contributing elements (DTA, high reliability NDT, and lower bound material properties) are interactive and integral to contribute to Sub-Claim 2 of high structural reliability components, which is based on a ‘defence in depth’ approach of:

a) Absence of crack-like defects at the end of the manufacturing process - confirmed by manufacturing inspections.

b) Material toughness offering good resistance to initiation and propagation of crack-like defects - underpinned by minimum material toughness requirements.

c) Acknowledging that defects may be present and demonstrating tolerance to them - demonstrated by fracture assessments to establish tolerance to all defects smaller than a QEDS by a size margin of two.

The approach of avoidance of fracture demonstration is described in the following six steps. Further information of Avoidance of Fracture is presented in Reference [4].

**Step 1 Identifying Limiting Locations**

*Weld Ranking Procedure* in Reference [39] is introduced to identify the limiting locations (weld locations and parent metal locations) for DTA and Technical Justification (TJ) of each HIC considering applied stresses, stress intensity factors, fatigue usage factors, and the inspectability. Furthermore, it is apparent that there are significant overlaps in the inspection requirements for these limiting HIC locations among different HIC components, the *Re-grouping of Application of Weld Ranking Procedure* in Reference [40] is then introduced to further identify the typical (most onerous) limiting locations for high reliability NDT demonstration.

Based on the *Application of Weld Ranking Procedure* in Reference [36], the limiting locations are identified to carry out DTA, and the typical limiting locations further identified by *Re-grouping of Application of Weld Ranking Procedure* in Reference [40] are required to carry out High Reliability NDT demonstration.

**Step 2 Specify and Verify the Material Properties**

a) Specify the Material Property.

   Conservative (lower bound) values of fracture toughness used in DTA are expected. Material properties which are used for the DTAs are taken from RCC-M appendix ZI&ZG and RSE-M appendix 5. Generally, the lower bound of these material properties is used, and material degradations are considered. Verify the Material Properties.

b) Verify the Material Properties

   Supplementary fracture toughness tests will be carried out to underpin the fracture toughness values assumed in the DTA in the licence stage, and the toughness test requirements are specified in the document Supplementary Fracture Toughness Test Requirements of Materials for HIC Components.

**Step 3 Determine the Limiting Defect Size Using R6 Approach**

DTAs are performed, which go beyond the design code provisions for ensuring defect tolerance, to provide a high level of reliability. The detail of this methodology is described in Reference [41], *Defect Tolerance Assessment Methodology for HIC Components*. The R6 procedure is in accordance with elastic-plastic fracture mechanics, will be used to assess postulated defects with in-service crack growth, even though significant defects are not expected (either from manufacture or from in-service mechanisms).

For each assessed location/item of a component, a Defect Size Margin (DSM) will be calculated in formula (1):

$$DSM = \frac{ELLDS}{QEDS + LFCG} \tag{1}$$

Where ELLDS is the End of Life Limiting Defect Size, QEDS is the Qualified Examination Defect Size (serving as initial crack size) and LFCG is the Lifetime Fatigue Crack Growth.

DSM is provided as a pragmatic approach to demonstrating the appropriate margin, and a value of at least 2 will be targeted. The DTA assessment should be conservative and provide sufficient margin. A simple but conservative method will be selected at the beginning. It should be noted that refinement of the DTA is possible if the previous assessment of DSM is not met due to over conservatism. However, refinement should not extend to undermine the expectation of using a conservative fracture assessment.

Defect characterisation is the term given to the process of modelling existing or postulated flaws as geometrically simpler ones more amenable to analysis. For defect tolerance assessment in GDA, only surface planar flaws are assumed. The flaw depth, width and shapes cause different stress intensity factor forms. The characteristics of defects are determined by engineering judgment. In order to demonstrate the robustness on this, parametric studies of different defect shape ratios, orientations and positions are considered.

Loading conditions are described in Sub-chapter 17.7. For normal operations, assessment is based upon bounding material properties, upper bound crack growth and conservative loadings. For faulted condition assessments, more representative measures are considered to provide a structural reliability which is commensurate with the initiating fault frequency.

The DTA assessment procedure includes two key aspects, the ELLDS and LFCG. ELLDS is evaluated by means of a Failure Assessment Diagram. This parameter varies with transients. Using stresses and stress intensity factors, maximum and minimum pairs for each transient are calculated. Then based on Paris' law, the LFCG is calculated. The sum of QEDS and LFCG provides the final defect size.

DTAs of limiting locations are performed and will be completed in the GDA stage.

**Step 4 Qualify the Manufacturing Inspections Using ENIQ Methodology**

Objective-based high reliability (qualified) manufacturing NDT will be applied to HIC components at the end of the manufacture. NDT system (including NDT procedure, equipment and personnel) according to ENIQ methodology are demonstrated that the NDT system can deliver the required performance and reliably detect planar defects equal to or larger than the QEDS.

High reliability (qualified) NDT is specified in combination with DTA, to support the avoidance of fracture demonstration. Inspection qualification will be performed according to ENIQ methodology to achieve the high reliability of objective-based manufacturing NDT for HIC welds, and to establish the inspection capability of reliable detection for planar defects equal to or larger than the Qualified Examination Defect Size (QEDS), which is established using DTA with a defect size margin (DSM) of at least 2. The postulated target defects established for inspection qualification will be advised by welding engineers and metallurgists, depending on the engineering experience and professional technical judgement, and the defect information (type, location, orientation and morphology) will be reviewed by an internal expert panel.

The qualification strategy and approach of objective-based manufacturing NDT for HIC components is presented in Reference [38]. In order to reduce the number of HIC areas to be assessed for NDT within GDA, two procedures named *Weld Ranking Procedure* in Reference [39] and *Re-grouping of Application of Weld Ranking Procedure* in Reference [40] have been developed for selecting the typical limiting HIC welds and limiting non-welded regions.

During GDA, partial inspection qualification will be limited to objective-based manufacturing NDT for the typical limiting welds. Manufacturing NDT for the limiting non-weld regions will not be subjected to inspection qualification, but capability statements will be produced for the following reasons:

a) The manufacturing process of base metal is mature, and will have a lower incidence rate of manufacturing defects compared to the welds.

b) The possible nature, position and orientation of manufacturing defects in base metal are well understood and readily detectable by manufacturing NDT.

c) The base metal will be subject to less residual stress compared to the welds.

d) The base metal has a higher fracture toughness compared to the welds.

The partial qualification activities for the typical limiting HIC welds and the capability statements for the limiting non-welded regions will be performed to provide confidence that these inspection requirements can be achieved. The partial qualification activities for the typical limiting HIC welds cover GDA TJ and independent review. The purpose of independent review by a quasi or formal qualification body is to justify whether the inspection is likely to be capable of successful qualification, when fully developed. The GDA TJ report, which will be based on ENIQ Recommended Practice 2 in Reference [42], is a reduced version of the Technical Justification Report and includes a summary of relevant input information, the overview of the proposed objective-based manufacturing NDT system and physical reasoning.

During step 3 of GDA, the document *GDA Technical Justification for RPV Flange-nozzle Shell to Core Shell Weld*, Reference [43], has been produced in the context of GDA to describe in some detail the approach composed of physical arguments and simulation-based analysis. The GDA TJ sets out arguments which show that the application of ultrasonic testing for the weld will ensure that any such defects of sizes of concern or greater would be reliably detected. After the completion of GDA TJs, the independent review will be performed by the Qualification Body.

During step 4 of GDA, the other typical limiting HIC welds of the RPV/SG/MCL will be produced for GDA technical justification reports and will be reviewed by the Qualification Body.

**Step 5 Reconciliation of DTA and TJ**

In the GDA stage of the UK HPR1000, DTA and GDA TJ are normally carried out in parallel, material properties taken from codes are used as the input of DTA, and direct testing of material property (fracture toughness) need not to be carried out in GDA stage and will be carried out in the licensing stage (post GDA).

Adequate margin is normally demonstrated through a reconciliation process where the inputs (namely DTA results, inspection capability and material properties) are compared.

The limiting QEDS derived by DTA and the QEDS qualified by the GDA TJ are compared to judge whether the reconciliation is achieved:

a) If the limiting QEDS is larger than or equal to the QEDS and the DSM reach the expected value of 2, the reconciliation process will not be initiated.

b) If the limiting QEDS is smaller than the QEDS, the reconciliation will be initiated. In this case, further actions (such as re-assessment of DTA) will be taken to justify the reconciliation process until the Limiting QEDS is larger than or equal to the QEDS.

The details of the reconciliation process will be described in *Avoidance of Fracture Reconciliation Strategy*.

**Step 6 Overall avoidance of fracture demonstration**

For each HIC component, a relevant CSR report will describe the overall avoidance of fracture demonstration by bringing together the outputs from fracture mechanical assessment, inspection qualification QEDS during manufacturing and applied low bound material properties and their potential reconciliations, to show that structurally significant defects would be reliably detected using suitably redundant, diverse and, where appropriate, qualified inspection techniques. Even if the defect is present, it is possible to demonstrate tolerance to through life degradation. This will combine the DTA with the TJ to provide a strong leg to the safety case.

In Step 3, one limiting weld is chosen to perform the avoidance of fracture demonstration. The rest of the limiting welds and non-weld regions will undergo avoidance of fracture demonstration in step 4.

17.6.1.3 Forewarning of Failure

In order to support Sub-Claim 3, there are 3 arguments identified for demonstrating the highest reliability of components by establishing that effective systems are in place to provide forewarning of failure (which includes ISI, irradiation surveillance, monitoring of plant transients and leak detection). These are as follows:

a) Suitable and effective in-service inspections are implemented to provide forewarning of failure.

b) Diverse systems are provided to monitor the plant transients, and detect, locate and monitor reactor coolant leakage.

c) Material irradiation surveillance is provided to forewarn the degradation level, if any, in service stage.

17.6.1.3.1 In-Service Inspection

Early indication of degradation should be provided to prompt corrective actions before gross failure occurs. Appropriately-defined periodic ISI will reveal degradation in good time before structural integrity is compromised to unacceptable levels, and to confirm the absence of unanticipated degradation. ISI is an effective method and a particularly important provision to forewarn of failure. ISI is used to confirm the absence of defects that could eventually lead to failure, where the tolerable defect size is combined with a conservative DTA of in-service defect growth throughout the inspection interval.

PSI/ISI requirements will be determined based on the characteristics of each HIC component and inspection area, RSE-M code, OPEX and UK requirements (Ultrasonic Testing is preferred). PSI will be performed prior to servicing using the same NDT techniques and equipment used for future ISI. The purpose of PSI is to establish reference material or a “zero point” for future ISI and confirm the absence of structural defects of concern before entering service for HIC components.

All HIC components will be included in PSI/ISI programme and PSI/ISI techniques for HIC components will be qualified according to ENIQ methodology as described in Reference [24]. The document *Outline of PSI and ISI requirements for the UK HPR1000* in Reference [44] (PSI/ISI programme is out of GDA scope) has been produced for a high level description of the general principle that will be used for the PSI/ISI of the UK HPR1000 and the inspection requirements for HIC components.

17.6.1.3.2 Monitoring Plant Transients and Leak Detection

The descriptions of safe plant operating limits are presented in PCSR Chapter 31. Diverse systems are provided to monitor the plant transients and detect, locate and monitor reactor coolant leakage.

The Fatigue Monitoring System (KIF [FMS]) monitors thermal fatigue loads of fatigue relevant components. This monitoring contributes to long-term reliability and safety assurance in a nuclear power plant.

The KIF [FMS] is mainly used to monitor the main pipeline, where there are cold and hot water intersections, including the primary loop line next to the safety injection nozzle, RCV charging nozzle, RIS/RHR nozzle, surge line, RIS safety injection line, primary loop transition drain line, pressurizer Spray line, main feed-water line, downstream pipeline of RIS/RHR heat exchanger, etc.

The KIF [FMS] consists of a data acquisition subsystem and a fatigue analysis subsystem.

a) Data Acquisition Subsystem

The data acquisition subsystem uses specially arranged sensors to measure the temperature field on the outer wall of the pipe and the fluid data inside the pipeline. The acquired data is sent to the fatigue analysis subsystem in real time, and the data is backed up on the server at the same time.

b) Fatigue Analysis Subsystem

The fatigue analysis subsystem includes the analysis of the temperature field on the inner wall of the pipeline and the fatigue analysis of the pipeline.

1) Analysis of the temperature field of the inner wall

The temperature field of the inner wall of the pipeline is calculated from the outer wall temperature and flow rate data received from the data acquisition subsystem.

2) Fatigue analysis

Based on the analysis of the inner wall temperature field, the pipeline fatigue analysis and life prediction are carried out. The thermal stress caused by fluid temperature change is calculated, and the information of mechanical load, pressure load and seismic load is combined. At the same time, considering the influence of coolant environment on fatigue of the pipeline, the fatigue coefficient of the pipeline and equipment is calculated in real time, and its service life is predicted.

The Loose Parts and Vibration Monitoring system (KIR [LPVM]) information will be used to detect and locate loose parts in the Reactor Coolant System during reactor operation in order to prevent potential damage to the SG, Reactor Coolant Pump or RVI, and to monitor in-service vibration response of the RPV and RVI, which is useful to detect mechanical deterioration.

The KIR [LPVM] is categorised and classified NC and consists of two sub-systems: Loose Parts Monitoring Sub-system (LPMS) and Vibration Monitoring Sub-system

(VMS).

a) Loose Parts Monitoring Sub-system (LPMS)

The function of the LPMS is to detect and locate the loose parts for primary coolant under normal operating conditions of reactor. The LPMS allows simultaneously monitoring for three SGs, three reactor coolant pumps, and the RPV.

The LPMS will generate two alarms (loose parts monitoring and system faults) and send them to the main control room via the Plant Computer Information & Control System (KIC [PCICS]).

The vibration data of 27 channels are collected continuously by the LPMS. Each channel includes an accelerometer and a remote pre-amplifier. The high frequency vibration signals of accelerometers are caused by impact of loose parts. The accelerometer produces an electrical signal proportional to the vibration acceleration, and the pre-amplifier converts it into a voltage signal. The voltage signal is adjusted and amplified by the signal processor, which is installed in the cabinet.

b) Vibration Monitoring Sub-System (VMS)

The function of the VMS is to monitor the actual vibration response of the RPV and internals to detect mechanical degradation.

The VMS also monitors the neutron level and neutron noise signals of the Nuclear Instrumentation System (RPN [NIS]) and the signals from accelerometers on RPV. The neutron fluctuation or noise reflects the change to thickness of the water gap between the reactor vessel and internals and thus becomes an indicator of movement of reactor internals.

The computer carries out the spectra analysis and generates the diagnostic report of the vibration condition of the RPV.

The leak detection is used to detect the leakage from the RCPB and MSL (inboard containment) through water level, flow, temperature, humidity measurements, and includes both identifiable leakage and unidentifiable leakage.

For example, the Leakage Monitoring System (KIL [LMS]) is designed to monitor the leakage in the MCL, Surge Line and the MSL (steam generator outlet to containment penetration). The KIL system and other systems provide condensate flow monitoring, sump level monitoring, and total inventory testing which are used to evaluate the leakage rate, and temperature and humidity monitoring to locate the leak position. The readings of pressure, temperature and radioactivity of the containment air monitoring are used as supplement parameters for leakage identification.

It should be noted that Leak Before Break (LBB) is not an argument in the demonstration of structural integrity of the UK HPR1000 components. It is only considered as a supplementary and additional supporting measure to provide further defence in depth for pipework which is significant to the safety and integrity of the nuclear plant.

17.6.1.3.3 Irradiation Surveillance

The surveillance programme for the RPV core region low alloy steel base metal and weld metal will be established in order to provide mechanical properties data and irradiation embrittlement temperature shift of RPV core shell material after irradiation. This data will support fracture mechanical analysis, determination of Pressure-Temperature limits and hydrotest temperature during operation.

For the UK HPR1000, irradiation surveillance capsules will be installed into the outside brackets of RVI barrel to undergo high neutron flux in the core section. After a specified duration of neutron irradiation, the radiation damage of the core region material will be evaluated by analysing the mechanical property test results of the irradiated specimens and thus predicting the degradation level of RPV core shell.

The irradiation surveillance requirement of RPV core region material is presented in Reference [45], *Irradiation Surveillance Requirement of RPV Core Region Material*, which includes

a) Applicable codes and standards.

b) Irradiation capsule quantity and composition.

c) Pre-irradiation and post-irradiation tests.

d) Determination of material strength.

e) Determination of the critical stress intensity factor $K_{IC}$.

f) Determination of Reference Nil Ductility Transition Temperature ($RT_{NDT}$).

g) Determination of upper shelf energy.

h) Determination of neutron flux and temperature, and evaluation of test data.

### 17.6.2 Structural Integrity Class 1 Components

For SIC-1 components, assurance of integrity will be provided through compliance with the appropriate design codes and standards, and taking into account additional relevant requirements according to CGN's experience from CPR1000 projects, as well as international OPEX from existing PWRs.

A number of CSRs are provided for the significant SIC-1 components, based upon the similar Sub-claims (1&3) and demonstration approach used for the HIC components, to demonstrate structural reliability achieved through design, manufacture and installation, inspection and testing, operation and analysis. When compared to HICs, the requirements for quality of build (e.g. relating to manufacturing inspection) and design analysis are less stringent, and are dictated mainly by the design code. Thus SIC-1 component quality will be demonstrated through design and manufacture in compliance with the requirements of recognised nuclear design codes and standards.

The safety arguments and evidence for SIC-1 components are structured according to Sub-Claims 1&3 presented in Sub-chapter 17.6.1. Detailed information is presented in Reference [4] *Safety Case Methodology for HIC and SIC Components*. The key elements are summarised below.

a) The design code, which embodies extensive OPEX relevant to PWR plant components, is adopted to provide systematic and widely-approved measures for controlling quality of design and manufacture. The relevant evidence related to compliance with code requirements will be presented in the relevant CSRs, in order to ensure a structurally robust design and provide effective measures to prevent failure and to minimise and control component degradation at the design and construction stage. Based on the feedback and OPEX in previous units, the stringent requirements above the codes' requirements on materials, welding and inspection are also presented in the CSRs.

b) For the design analysis of SIC-1 components, the methods and requirements stipulated in the design code are adopted to support the design for all conditions within the design basis. The associated assessments and results will be presented in the relevant CSRs as evidence to deterministically justify the structural integrity of UK HPR1000 components against stress and fatigue limits established in the design code.

c) Regarding SIC-1 components, the materials selected meet the code requirements, as well as a number of additional requirements according to the feedback from existing PWRs for some significant components, which will be provided in the related CSRs. The intent is to ensure well-proven materials are chosen, that are resistant to fracture and are of suitable composition to limit the effect of through-life degradation.

   In Step 3 of GDA, *Material Selection Report of RVI* has been established, in Reference [46], in addition, the ageing and degradation report has also been established for RVI, in Reference [47], which is used to support the related material selection report. In Step 4, the representative SIC-1 SSCs will be sampled for judicious material selection demonstration.

d) The manufacture of SIC-1 components is controlled based on the code requirements. It covers approved manufacturing procedures, the mechanical testing of materials, qualified welding procedures, welder qualification, manufacturing inspection, cleanliness, package and shipment. Evidence related to these aspects will be provided in relevant CSRs to demonstrate compliance with appropriate manufacturing requirements of relevant codes. Relevant OPEX and good practice will be considered as well.

e) For SIC-1 components, NDT requirements prescribed in the RCC-M code will be performed to confirm the quality of manufacture. PSI and ISI requirements of SIC-1 components will be determined and qualified according to the RSE-M code.

### 17.6.3 Structural Integrity Class 2 and 3 Components

Structural integrity of SIC-2 and SIC-3 components will be demonstrated through compliance with the requirements of recognised nuclear design codes or appropriate industrial standards. The specific safety case methodology is presented in Reference [4].

The standard design, manufacture and inspection technical specifications in accordance with code requirements are applied to underpin integrity Sub-claims 1&3. The specific applicable codes and standards will be presented in the related technical documents, which are consistent with the code requirements for the choice of material that has been successfully applied for existing PWR projects.

A number of CSRs will be developed for significant SIC-2 and SIC-3 components.

## 17.7 Loading Conditions

The integrity of each class of metal component and structure is demonstrated by taking account the design loading conditions, which are selected to constitute a reference base for evaluation of equipment. The loads are defined by the identification of credible scenarios likely to occur during the plant throughout design life, the faulted conditions for the UK HPR1000 and relevant OPEX.

Loading conditions are those bounding conditions which are considered to happen during design life of the plant. Each loading condition is characterised by a set of loads parameters related to pressure, temperature, flow rate, reaction forces, and other service loading. Loading conditions due to plant events are considered to assess the structural integrity. Plant events are classified under the following five categories:

a) First category (reference condition).

b) Second category conditions (normal and upset condition).

c) Third category conditions (emergency conditions).

d) Fourth category conditions (faulted conditions).

e) Test conditions.

The detailed loading conditions for structural integrity assessments will be described in the respective assessment reports. Each assessment report covers the detailed information on loading and combination. Loading conditions are typically drawn from the design specification documents for respective components. Uncertainty of loading and loading combination is considered, thus reasonable and conservative loadings and combinations for stress analysis, fatigue and DTAs are determined.

### 17.7.1 Service Loading

The system and component loadings include normal loads, transient loads (in Reference [48]), other dynamic loads, and loadings induced by internal and external hazards (if relevant). Residual stresses are also considered in DTA assessment for welds.

a) Loadings induced by operation conditions:

   1) Normal loads: Internal and external pressure, weight of the component and normal contents under operating or test conditions, dead weight and permanent loads, reaction forces, restraint thermal expansion, etc.

   2) Transient loads: Pressure, thermal and flow-rate of transient fluctuating (including tests).

   3) Other dynamic loadings in operation.

b) Loadings induced by hazards

   1) Loading induced by internal hazards, such as pipe whip.

   2) Loading induced by external hazards, such as Safe Shutdown Earthquake (SSE) and Seismic anchor motion displacements.

c) Residual stresses of welds.

### 17.7.2 Loading Combinations

Loading combinations are considered in each plant event category (operation conditions). Loadings should be properly combined according to the following principles:

a) The loadings induced by a situation are taken into account.

b) When situations independently occur at the same time in certain probabilities, the loads from each situation are combined.

c) The loadings induced by internal (if relevant) or external hazards are combined with proper loading in the specific event category, considering the frequency of the initiating hazard.

The document presenting *System and Components Loadings for Defect Tolerance Assessment* is presented in Reference [49].

## 17.8 ALARP Assessment

The PCSR Chapter 33 presents a generic approach and requirements used for implementing demonstration of ALARP.

By the overall application of the ALARP principle, an ALARP demonstration report for structural integrity is developed to outline the ALARP justification for the structural integrity of metallic components and structures in the UK HPR1000. The *ALARP Demonstration Report of PCSR Chapter 17* is presented in Reference [50]. This report presents a high level review of the UK HPR1000 design against RGP and OPEX in order to identify potential gaps. This ALARP demonstration report also provides links to the lower level supporting documents for the structural integrity area and provides evidence to support Claim 3.4.8.

As a part of the ALARP demonstration, in terms of structural integrity, the relevant ALARP assessment process follows a generic approach, and is summarised as follows:

a) Identifying the RGP.

b) Systematic review of UK HPR1000 design against RGP to identify gaps, and to further determine potential improvements.

c) Undertaking the optioneering process for the specific potential improvement.

d) Reviews to determine no further improvement options are reasonably practicable.

### 17.8.1 Sources of RGP and OPEX

The following are sources of RGP relevant to structural integrity:

a) IAEA Safety Reports, Guides and TECDOCs, WENRA Guidance.

b) Relevant UK Legislation, Approved Code of Practice and guidance (ACoPs).

c) UK Requirements, including SAPs and TAGs.

d) Recognised Design Codes and Standards.

The following are sources of OPEX relevant to structural integrity:

a) Lessons learnt from previous GDAs.

b) Advanced PWR design features.

c) International OPEX sharing from authority website.

d) CGN OPEX feedback.

These RGP and OPEX are important parts and integrated in UK HPR1000 structural integrity area.

### 17.8.2 Consistency Review of RGP

RGP conformance analysis is the starting point of the ALARP analysis. A thorough review of the RGP is undertaken to help in identifying gaps between the UK HPR1000 current design and RGP.

Based on step 2 and step 3 progresses in the structural integrity area, the following measures have been established or are being implemented for the UK HPR1000 to demonstrate the risk of failure is reduced to ALARP:

a) Establish a systematic method to determine and allocate an appropriate structural integrity class to each metal SSC, so that the appropriate and reasonable design and manufacturing codes and standards can be applied, commensurate with their safety class and structural integrity class.

b) Supplementing the avoidance of fracture philosophy into the structural integrity demonstration for the highest reliability components, to enhance integrity reliability and meet the highest reliability claims. The avoidance of fracture demonstration comprises use of DTA based on R6, high reliability NDT based on ENIQ and interaction with conservative material properties.

c) Establish a safety case methodology based on a multi-legged approach which is in line with UK good practice for guiding designers to develop systematic and comprehensive arguments to enhance the reliability of SSCs.

d) Establish an ALARP methodology to guide the systematic ALARP assessments which can be implemented for potential improvement.

In addition, the consistency review is an ongoing process during the whole GDA phase, the new potential improvements could be identified along with the progress of subsequent stages.

In summary, the ALARP assessment is an iterative process. This report identifies the finished works in step 3 and future potential works that will be completed during step 4. Where relevant, new ALARP work is identified, evaluated and implemented to optimise the design and further reduce risk in subsequent steps. The report *ALARP demonstration report of structural integrity area* summarises all ALARP activities relevant to structural integrity.

## 17.9 Concluding Remarks

This chapter presents the route map for the structural integrity demonstration and the methodology for constructing safety case reports of all classes of components and structures which are significant to nuclear safety in the UK HPR1000. The process commences with the identification of safety functional requirements of different classes of metal components and structures. A systematic approach is then established to present the structural integrity classification process and relevant requirements.

Based on the classification of structural integrity, the methodology of constructing safety case reports for each class of components and structures and main contents are presented, with a particular focus on HIC components, the structure of CSR reports is presented under the CAE format.

During step 3, the CSRs of PZR, SG, MCL and RVI, the code applicability justification report, material selections of RPV, PZR, SG, MCL and RVI, outline of PSI and ISI requirements, quality assurance grading, irradiation surveillance requirements, the avoidance of fracture of one RPV limiting weld, a number of ALARP demonstration reports, etc., are already finished and submitted.

In step 4, the remaining CSRs, the material selection reports, the ageing and degradation reports, the DTA reports and TJ for HIC components, the code compliance analysis report, the other ALARP demonstration reports will be finished, in order to support the structural integrity demonstration to ensure the nuclear safety and security of plant throughout the lifetime.

At this stage, the contents of this chapter describe the current state of development regarding the demonstration of structural integrity for the UK HPR1000. The future revisions of this chapter, its appendices and supporting documents will present more detailed information to substantiate the structural integrity safety case.

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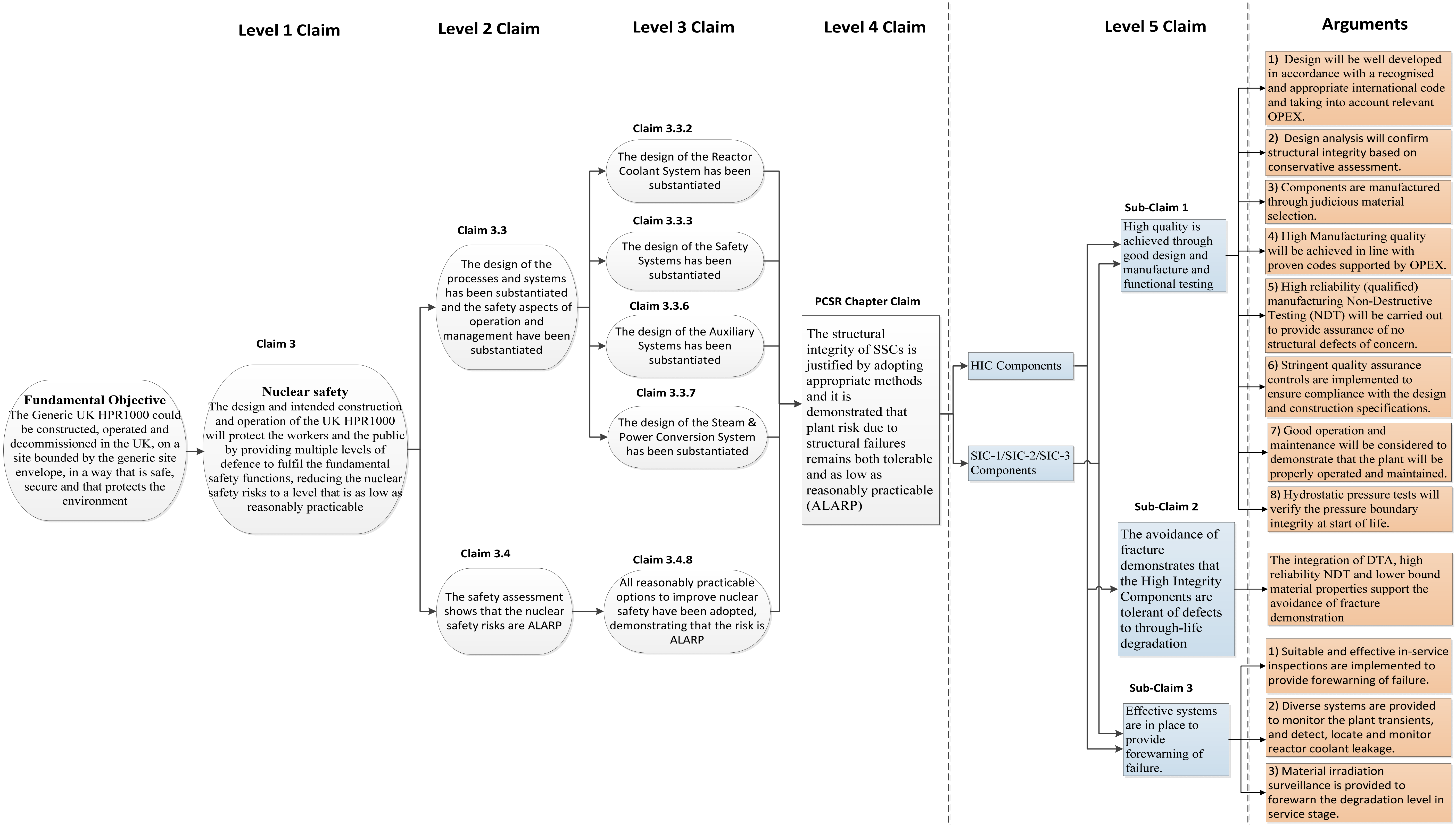
## Appendix 17A Chapter Route Map of Structural Integrity




F-17A-1 Chapter Route Map of Structural Integrity

# Appendix 17B Reactor Pressure Vessel Component Safety Report

This appendix provides the summary of the RPV Component Safety Report. This CSR supports the chapter claim of structural integrity, and demonstrates that gross failure of the RPV can be discounted for the design life of 60 years.

The objective of this CSR is to demonstrate the structural integrity of RPV over 60 years of design life in the format of CAE, and to ensure the safety functional requirements in terms of quality of design, manufacture, installation and operation of the RPV. This CSR is demonstrated through the TAGSI approach based on offering arguments, underpinned with appropriate and relevant evidence. The claims are as follows:

Sub-Claim 1: High quality is achieved through good design and manufacture, and functional testing.

Sub-Claim 2: The avoidance of fracture demonstrates that the High Integrity Components are tolerant of defects to through-life degradation.

Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

These claims are supported by a series of safety arguments and evidence, which cover a number of aspects, such as design, manufacture, inspection, operation, maintenance and defect tolerance assessment.

The approach has been chosen to conform to UK good practice for the RPV classified as HIC. A series of safety arguments and evidence are provided to support each claim. For Sub-claim 1, the arguments and evidence are from the following aspects:

a) The design is well developed in accordance with recognised and appropriate international codes. Known degradation mechanisms and OPEX were considered.
b) Proven materials based on or exceeding the requirements of RCC-M are selected. Through strict control on chemical composition, the effects of irradiation embrittlement and thermal ageing are limited. Material mechanical properties are confirmed by sufficient mechanical testing in accordance with pre-defined material specifications. Meanwhile, materials are compatible with the environment.
c) Design analysis complies with the allowable stress limits and fatigue usage factors as specified in RCC-M B3000. Seal analysis is carried out to ensure no leakage through the seal gasket. Fast fracture analysis is performed according to RCC-M appendix ZG through a postulated conventional defect.
d) High manufacturing quality will be achieved through a series of measures, including experienced suppliers, using recognised nuclear codes, qualified production workshop, qualified welding procedures and operators, recorded deviations and repairs, manufacture records and procedures.

e) Manufacturing NDT requirements stipulated in RCC-M code will be performed to confirm the quality of manufacture.
f) Stringent quality assurance is very important to ensure the design and manufacture quality. The corresponding requirements and procedures are performed to control the item and activity quality.
g) The structural integrity is confirmed by functional testing.

For Sub-claim 2, the avoidance of fracture is composed of three important contributing elements: DTA, high reliability NDT and lower bound material properties. This will involve undertaking a detailed fracture mechanics based defect tolerance assessment to determine the limiting defect sizes on the HIC components at the end of life taking account of any potential to grow the defects through life. The qualified non-destructive examinations being proposed for the components can then be shown to be able to reliably detect such postulated start of life defects with a suitable margin. Three welds and one vulnerable non-welded region are considered as following during the GDA process: 1) Core shell to flange-nozzle shell weld, 2) Inlet nozzle to flange-nozzle shell weld, 3) Inlet nozzle to safe end weld and 4) core shell.

For Sub-claim 3, ISI will be implemented to assure that any defect will be detected before it threatens the structural integrity of RPV and ISI technique for HIC welds will be qualified according to ENIQ methodology. Diverse systems are provided to monitor the reactor coolant identified leakage between closure head flange and vessel body flange. Material irradiation surveillance of core shell is required to forewarn the failure.

The above arguments act as diverse defence in depth to prevent failure through conservative, robust design and construction. Surveillance, inspections, operation rules, leakage monitoring, periodic testing and lifetime record requirement as presented in this CSR which validate the integrity though lifetime provide secondary defence in depth measures to detect and provide forewarning of structural failure.

It is concluded that the safety arguments presented in CSR identifies a suitable and sufficient diversity of evidence to substantiate the structural reliability claimed for the RPV.

Besides, the ALARP principle is considered and applied throughout the whole process of structural integrity demonstration. The details and the evidence are included in the document of *Reactor Pressure Vessel Component Safety Report*, Reference [6].

# Appendix 17C Pressuriser Component Safety Report

This appendix provides the summary of Pressuriser Component Safety Report. The objective of this report is to demonstrate the structural integrity of the PZR over 60 years of design life in the format of Claims, Arguments, Evidence (CAE), and to ensure the safety functional requirements in terms of quality of design, manufacture, installation and operation of the PZR.

The structural integrity demonstration of the PZR is developed to meet the following three Sub-Claims:

Sub-Claim 1: High quality is achieved through good design and manufacture, and functional testing.

Sub-Claim 2: The avoidance of fracture demonstrates that the High Integrity Components are tolerant of defects to through-life degradation.

Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

These claims are supported by a series of safety arguments and evidence, which cover a number of aspects, such as design, manufacture, inspection, operation, maintenance and defect tolerance assessment. The reliability claims of the PZR are based on methods and requirements of structural integrity classification. The heads, shells and manway forging of PZR have been classified as HIC, for which the highest level of structural reliability must be demonstrated. The heaters, manway cover, bolts and nuts of manway are classified as SIC-1 which requires somewhat less stringent and extensive substantiation of structural reliability than HIC.

A series of safety arguments and evidence are provided to support each claim. For Sub-claim 1, the arguments and evidence are from the following aspects:

a) The design is well developed in accordance with recognised and appropriate international codes. Known degradation mechanisms and OPEX were considered.
b) Proven materials based on or exceeding the requirements of RCC-M are selected. Material mechanical properties are confirmed by sufficient mechanical testing in accordance with pre-defined material specifications. Meanwhile, materials are compatible with the environment.
c) Design analysis complies with the allowable stress limits and fatigue usage factors as specified in RCC-M B3000. Fast fracture analysis is performed according to RCC-M appendix ZG through a postulated conventional defect.
d) High manufacturing quality will be achieved through a series of measures, including experienced suppliers, using recognised nuclear codes, qualified production workshop, qualified welding procedures and operators, recorded deviations and repairs, manufacture records and procedures.

e) Manufacturing NDT requirements stipulated in RCC-M code will be performed to confirm the quality of manufacture.
f) Stringent quality assurance is very important to ensure the design and manufacture quality. The corresponding requirements and procedures are performed to control the item and activity quality.
g) The structural integrity is confirmed by functional testing.

For Sub-claim 2, the avoidance of fracture is composed of three important contributing elements: DTA, high reliability NDT and lower bound material properties. This will involve undertaking a detailed fracture mechanics based defect tolerance assessment to determine the limiting defect sizes on the HIC components at the end of life taking account of any potential to grow the defects through life. The qualified non-destructive examinations being proposed for the components can then be shown to be able to reliably detect such postulated start of life defects with a suitable margin. Two welds are considered as following during the GDA process: 1) PZR upper cylindrical shell to middle cylindrical shell weld, 2) and PZR man-way flange to upper cylindrical shell weld.

For Sub-claim 3, ISI will be implemented to assure that any defect will be detected before it threatens the structural integrity of PZR and ISI technique for HIC welds will be qualified according to ENIQ methodology. Diverse systems are provided to monitor the plant transients, and detect, locate and monitor reactor coolant leakage, for example, the Leakage Detection System (KIL [LMS]), Plant Radiation Monitoring System (KRT [PRMS]).

The above arguments act as diverse defence in depth to prevent failure through conservative, robust design and construction. Surveillance, inspections, operation rules, leakage monitoring and periodic testing and lifetime record requirement as presented in this CSR which validate the integrity though lifetime provide secondary defence in depth measures to detect and provide forewarning of structural failure.

It is concluded that the safety arguments presented in CSR identifies a suitable and sufficient diversity of evidence to substantiate the structural reliability claimed for the PZR.

Besides, the ALARP principle is considered and applied throughout the whole process of structural integrity demonstration. The details and the evidence are included in the document of *Pressuriser Component Safety Report,* Reference [7].

# Appendix 17D Steam Generator Component Safety Report

This appendix provides the summary of the Steam Generator Component Safety Report. The objective of this report is to demonstrate the structural integrity of the SG over 60 years of design life in the format of Claims, Arguments, Evidence (CAE), and to ensure the safety functional requirements in terms of quality of design, manufacture, installation and operation of the SG.

The structural integrity demonstration of the SG is developed to meet the following three Sub-Claims:

Sub-Claim 1: High quality is achieved through good design and manufacture, and functional testing.

Sub-Claim 2: The avoidance of fracture demonstrates that the HIC are tolerant of defects to through-life degradation.

Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

These claims are supported by a series of safety arguments and evidence, which cover a number of aspects, such as design, manufacture, inspection, operation, maintenance and defect tolerance assessment. The reliability claims of the SG are based on methods and requirements of structural integrity classification. The tubesheet, primary head, steam drum head, secondary shells of SG have been classified as HIC, for which the highest level of structural reliability must be demonstrated. The main feedwater nozzle and auxiliary feedwater nozzle are preliminary classified as HIC. And the other parts are classified as non-HIC which required somewhat less stringent and extensive substantiation of structural reliability than HIC.

A series of safety arguments and evidence are provided to support each claim. For Sub-claim 1, the arguments and evidence are from the following aspects:

a) Structural integrity is assured by how modern and well established good practices have been implemented in SG design (based on ASME code), benefiting from the long operating history, known in-service degradation mechanisms, proven materials, extensive design analyses fully considering loads and load combinations covering all design conditions and good safety records of similar SGs, and also with balanced consideration of the benefits and detriments of alternative design options.
b) Proven materials based on or exceeding the requirements of ASME are selected. Material mechanical properties are confirmed by sufficient mechanical testing in accordance with pre-defined material specifications. Meanwhile, materials are compatible with the environment.
c) High manufacturing quality will be achieved through a series of measures, including experienced suppliers, using recognised nuclear codes, qualified

production workshop, qualified welding procedures and operators, recorded deviations and repairs, manufacture records and procedures.

d) Manufacturing NDT requirements stipulated in ASME code will be performed to confirm the quality of manufacture.

e) Stringent quality assurance is very important to ensure the design and manufacture quality. The corresponding requirements and procedures are performed to control the item and activity quality.

f) The structural integrity is confirmed by functional testing.

For Sub-claim 2, the avoidance of fracture is composed of three important contributing elements: DTA, high reliability NDT and lower bound material properties. This will involve undertaking a detailed fracture mechanics based defect tolerance assessment to determine the limiting defect sizes on the HIC components at the end of life taking account of any potential to grow the defects through life. The qualified non-destructive examinations being proposed for the components can then be shown to be able to reliably detect such postulated start of life defects with a suitable margin. For SG, three welds and one vulnerable non-welded region are considered during the GDA process: 1) Tube sheet to primary head weld, 2) Main feeder water nozzle to steam drum can no.2 weld, 3) Primary nozzle safe end to primary head. The assessment is to support the demonstration of SG is tolerant to potential manufacturing defects in considering the effects of through-life degradation mechanisms.

For Sub-claim 3, ISI will be implemented to assure that any defect will be detected before it threatens the structural integrity of SG and ISI technique for HIC welds will be qualified according to ENIQ methodology. Diverse systems are provided to monitor the plant transients, and detect, locate and monitor reactor coolant leakage, for example, the Leakage Detection System (KIL [LMS]), Plant Radiation Monitoring System (KRT [PRMS]), and Loose Part Monitoring System (LPMS).

The above arguments act as diverse defence in depth to prevent failure through conservative, robust design and construction. Surveillance, inspections, operation rules, leakage monitoring and periodic testing and lifetime record requirement as presented in this CSR which validate the integrity though lifetime provide secondary defence in depth measures to detect and provide forewarning of structural failure.

The strength of the safety case is founded on the diverse evidences which combine conservative and robust design, high reliability manufacturing NDT and high quality manufacture, suitable and effective ISI and diverse monitoring systems together.

It is concluded that the safety arguments presented in this CSR identifies a suitable and sufficient diversity of evidence to substantiate the structural reliability claimed for the SG.

Besides, the ALARP principle is considered and applied throughout the whole process of structural integrity demonstration. The details and the evidence are included in the document of *Steam Generator Component Safety Report*, Reference [8].

# Appendix 17E Main Coolant Lines Component Safety Report

This appendix provides the summary of the Main Coolant Lines Component Safety Report. The objective of this report is to demonstrate the structural integrity of the MCLs over 60 years of design life in the format of Claims, Arguments, Evidence (CAE), and to ensure the safety functional requirements in terms of quality of design, manufacture, installation and operation of the MCLs.

The structural integrity demonstration of the MCLs is developed to meet the following three Sub-Claims:

Sub-Claim 1: High quality is achieved through good design and manufacture, and functional testing.

Sub-Claim 2: The avoidance of fracture demonstrates that the High Integrity Components are tolerant of defects to through-life degradation.

Sub-Claim 3: Effective systems are in place to provide forewarning of failure.

These claims are supported by a series of safety arguments and evidence, which cover a number of aspects, such as design, manufacture, inspection, operation, maintenance and defect tolerance assessment. The reliability claims of the MCLs are based on methods and requirements of structural integrity classification. The hot legs, cold legs and crossover legs of the MCLs have been classified as HIC, for which the highest level of structural reliability must be demonstrated. The welded nozzles are preliminary classified as non-HICs which require somewhat less stringent and extensive substantiation of structural reliability than HIC.

A series of safety arguments and evidence are provided to support each claim. For Sub-claim 1, the arguments and evidence are from the following aspects:

a) The design is well developed in accordance with recognised and appropriate international codes. Known degradation mechanisms and OPEX were considered.
b) Proven materials based on or exceeding the requirements of RCC-M are selected. Material mechanical properties are confirmed by sufficient mechanical testing in accordance with pre-defined material specifications. Meanwhile, materials are compatible with the environment.
c) Design analysis complies with the allowable stress limits and fatigue usage factors as specified in RCC-M B3000.
d) High manufacturing quality will be achieved through a series of measures, including experienced suppliers, using recognised nuclear codes, qualified production workshop, qualified welding procedures and operators, recorded deviations and repairs, manufacture records and procedures.
e) Manufacturing NDT requirements stipulated in RCC-M code will be performed to confirm the quality of manufacture.

f) Stringent quality assurance is very important to ensure the design and manufacture quality. The corresponding requirements and procedures are performed to control the item and activity quality.

g) The structural integrity is confirmed by functional testing.

For Sub-claim 2, the avoidance of fracture is composed of three important contributing elements: DTA, high reliability NDT and lower bound material properties. This will involve undertaking a detailed fracture mechanics based defect tolerance assessment to determine the limiting defect sizes on the HIC components at the end of life taking account of any potential to grow the defects through life. The qualified non-destructive examinations being proposed for the components can then be shown to be able to reliably detect such postulated start of life defects with a suitable margin. For MCL, one weld is considered during the GDA process: MCL hot leg to SG inlet weld.

For Sub-claim 3, ISI will be implemented to assure that any defect will be detected before it threatens the structural integrity of MCL and ISI technique for HIC welds will be qualified according to ENIQ methodology. Diverse systems are provided to monitor the plant transients, and detect, locate and monitor reactor coolant leakage, for example, the Leakage Detection System (KIL [LMS]), the Fatigue Monitoring System (KIF [FMS]).

The above arguments act as diverse defence in depth to prevent failure through conservative, robust design and construction. Surveillance, inspections, operation rules, leakage monitoring and periodic testing and lifetime record requirement as presented in this CSR which validate the integrity though lifetime provide secondary defence in depth measures to detect and provide forewarning of structural failure.

It is concluded that the safety arguments presented in CSR identifies a suitable and sufficient diversity of evidence to substantiate the structural reliability claimed for the MCL.

Besides, the ALARP principle is considered and applied throughout the whole process of structural integrity demonstration. The details and the evidence are included in the document of *Main Coolant Lines Component Safety Report,* Reference [9].

# Appendix 17F Reactor Vessel Internals Component Safety Report

This appendix provides the summary of the RVI Component Safety Report (CSR). The objective of this CSR is to demonstrate the structural integrity of RVI over 60 years of design life in the format of Claims, Arguments, Evidence (CAE), and to ensure the safety functional requirements in terms of quality of design, manufacture, installation and operation of the RVI.

The structural integrity demonstration of the RVI is developed to meet the following two Sub-Claims:

Sub-Claim 1: High quality is achieved through good design and manufacture

Sub-Claim 2: Effective systems are in place to provide forewarning of failure.

These claims are supported by a series of safety arguments and evidence, which cover a number of aspects, such as design, manufacture, inspection, operation, maintenance and defect tolerance assessment. The reliability claims of the RVI are based on methods and requirements of structural integrity classification. The core barrel and upper internals have been classified as SIC-1.

A series of safety arguments and evidence are provided to support each claim. For Sub-claim 1, the arguments and evidence are from the following aspects:

a) The design is well developed in accordance with a recognised and appropriate international code. Extensive experience in design and construction as well as known degradation mechanisms and OPEX were considered. Design analysis is performed according to RCC-M code to demonstrate that design assessment has been done using conservative procedures and has adequate margin.
b) Proven materials based on or exceeding the requirements of RCC-M code are selected. Material mechanical properties are confirmed by sufficient mechanical testing in accordance with pre-defined material specifications. Meanwhile, materials are compatible with the environment.
c) High manufacturing quality will be achieved through a series of measures, including selecting experienced suppliers with a track record for producing similar components, manufacturing complies with the requirements of recognised nuclear codes and the relevant specifications, qualification of production workshop, qualification of welding procedures and operators, recording deviations and repairs record. A series of NDT methods and techniques are performed during manufacturing stages according to RCC-M. The manufacture records and procedures will ensure the high quality of product.
d) Functional testing will be performed to fulfill the safety functions at start of life. Chemistry parameter limits, condition of operation and in-service

maintenance will be considered to demonstrate that the plant will be properly operated and maintained.

For Sub-claim 2, suitable and effective inspections and monitor system are implemented to provide forewarning of failure during outage. Periodical visual inspection will identify defects that could lead to component failure. Loose part and vibration monitor system will monitor loose part and abnormal vibration of core barrel that could lead to component failure.

From the aspects of design, manufacture, material, test, inspection, operation and so on, arguments and evidence are provided to demonstrate that RVI can meet the requirements of safety function in the range of design basis to ensure nuclear safety of power plant.

It is concluded that the safety arguments presented in CSR identifies a suitable and sufficient diversity of evidence to substantiate the structural reliability claimed for the RVI.

Besides, the ALARP principle is considered and applied throughout the whole process of structural integrity demonstration. The details and the evidence are included in the document of *Reactor Vessel Internals Component Safety Report*, Reference [10].