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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

| | |
|---|---|
| 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 |
| FAD | Failure Assessment Diagram |
| F&PS | Fault and Protection Schedule |
| GDA | Generic Design Assessment |
| HIC | High Integrity Component |
| 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 |
| MSQA | Management for 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 System [RCS] |
| 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 |
| SIC-1 | Structural Integrity Class 1 |
| SIC-2 | Structural Integrity Class 2 |
| SIC-3 | Structural Integrity Class 3 |
| SSC | Structures, Systems and Components |
| SSE | Safety Shutdown Earthquake |
| TAGSI | Technical Advisory Group on Structural Integrity |
| 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. Reactor Coolant System (RCP [RCS]).

## 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 how the safety arguments and evidence for nuclear safety-related metal Structures, Systems and Components (SSC) 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 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) methodology are shown to provide a robust demonstration that the component will be free from structural defects of concern before entering service. 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, manufacturing inspection qualification, defect tolerance assessments and In-Service Inspection (ISI) requirements. The remainder of the CSR will be submitted in step 3.

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.

### 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 (reactivity control, fuel cooling and confinement of radioactive material),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.*

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

However, ***Claim 3.3*** and ***Claim 3.4*** can be split further into level 3 claims, according to the study and assessment scope of 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 demonstrates 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 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 due to structural failures is and remains both tolerable and ALARP throughout design lifetime.

### 17.2.2 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 lifetime for all conditions within the design basis.

According to 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 steel concrete containment liners.

The detailed scope of HIC components and significant Structural Integrity Class 1 (SIC-1) components will be presented in their respective CSRs.

A total of five CSRs will be submitted in GDA step 3, including the Reactor Pressure Vessel (RPV), Pressuriser (PZR), Main Coolant Line (MCL), Steam Generator (SG) and Reactor Vessel Internals (RVI), which provide the safety arguments and evidence to support structural integrity claims in the CAE format. 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.

CSRs related to the remaining HIC components and other significant SIC-1 components will be subsequently developed in the appropriate stage.

### 17.2.3 Chapter Route Map

For the structural integrity area, the Fundamental Objective, Level 1 Claim 3, Level 2 Claim 3.3&3.4, Level 3 Claim 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 is presented in Appendix 17A. The detailed CAE for HIC and SIC-1, SIC-2 and SIC-3 components are presented in Reference [1], *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 as 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 will be established, which can be found in References [1] , [2] and [3] .

### 17.2.4 Chapter Structure

This chapter comprises 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 linkage to 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 associated 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 some considerations and activities to present how the ALARP principle is applied in the structural integrity demonstration.

i) Sub-chapter 17.9 summarises the concluding remarks.

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

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

l) Appendix 17B presents a summary of RPV CSR.

### 17.2.5 Supporting Documents

In addition references quoted in this chapter, a number of CSRs will be provided and will present detailed arguments and evidence to demonstrate the structural integrity of HIC components and some significant SIC-1 components. There are five CSRs covered at this stage as below shown, the RPV CSR can be found in Reference [4], the remaining CSRs will be finished during step 3.

a) Reactor Pressure Vessel Component Safety Report;

b) Pressuriser Component Safety Report;

c) Main Coolant Lines Component Safety Report;

d) Steam Generator Component Safety Report;

e) Reactor Vessel Internals Component Safety Report.

### 17.2.6 Interfaces with Other Chapters

The interfaces with other chapters are listed in the following 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, Chapter 17 provides chapter claims, 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. |

| PCSR Chapter | Interface |
|---|---|
| Chapter 5 Reactor Core | Chapter 5 Reactor Core describes 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 will be discussed in Chapter 17. |
| Chapter 6 Reactor Coolant System | Chapter 6 provides general design information relevant to the main components of the Reactor Coolant System (RCP [RCS]).<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, Engineered Safety Features, etc.<br>The structural integrity classification and demonstration of relevant components are presented in Chapter 17. |
| Chapter 10 Auxiliary Systems | Chapter 10 provides the system description of the nuclear auxiliary systems.<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 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 are present in Chapter 17. |

| PCSR Chapter | Interface |
|---|---|
| Chapter 18 External Hazards | Chapter 18 presents information for external hazards which is applicable to the structural integrity demonstration. |
| Chapter 19 Internal Hazards | Chapter 19 presents the information for internal hazards which is applicable to the structural integrity demonstration. |
| Chapter 20 MSQA and Safety Case Management | Chapter 20 presents the organisational and quality assurance arrangements which are implemented in the design process and in the production of Chapter 17. |
| Chapter 21 Reactor Chemistry | Chapter 21 presents information on reactor chemistry which is applicable to the assessment of ageing and degradation for the materials selection and chemistry regime for Chapter 17. |
| Chapter 22 Radiological Protection | Chapter 22 provides radiological protection design considerations relevant to material selection.<br>Chapter 17 provides optimum material selection on minimisation of source term. |
| Chapter 24 Decommissioning | Chapter 24 presents the safe decommissioning design and adequate preparation of a decommissioning strategy and plan.<br>Chapter 17 covers material selection to minimise waste generation. |
| Chapter 31 Operational Management | Chapter 31 presents the arrangements for plant operational management.<br>Chapter 17 demonstrates the structural integrity of metal SSCs by taking into account plant operational management. |
| Chapter 33 ALARP Evaluation | Chapter 33 provides the general ALARP methodology and relevant requirements for UK HPR1000.<br>Chapter 17 presents ALARP demonstrations for structural integrity by applying the ALARP methodology provided in Chapter 33. |

## 17.3 Applicable Codes and Standards

For the UK HPR1000, the applicable codes and standards for the structural integrity area 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)), the document on *General Principles for Application of Laws, Regulations, Codes and Standards* in Reference [5] 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 cognisance of the 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. *Applicability Justification of Codes and Standards in Structural Integrity Area* will be developed and submitted during GDA step 3.

T-17.3-1 Main Applicable Codes and Standards

| Codes and Standards | Title |
|---|---|
| RCC-M 2007 Edition, in Reference [6] | Design and Construction Rules for Mechanical Components of PWR Nuclear Islands |
| RSE-M 2010+2012 Addendum, in Reference [7] | In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands |
| IAEA SSG-30, in Reference [8] | Safety Classification of Structures, Systems and Components in Nuclear Power Plants |
| ASME 2007+ 2008 Addendum Section II, III, V, IX and XI, in Reference [9] | ASME Boiler & Pressure Vessel Code |
| R6 Revision 4, in Reference [10] | Assessment of the Integrity of Structures Containing Defects |
| TEMA Revision 9, in Reference [11] | Standards of the Tubular Exchanger Manufacturers Association |

The applicable codes and standards of 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<br>Main Coolant Line<br>Surge Line<br>Pressuriser<br>Pump<br>Valve | RCC-M-2007 Section I Subsection A<br>RCC-M-2007 Section I Subsection B<br>RCC-M-2007 Section I Subsection Z<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 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 |
| 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 |

| Structures, Systems and Components (SSC) Type | Applicable Codes and Standards |
|---|---|
| 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="4">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</td></tr>
<tr><td>Piping</td></tr>
<tr><td>Pump</td></tr>
<tr><td>Valve</td></tr>
<tr><td>Pump</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</td></tr>
</table>

| Structures, Systems and Components (SSC) Type | Applicable Codes and Standards |
|---|---|
| Heat Exchanger | 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 |

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 [6], and the pre-service inspection (PSI) and ISI shall comply with RSE-M in Reference [7]. In addition, the R6 procedure is applied to defect tolerance assessments for HIC components.

For the SG, the design and manufacture are in line with ASME III in Reference [9] with additional enhancements drawn from RCC-M, with the PSI and ISI being based on RSE-M.

The additional requirements and their reasons are as follows:

a) Additional requirements from RCC-M

Based on the comparison between ASME and RCC-M, judging from the experience and feedback from CGN CPR1000 fleet construction, decisions regarding necessary additional requirements were made.

b) Additional Non-Destructive Testing (NDT) requirements from RCC-M/RSE-M

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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 [12].

For the topic area as a whole, the differences between the current applied codes and the latest version which may have an effect on the existing design of an SSC will be considered. These forward activities are covered in the report of *Applicability Justification of Codes and Standards in Structural Integrity Area*.

## 17.4 Safety Functional Requirements

Satisfying the safety design bases 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 bases of nuclear-safety systems. For example, the performance and safety design bases for the RCP [RCS] 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 [13] and [14] and 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 [2].

## 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 are ensured through meeting the requirements of codes and standards which are commensurate with the safety class of components. However, the safety classification methodology is not fully comprehensive and so lack of complete criteria means component 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 components and structures has been established.

The method of safety categorisation and classification discussed in PCSR Chapter 4.4.5 is based on IAEA SSG 30 in Reference [8], and UK expectations are also taken into account. The safety categorisation and classification of UK HPR1000 SSCs provides a systemic method for classifying components or structures and selecting applicable engineering design rules (such as 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 and process, definition of standards and corresponding relationship between safety class and design standard class are presented in detail in Reference [2].

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. In addition, the relationship among standard class determined by safety classification, structural integrity class, protection measures and potential failure consequence of components is 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 Single Failure Criterion (SFC) in analysis of Fault and Protection Schedule (F&PS) | 1. Limited core damage<br>2. Offsite minor radioactive release<br>3.Significant mass and energy release within nuclear island | SIC-1 |
| Standard Class 2 | Two diverse lines of protection together meeting SFC in F&PS | 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 F&PS | 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 will take into consideration previous GDA experience.

### 17.5.2 Results of Structural Integrity Classification

The components which have been assessed as HIC include the RPV, SG (the secondary shell, bottom head and tube sheet), PZR and MCLs. The detailed classification processes are presented in Reference [3].

The candidate HIC components include the surge line, reactor coolant pump and Main Steam Line (MSL). The analysis and assessment of these components are in progress. The latest information will be gradually supplemented according to the progress of structural integrity classification justifications of candidate HIC components.

## 17.6 Component Safety Cases

This sub-chapter presents the methodology for developing structural integrity safety cases for each of the classes 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 [1], which has been developed to be consistent with the recommendations of TAGSI.

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

For each HIC component, justifications will be presented in CSRs according to the following safety claims and the supporting arguments. More detailed information of claims, arguments and evidence can be found in Reference [1]. Herein shows the sub-claims for HICs to support the chapter claim.

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 are derived from TAGSI concepts, and the relevant information is presented in Reference [1].

Within the CSRs, relevant arguments will be produced from the above three perspectives 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 in this stage for demonstrating the highest reliability of components as follows. Further information is presented in Reference [1].

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, exceeding the requirements of codes, are identified and implemented as follows:

a) The appropriate 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 [15] 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 aging 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/HIC component sections will be designed to be as large as practicable with respect to manufacturing capability, in order to reduce the number 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 are to be generated, and representative boundary conditions and loads will 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 on which the designers may wish to focus. The stress on the path is linearised for the assessment of membrane stresses, 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 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 mechanics analyses based on design code, present acceptable provisions for fast fracture prevention. Fast fracture damage is considered to include brittle fracture and ductile tearing. Allowable pressure temperature (P-T) curves may be established according to these provisions, and used for operating and hydrotest conditions. The loading conditions are described in 17.7, and appropriate load combinations are 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 loads, while temperature is considered for judging impact on toughness of material. Fast fracture analyses cover Level A and B, C, D and level T.

Conservative methods will be used for the stress analysis. For example, conservative values will be used for parameter such as the loading combination, structural dimensions, material property and cladding, etc.

Detailed approaches of 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 stress analysis reports, which support the CSRs for HIC components.

17.6.1.1.3 Material selection

Materials are selected to satisfy the design requirements for components (HIC components, as well as SIC-1, SIC-2 and SIC-3 components), contributing to their ability to carry out their safety and environmental duty throughout the design life time of the plant.

Material selection for UK HPR1000 components is performed according to recognised and appropriate international codes and CGN experience from construction and operation of China CPR1000 projects, as well as OPEX from worldwide PWRs. A *Material Selection Methodology* report has been prepared for the UK HPR1000 in Reference [16], the key principles and process of material selection is presented in this reference. In addition, a *Material Selection Report of SSC* will be established according to this methodology and be submitted during GDA step 4.

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 worldwide 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’s minimum requirements are specified on impurity elements such as S, P, As, Sn, Sb, content of these elements is strictly limited to mitigate the risk of hot cracking and reheat cracking.

d) In addition, non-metal inclusion requirements are provided to guarantee the quality of the forgings.

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

The potential for in-service aging and degradation of materials to affect the design life of the HIC components has been considered in the design stage. Appropriate materials selection ensures compatibility with the operating environment against potential risks associated with identified degradation mechanisms, which comprise 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.

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 for the latter, may be more susceptible to embrittlement due to thermal aging after 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, the 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, 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.

More detail of the in-service material degradations (including the degradation modes, mechanisms, and the elimination or mitigation measures) will be presented in the report *Ageing and Degradation of SSC,* which is in progress and will be submitted during GDA step 4. However, 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 are to undergo periodic irradiation surveillance assessment to ensure that an adequate safety margin is maintained.

17.6.1.1.4 High quality of Manufacture

Based on the code requirements, the manufacture of HIC components is to adopt the proven techniques and approved procedures. The manufacturing is to be carried out in accordance with approved procedures. Qualified welding procedures complete with good practice and qualified welders will be used, and production test coupons are required for significant welds to ensure conformity with the requirements determined by the welding procedure qualification test.

Suppliers will be selected based on good practice and relevant experience. The production workshop will be technically qualified to evaluate the capacity and technical resources for carrying out required manufacturing and welding. 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 of product.

Evidence related to the aspects above will be provided to demonstrate compliance with the appropriate manufacturing requirements of RCC-M or ASME codes. Relevant OPEX and good practice will be considered and integrated into the appropriate evidence.

17.6.1.1.5 Manufacturing NDT

For HIC components, NDT requirements prescribed in the design code 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 at the end of manufacture.

The high reliability demonstration strategy and approach of objective-based manufacturing NDT for HIC components is presented in Reference [17]. It establishes several measures which will be taken to achieve high reliability of objective-based manufacturing NDT for HIC components:

a) Providing the inspection access assurance and applying the concept of ‘design for inspectability’ to maximise the effectiveness of NDT. During design and manufacture stage, access and inspectability for the implementation of manufacturing NDT should be provided for HIC components. Evidence demonstrating how this will be achieved will be provided during GDA step 3.

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.

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 will be established during GDA step 3, 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) is to contain welder qualification, weld procedures, testing of weld materials, the detailed manufacturing records, non-conformance reports and manufacturing inspection records. The relevant information will be presented in CSRs.

17.6.1.1.7 Operation and maintenance

According to the requirements which are defined in PCSR Chapter 4.4.6.2.3, the design should be such that activities for examination, maintenance, inspection and testing are facilitated for the purpose of maintaining 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 before entering service 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 condition of operation are to be provided in order to ensure good compatibility between primary pressure-retaining boundary materials and reactor coolant.

c) Periodic hydrostatic proof tests 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.

   In order to ensure that operators can safely operate and implement maintenance activities during operation, the operating and maintenance manuals that are compliant with the design specifications will be developed in appropriate stage.

17.6.1.1.8 Hydrostatic Pressure Test

Pressure testing is typically conducted on pressure vessels, pipework and systems after completion of manufacture and after installation according to code requirements, to confirm integrity at start of life through verifying strength and leak-tightness of relevant components and piping systems. 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 will also be presented in 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 arguments identified at this stage 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.

Further information is presented in Reference [1].

For UK HPR1000, HIC components will be assessed for their defect tolerance, taking account of material ageing and degradation. The supplementary fracture toughness testing will be carried out to underpin the fracture toughness values assumed in the defect tolerance assessment, and the toughness test requirements are listed in the document of *Supplementary Fracture Toughness Test Requirements of Materials for HIC Components* in step 4.

DTAs are performed, which exceed the code provisions for ensuring defect tolerance, to underpin a high level of reliability. The methodology of DTA is described in Reference [18]. The R6 procedure, which is in accordance with the elastic-plastic fracture mechanics, will be used to assess postulated defects with in-service crack growth, even though there is no expectation of significant defects (either from manufacture or from in-service mechanisms).

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

$$\mathrm{DSM} = \frac{\mathrm{ELLDS}}{\mathrm{QEDS} + \mathrm{LFCG}}$$

Where ELLDS is the End of Life Limiting Defect Size, QEDS is the Qualified Examination Defect Size and LFCG is the Lifetime Fatigue Crack Growth.

DSM is provided as a pragmatic approach to demonstrating the appropriate margin, and a target value of at least 2 will be sought. The DTA assessment undertaken should be conservative and provide sufficient margin. A simple but conservative method will be selected. It should be noted that refinement of demonstration of DTA is possible. However, this should not extent to undermine the expectation of using a conservative fracture assessment.

Since defects most likely occur in welds, the welds of HIC components are considered for defect tolerance assessment. Homogeneous and dissimilar metal welds are included, along with weld repair, if any. The *Application of Weld Ranking Procedure* in Reference [19] identifies the locations which are considered bounding with regard to defect tolerance, and for which R6 defect tolerance assessment is required. Weld ranking considers applied stresses, stress intensity factors, fatigue usage factors, and the NDT inspectability. Based upon this, the extent of the assessments required is established.

Additionally, non-welded regions of HIC components are also studied. As a result, non-weld locations of HIC components, such as RPV core shell, are included to support the demonstration of structural integrity of HIC components.

Defect characterisation is the term given to the process of modelling existing or postulated flaws by 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, parametric study of different defect shape ratios, orientations and positions are considered.

Loading condition is described in sub-chapter 17.7. For normal operations, assessment is based upon bounding material properties, upper bound crack growth and with 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 (FAD). This parameter varies and contains transients. Using stresses and stress intensity factors, calculate maximum and minimum pair for each transient. Based on Paris' law, the LFCG is calculated. The sum of QEDS and LFCG provides final defect size. It is necessary to inform NDT designers of the QEDS to enable NDT inspection and qualification to be undertaken.

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 [20]. In order to reduce the number of HIC areas to be assessed within GDA, a procedure named *Weld Ranking Procedure* in Reference [21] has been developed for sampling and selection of the limiting HIC welds and non-welded regions. Inspection qualification will be limited to objective-based manufacturing NDT for HIC welds. Manufacturing NDT for HIC 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 subject to less residual stress compared to the welds.

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

During GDA, partial qualification activities for the typical limiting HIC welds and the capability statements for the typical limiting non-welded regions will be performed to provide confidence that these inspection requirements can be achieved. The limiting HIC welds and non-welded regions are derived from *Application of Weld Ranking Procedure* in Reference [19]. The partial qualification activities for the typical limiting HIC welds cover GDA Technical Justification 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 Technical Justification Report, which will be based on ENIQ Recommended Practice 2 in Reference [22], is a reduced version of Technical Justification Report and includes a summary of relevant input information, the overview of the proposed objective-based manufacturing NDT system and physical reasoning.

In GDA, inspection qualification and DTA will be undertaken in parallel; assumed QEDS, which is based on judgement of engineering experience and practice, will be used for GDA technical justification of inspection qualification prior to the completion of DTA. After completion of the DTA using a simplified and conservative method, reconciliation will be performed if the initially assumed QEDS is greater than QEDS established by DTA. Re-justification with the QEDS established through DTA, or another NDT technique with a higher inspection capability or re-assessment through detailed analysis of the DTA may be needed.

#### 17.6.1.3 Forewarning of Failure

In order to support Sub-Claim 3, there are 3 arguments identified in this stage for demonstrating the highest reliability of components through 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 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, UT, 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 [15]. A document outlining PSI and ISI requirements for UK HPR1000 (PSI/ISI programme is out of GDA scope) will be provided in GDA step 3.

17.6.1.3.2 Monitor 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 Loose Parts and Vibration Monitoring system (KIR [LPVM]) information will be used to detect and locate loose parts in the RCP [RCS] 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 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 the Leak Before Break (LBB) is not an argument in the demonstration of structural integrity of 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 material will be established during the manufacturing stage, in order to provide mechanical properties data and irradiation embrittlement temperature shift of RPV core shell material after irradiation for 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 for undergoing the high neutron fluence in the core section. After a certain time of neutron irradiation, the radiation damage of the core region material will be evaluated according to the mechanical property test results of the irradiated specimens in order to predict the degradation level of RPV core shell.

The irradiation surveillance requirement of RPV core region material will be developed and submitted in GDA step 3, which includes applicable codes and standards, irradiation capsule quantity and composition, pre-irradiation and post-irradiation tests, determination of material strength, determination of the critical stress intensity factor $K_{IC}$, determination of Reference Nil Ductility Transition Temperature ($RT_{NDT}$), determination of upper shelf energy, determination of neutron flux and temperature, and evaluation of test data. The relevant descriptions will be presented in the RPV CSR.

### 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 the CGN’s experience from China CPR1000 projects, as well as OPEX from the worldwide similar PWRs.

Some CSRs will be provided for the significant SIC-1 components, based upon the similar claims and arguments 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 by the design code alone. 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. The detailed information is presented in Reference [1], *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 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 CSRs. The intent is to ensure well-proven materials are chosen, and the materials are resistant to fracture and are of suitable composition to limit the effect of through-life degradation.

d) The manufacture of SIC-1 components is controlled based on the code requirements. It covers approved manufacturing procedures, mechanical testing of material, 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 the RCC-M or ASME code. 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 [1].

The standard design, manufacture and inspection technical specifications in accordance with code requirements are applied to underpin integrity claims. 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.

## 17.7 Loading Condition

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 loading is defined by the identification of credible scenarios likely to occur during the plant throughout design life, the fault 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 loading 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) Hydrotest conditions.

The detailed loading conditions for structural integrity assessments, which will include loading types, magnitudes, transient curves and combinations, will be described in the respective assessment reports. Each assessment report will cover the detailed information on loading types, conservative loading combinations, etc. Loading conditions are typically drawn from the design specification documents for respective components.

### 17.7.1 Service Loading

The system and component loadings include static loads, transient loads (in Reference [23]), other dynamic loads, and loadings induced by internal and external hazards (if relevant), etc. Residual stresses are also considered in DTA assessment.

a) Loadings induced by operation conditions:

   1) Static loads: Design pressure, Design temperature, Operating pressure, Operating temperature, Test pressure (including hydrostatic tests), Test temperature (including hydrostatic tests), Dead weight and permanent loads, Variable mechanical loads (reaction forces), Restraint thermal expansion, etc.

   2) Transient loads: Transient loads (including tests).

3) Other dynamic loadings in operation.

b) Loadings induced by hazards

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

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

c) Residual stresses of welds location.

### 17.7.2 Loading Combinations

Loading combinations are considered in each plant event category (operation conditions). Loadings should be properly combined for defect tolerance assessment 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 specific event category, considering the frequency of initiating hazard.

The document presenting *System and Components Loadings for Defect Tolerance Assessment* is presented in Reference [24].

## 17.8 ALARP Assessment

The PCSR Chapter 33 presents a generic approach and requirements used for implementing demonstration of ALARP. As a part of the ALARP demonstrations, in terms 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 for the structural integrity area:

a) International Atomic Energy Agency (IAEA) safety standards;

b) SAPs and TAGs;

c) Recognised design codes and standards;

d) Regulator expectations.

The previous GDA experiences are the source of OPEX. These information and regulator expectations 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 UK HPR1000 current design and RGP.

At this current stage, the gaps in the structural integrity area have been identified against existing RGP, and will be detailed analysed in GDA step 3 and 4.

In addition, *The ALARP Demonstration Report of Structural Integrity Area* will be developed in step 3 to describe the overall application of the ALARP principle applied to structural integrity area.

## 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. Finally, PZR CSR, SG CSR, MCL CSR, RVI CSR, code applicability justification reports, material selection, outline of PSI and ISI Requirements, quality assurance grading, irradiation surveillance requirements, ALARP demonstration reports, etc., will all be finished during step 3, in order to support the structural integrity demonstration to ensure the nuclear safety and security of plant throughout the lifetime.

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.

## 17.10 References

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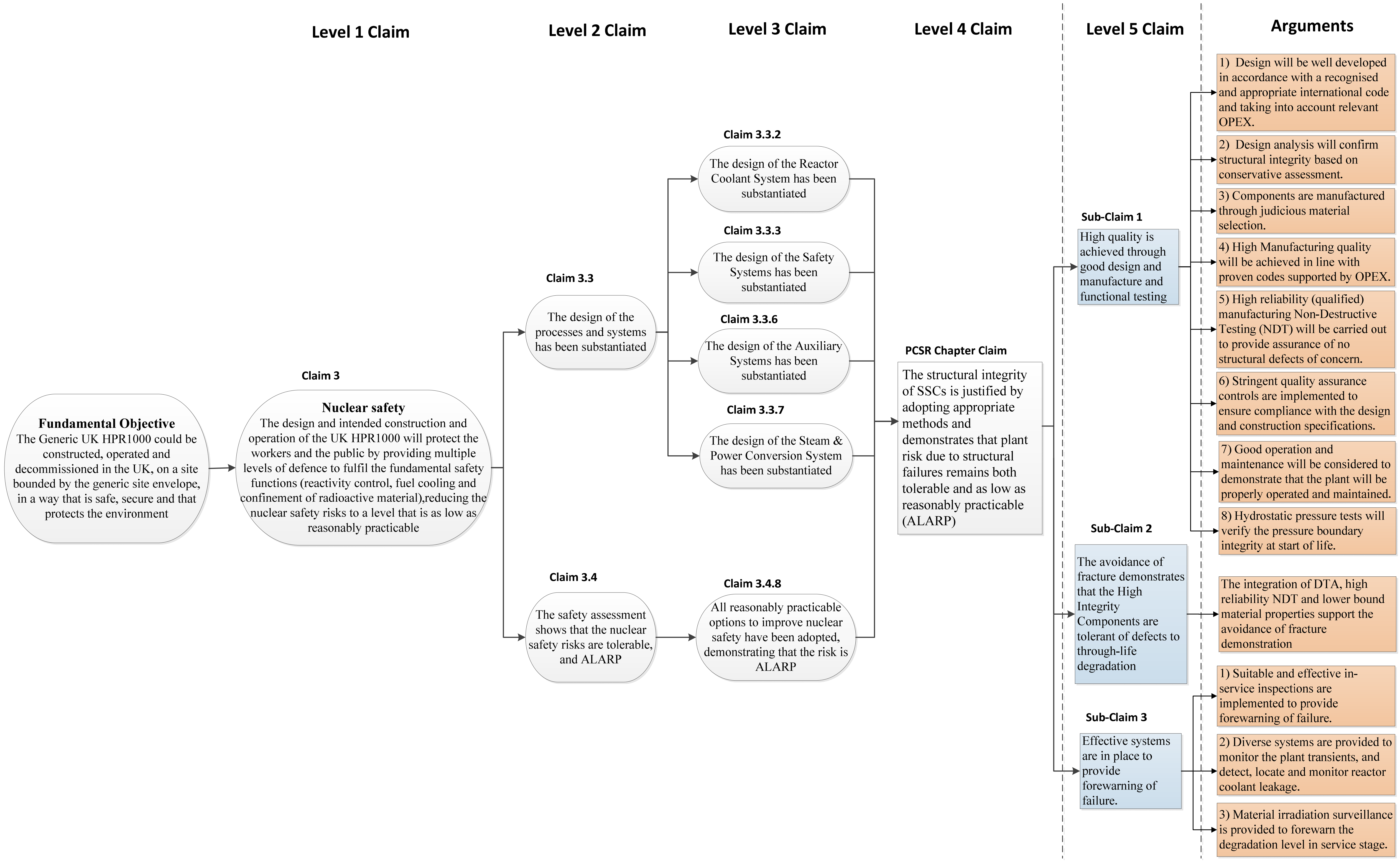
## Appendix 17A Chapter Route Map of Structural Integrity




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 in four legs, 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. Detailed information is presented in Reference [4], *Reactor Pressure Vessel Component Safety Report*.