Office for Nuclear Regulation

New Reactors Division – Generic Design Assessment

Step 4 Assessment of Structural Integrity for the UK HPR1000 Reactor

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EXECUTIVE SUMMARY

This report presents the findings of my assessment of the Structural Integrity aspects of the UK HPR1000 reactor design undertaken as part of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA). My assessment was carried out using the Pre-Construction Safety Report (PCSR) and supporting documentation submitted by the Requesting Party (RP).

The objective of my assessment was to make a judgement, from a Structural Integrity perspective, on whether the generic UK HPR1000 design could be built and operated in Great Britain, in a way that is acceptably safe and secure (subject to site specific assessment and licensing), as an input into ONR's overall decision on whether to grant a Design Acceptance Confirmation (DAC).

The scope of my GDA assessment was to review the safety aspects of the generic UK HPR1000 design by examining the claims, arguments and supporting evidence in the safety case. My GDA Step 4 assessment built upon the work undertaken in GDA Steps 2 and 3, and enabled a judgement to be made on the adequacy of the Structural Integrity information contained within the PCSR and supporting documentation.

My assessment focussed on the following aspects of the generic UK HPR1000 safety case:

- Development of the Structural Integrity safety case, in terms of claims, arguments and the provision of evidence to underwrite the assigned classifications of structures, systems and components.
- Structural Integrity classification of components, including highest reliability claims.
- Avoidance of fracture demonstration; including defect tolerance assessment (DTA), the development of part-technical justifications, the strategy to justify material properties and reconciliation process.
- Structural integrity provisions including design philosophy, design features/specifications and consideration of relevant OPEX.
- Demonstration of compliance with appropriate design codes and standards and the outputs from these analyses with the emphasis on the most safety significant components. I have also considered the RP's approach for the management of risks associated with combining codes and standards for certain high integrity components.
- Material selection, specifications, testing and control of design and manufacture, including third-party inspection (surveillance) in the design and manufacture of highest reliability structures and components.
- Application of design for inspectability of systems, structures and components, as well as review of the strategy for manufacturing and pre-service inspections.

The conclusions from my assessment are:

- The RP has developed an adequate safety case methodology and structure for the UK HPR1000, which demonstrates how the risks associated with structural integrity of the plant are identified, assessed and reduced as low as reasonably practicable.
- The RP has developed a suitable approach for the classification of Systems, Structures and Components (SSCs) important for safety. This shows how the SI approach is commensurate with safety significance, with additional measures where claims of the highest reliability are justified.
- The RP has provided adequate avoidance of fracture demonstrations for a selection of challenging welds for the purpose of GDA. These are based on conservative Defect Tolerance Assessments and appropriate GDA technical

justifications, that provide confidence in the future qualification of manufacturing inspections.

- The RP has selected and applied relevant design and construction codes, with sufficient evidence to demonstrate code compliance is achieved, based on conservative assumptions.
- The RP has proposed to combine codes and standards for demonstrating the safety of the Steam Generator and Reactor Coolant Pump components. ONR guidance states that "the combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated". From the information that I have sampled, I am satisfied that the RP has presented an adequate justification for adopting an approach of combining codes. Whilst I consider this adequate for the purposes of GDA, several items have been identified which will require further demonstration during site specific stages.
- The RP has developed and applied an adequate materials selection and testing strategy, which I consider provides sufficient evidence to underpin safety case claims of high-quality components and consideration of through life ageing and degradation.
- I am satisfied that the RP has presented sufficiently detailed proposals for Non-Destructive Examination (NDE) inspection that support the structural integrity claims in the PCSR and supporting documents. I am satisfied that the RP has paid due attention to the 'design for inspectability', resulting in several design modifications to improve NDE reliability.
- From a structural integrity perspective, the RP has demonstrated an understanding of ONR expectations through an adequate process for assessing and reducing risk ALARP, in terms of classification, component design, design code selection and materials selection. Whilst I have identified several opportunities for improvement in the demonstration of reducing risks to ALARP, I am content that these can be developed further during detailed design, construction and commissioning stages.

These conclusions are based upon the following factors:

- A detailed and in-depth technical assessment, on a sampling basis, of the full scope of safety submissions at all levels of the hierarchy of the generic UK HPR1000 safety case documentation.
- Independent information, reviews and analysis of key aspects of the generic Structural Integrity safety case undertaken by Technical Support Contractors (TSCs).
- Detailed technical interactions on many occasions with the RP, alongside the assessment of the responses to the substantial number of Regulatory Queries (RQs) and Regulatory Observations (ROs) raised during the GDA.

A number of matters also remain, which I judge are appropriate for a licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic UK HPR1000 design and safety submissions, but are primarily concerned with the provision of site-specific safety case evidence which will become available as the project progresses through the detailed design, construction and commissioning stages. These matters have been captured in 29 Assessment Findings.

Overall, based on my assessment undertaken in accordance with ONR's procedures, the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of the GDA process present an adequate safety case for the generic UK HPR1000 design. I recommend that, from a Structural Integrity perspective, a DAC may be granted.

LIST OF ABBREVIATIONS

ACC	Accumulator
AF	Assessment Finding
AFCEN	French Association for Design, Construction and In-Service Inspection Rules for Nuclear Steam Supply System Components
ALARP	As Low As Reasonably Practicable
AOFD	Avoidance of Fracture Demonstration
ARN	(Argentina Nuclear Safety Authority) Autoridad Regulatoria Nuclear
ASN	(French Nuclear Safety Authority) Autorité de Sûreté Nucléaire
ASDS	Atmospheric Steam Dump System
ASME	The American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BFX	UK HPR1000 Fuel Building
BMS	business Management System
BPVC	Boiler and Pressure Vessel Code
BRX	UK HPR1000 Reactor Building
BSA	UK HPR1000 Safeguards Building A
BSB	UK HPR1000 Safeguards Building B
BWXT	Designer and Manufacturer of UK HPR1000 Steam Generator
CAE	Claims-Arguments-Evidence
CCWS	Component Cooling Water System
CE	Civil Engineering
CH	Closure Head
CRDM	Control Rod Drive Mechanism
CGN	China General Nuclear Power Corporation Ltd
CI	Conventional Island
CSR	Component Safety Reports
CUF	Cumulative Usage Factor
DAC	Design Acceptance Confirmation
DBC	Design Basis Condition
DEC	Design Extension Condition
DMW	Dissimilar Metal Weld
DSM	Defect Size Margin
DTA	Defect Tolerance Assessment
ELLDS	End of Life Limiting Defect Size
ELPO	Extended Low Power Operation
EMIT	Examination. Maintenance, Inspection and Testing
ENIQ	European Network for Inspection and Qualification
EOMM	Equipment Operation and Maintenance Manual

ESPN	French Order concerning Nuclear Pressure Equipment
ESR	Essential Safety Requirements
FAD	Failure Assessment Diagram
FEA	Finite Element Analysis
FR	Frequency Response
FS	Fault Studies
FTT	Fuel Transfer Tube
GDA	Generic Design Assessment
GNI	General Nuclear International Ltd.
GNSL	General Nuclear System Ltd.
HAZ	Heat Affected Zone
HEPF	High Energy Pipe Failure
HIC	High Integrity Component
HOW2	(ONR) Business Management System
HPR1000WG	HPR1000 Design Specific Working Group (within MDEP)
IAEA	International Atomic Energy Agency
IEWG	Independent Expert Working Group
IH	Internal Hazards
IRWST	In-Containment Water Storage Tank
ISI	In-Service Inspection
ITPIA	Independent Third-Party Inspection Agency
IVR	In-vessel Retention
LB	Lower Bound
LBB	Leak Before Break
LBLOCA	Large-Break Loss of Coolant Accident
LFCG	Lifetime Fatigue Crack Growth
LSP	Lower Support Plate
MCL	Main Coolant Line
MDEP	Multinational Design Evaluation Programme (within OECD-NEA)
MFL	Main Feedwater Line
MSIV	Main Steam isolation Valve
MSL	Main Steam Line
MSQA	Management for Safety and Quality Assurance
MSRIV	Main Steam Relief Isolation Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MW	Megawatts
NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency (within OECD)

NI	Nuclear Island
NNR	(South Africa's) National Nuclear Regulator
NNSA	National Nuclear Safety Administration
NPP	Nuclear Power Plant
OECD	Organisation for Economic Cooperation and Development
ONR	Office for Nuclear Regulation
OpEx	Operational Experience
PCSR	Pre-construction Safety Report
PE(S)R	Pressure Equipment (Safety) Regulations
PSA	Probabilistic Safety Analysis
PSI	Pre-Service Inspection
PSR	Preliminary Safety Report (includes security and environment)
PSSR	Pressure Systems Safety Regulations
PWHT	Post Weld Heat Treatment
PWR	Pressurised Water Reactor
PZR	Pressuriser
QA	Quality Assurance
QEDS	Qualified Examination Defect Size
R6	Assessment of the Integrity of Structures Containing Defects
RCCA	Rod Cluster Control Assembly
RCC-M	Design and Construction Rules for Mechanical Components of PWR Nuclear Islands
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RTNDT	Nil Ductility Transition Temperature
RGP	Relevant Good Practice
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RSE-M	In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands
RVI	Reactor Vessel Internals
SAA	Severe Accident Analysis
SAP(s)	Safety Assessment Principle(s)
SCC	Stress Corrosion Cracking
SDM	System Design Manual
SEC-KSB	Shanghai Electric-KSB Nuclear Pump & Valve Co., Ltd.

SFAIRP	So Far As Is Reasonably Practicable
SFTT	Supplementary Fracture Toughness Tests
SFR	Safety Functional Requirement
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SI	Structural Integrity
SIC	Structural Integrity Classification
SIF	Stress Intensity Factor
SIS	Safety Injection System
SL	Surge Line
SoDA	(Environment Agency's) Statement of Design Acceptability
SSC	Structures, Systems and Components
ST	Surge Tank
SZB	Sizewell B
TAG	Technical Assessment Guide(s)
TAGSI	Technical Advisory Group on Structural Integrity
TCN	Technical Change Notice
TOFD	Time of Flight Diffraction
TJ	Technical Justification
TSC	Technical Support Contractor
UT	Ultrasonic Testing
VT	Visual Testing
WENRA	Western European Nuclear Regulators' Association
WRS	Weld Residual Stress

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1 INTRODUCTION

1.1 Background

1. This report presents my assessment conducted as part of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the generic UK HPR1000 design within the topic of Structural Integrity.

2. The UK HPR1000 is a pressurised water reactor (PWR) design proposed for deployment in the UK. General Nuclear System Ltd (GNSL) is a UK-registered company that was established to implement the GDA on the UK HPR1000 design on behalf of three joint requesting parties (RP), i.e. China General Nuclear Power Corporation (CGN), EDF SA and General Nuclear International Ltd (GNI).

3. GDA is a process undertaken jointly by the ONR and the Environment Agency. Information on the GDA process is provided in a series of documents published on the joint regulators' website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by the RP is a Design Acceptance Confirmation (DAC) from ONR and a Statement of Design Acceptability (SoDA) from the Environment Agency.

4. The GDA for the generic UK HPR1000 design followed a step-wise approach in a claims-argument-evidence hierarchy which commenced in 2017. Major technical interactions started in Step 2 of GDA which focussed on an examination of the main claims made by the RP for the UK HPR1000. In Step 3 of GDA, the arguments which underpin those claims were examined. The GDA Step 2 reports for individual technical areas, and the GDA summary reports for Steps 2 and 3 are published on the joint regulators' website. The objective of Step 4 of GDA was to complete an in-depth assessment of the evidence presented by the RP to support and form the basis of the safety and security cases.

5. The full range of items that form part of my assessment is provided in ONR's GDA Guidance to Requesting Parties (Ref. 1). These include:

   - Consideration of issues identified during the earlier GDA Step 2 and 3 assessments.
   - Judging the design against the Safety Assessment Principles (SAPs) (Ref. 2) and whether the proposed design ensures risks are As Low As Reasonably Practicable (ALARP).
   - Reviewing details of the RP's design controls and quality control arrangements to secure compliance with the design intent.
   - Establishing whether the system performance, safety classification, and reliability requirements are substantiated by a more detailed engineering design.
   - Assessing arrangements for ensuring and assuring that safety claims and assumptions will be realised in the final as-built design.
   - Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

6. The purpose of this report is therefore to summarise my assessment in the Structural Integrity (SI) topic which provides an input to the ONR decision on whether to grant a DAC, or otherwise. This assessment was focused on the submissions made by the RP throughout GDA, including those provided in response to the Regulatory Queries (RQs) and Regulatory Observations (ROs) I raised. Any ROs issued to the RP are published on the GDA's joint regulators' website, together with the corresponding resolution plans.

1.2 Scope of this Report

7. This report presents the findings of my assessment of the structural integrity of the generic UK HPR1000 design undertaken as part of GDA. I carried out my assessment using the Pre-construction Safety Report (PCSR) (Ref. 3) and supporting documentation submitted by the RP. My assessment was focussed on considering whether the generic safety case provides an adequate justification for the generic UK HPR1000 design, in line with the objectives for GDA.

1.3 Methodology

8. The methodology for my assessment follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (Ref. 4).

9. My assessment was undertaken in accordance with the requirements of ONR's How2 Business Management System (BMS). ONR's Safety Assessment Principles (Ref. 2) together with supporting Technical Assessment Guides (TAG) (Ref. 5), were used as the basis for my assessment. Further details are provided in Section 2. The outputs from my assessment are consistent with ONR's GDA Guidance to RPs (Ref. 1).

2 ASSESSMENT STRATEGY

10. The strategy for my assessment of the structural integrity aspects of the UK HPR1000 design and safety case is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

11. Within Step 3 of GDA, ONR SI specialists focused on the important claims for structural integrity established in Step 2 (Ref. 6), with a greater emphasis on understanding the supporting arguments for these claims. The ONR GDA SI Step 3 assessment (Ref. 7) took key points from Step 2 (Ref. 6) based on a broad sampling approach, with more depth in some areas to gain an understanding of the RP's reasoning (arguments) and also to gain further familiarity with aspects of the UK HPR1000 design.

12. In preparing the scope and content for the ONR GDA SI Step 4 assessment plan, provided in ONR-GDA-UKHPR1000-AP-19-006 (Ref. 8), several technical themes important to demonstrating a robust structural integrity safety case were identified in accordance with ONR technical assessment guidance (Ref. 2) (Ref. 5) and the Step 3 conclusions. This resulted in several key focus areas for Step 4 of GDA, informed by structural integrity topics covered in previous GDAs.

13. In accordance with ONR-GDA-GD-001 (Ref. 1), the purpose of Step 4 of the GDA is to conduct an in-depth assessment of the safety case evidence. The general intention of this step is to move from the safety arguments and system level assessment completed in Step 3 of GDA, to a fully detailed examination of the available evidence, on a sampling basis, provided in the safety submissions.

14. I consider all of the main safety submissions within the remit of my assessment scope, to various degrees of breadth and depth. A particular focus of my assessment has been the RQs and ROs I raised as a result of my on-going assessment, and the resolution thereof.

2.2 Sampling Strategy

15. On completion of the GDA Step 3 assessment, several strengths and areas for improvement were identified. These were used to inform the GDA Step 4 assessment strategy (Ref. 8). In line with ONR's guidance in the SAPs (Ref. 2) and TAGs (Ref. 5), I chose a sample of the RP's submissions to undertake my assessment on those aspects that I judged to have the greatest safety significance, or where the hazards appeared least well controlled. My assessment was also influenced by the claims made by the RP, my previous experience of similar systems for reactors and other nuclear facilities, and any identified gaps raised during Step 3 of GDA related to the original submissions made by the RP. From these inputs, I identified a number of key themes that, in my opinion, are fundamental inputs for demonstrating a robust structural integrity safety case.

2.2.1 Traceability of Claims, Arguments and Evidence within the Structural Integrity Safety Case.

16. According to ONR guidance in the SAPs (Ref. 2) paragraph 100, a safety case should explicitly set out the argument for why risks are reduced So Far As Is Reasonably Practicable (SFAIRP); and link the information or evidence necessary to show that risks are reduced SFAIRP, including what will be needed to ensure that this can be maintained over the period for which the safety case is valid. The safety case should also support claims and arguments with appropriate evidence. I note that within its safety case, the RP refers to reducing risk 'As Low As Reasonably Practicable'

(ALARP). I consider that for the purposes of this assessment, the terms SFAIRP and ALARP are interchangeable. Therefore, to avoid confusion and to maintain consistency between the RP's submissions and my own assessment expectations, I will use the terminology 'ALARP' safety case when referring to the consideration of reducing risk SFAIRP.

17. From information I gathered during the SI GDA Step 3 assessment (Ref. 7), I identified several aspects of the safety case content specifically related to difficulty in identifying where key evidence was available to support the Structural Integrity claims and arguments presented. I therefore chose to sample this topic further in Step 4 of GDA, to ensure that the safety case documentation is clear and logically structured so that the information is easily accessible to those who need to use it.

2.2.2 Structural Integrity Classification

18. The starting point for a structural integrity assessment is the categorisation of functions (ECS.1) and the safety classification of Structures, Systems and Components (SSC) (ECS.2). The categorisation of the safety function and the safety classification of the SSC determine the requirements for design, manufacture, construction, installation, operation, monitoring, inspection, maintenance and testing. I therefore chose to sample the RP's approach to structural integrity classification of SSCs and the identification of those structures and components that form a principal means of ensuring nuclear safety. This included examples where the likelihood of gross failure is claimed to be so low that the consequences of gross failure can be discounted, i.e. highest reliability structures and components, typically including the Reactor Pressure Vessel (RPV). In particular, I wanted to confirm there exists clear linkage between Structural Integrity classification and the UK HPR1000 plant categorisation of safety functions and classification for SSC, informed by direct and in-direct consequences analyses.

2.2.3 Avoidance of Fracture Demonstration

19. A case that claims gross failure is so remote it may be discounted carries a high burden of proof (arguments and evidence). Such a case cannot be made by simple assertion of the robustness of an SSC alone. ONR guidance (Ref. 2) provides the regulatory expectations for a case where a high reliability claim is made. ONR SAP EMC.1 (Ref. 2) explains:

"The safety case should be especially robust and the corresponding assessment suitably demanding, in order that a properly informed engineering judgement can be made that:

(a) the metal component or structure is as defect-free as possible; and

(b) the metal component or structure is tolerant of defects".

20. To assess this important aspect of the RP's SI safety case, I chose to review the RP's approach to producing an avoidance of fracture demonstration (AOFD) including defect tolerance assessment (DTA), the development of part technical justifications, the strategy to justify material properties and reconciliation processes.

2.2.4 Structural Integrity Provisions, Design Codes and Standards

21. To demonstrate that structures meet their safety functional requirements, it is necessary to establish that sound design concepts, rules, standards, methodologies and proven design features have been used, and that the design is robust. The design requirements depend on the safety classification of the SSC.

22. To sample these aspects of the RP's SI safety case, I chose to assess a sample of the RP's design philosophy with respect to the UK HPR1000 design features/specifications, use of Operational Experience (OPEX) and selection/application of codes and standards covering the range of structural integrity classifications and how these provisions ensure risks are ALARP. In particular, I have reviewed compliance with appropriate design codes and standards through outputs from analyses e.g. basic sizing of pressure boundary and operating limits, with the higher safety significant components.

2.2.5 Material Selection, Testing and Surveillance

23. The SI safety case claims need to be underwritten by robust evidence, which in turn is reliant on the demonstration of high-quality manufacture and through life integrity of the components. Material specifications, manufacturing processes and inspections should be suitable and ensure that the SSC is free from significant defects and tolerant of any remaining defects (ONR SAPs EMC.5, EMC.6 & ECS.3 with paragraph 169, Ref. 2).

24. I have therefore sampled aspects of the RP's safety case to confirm that it has developed and applied suitable processes for material selection, specification and manufacture of SSCs, along with mitigation measures to reduce risk, taking cognisance of aging and degradation mechanisms and relevant OPEX. I have also chosen to sample the RP's approach to material testing and surveillance strategies with the emphasis on highest reliability but with consideration of lower structural integrity classes.

2.2.6 Inspection

25. To ensure that claims made that components are free from and tolerant of defects are robust, I have sampled the RP's consideration of design for inspectability in the UK HPR1000. This has included a review of the inspections proposed to be performed during manufacture. The detailed scope of the pre-service and in-service inspections (PSI/ISI) is outside of the scope of the GDA. However, from my engagements during Step 3 of GDA, I identified some important principles that I consider reasonable to address in advance of any site-specific phases. I have therefore raised these for discussion with the RP within Step 4 of GDA, as detailed under Section 4.6.2 below.

2.2.7 Demonstration that Relevant Risks Have Been Reduced to ALARP

26. I have sampled a variety of aspects presented within the SI safety case to determine whether the judgements made by the RP align with ONR expectations for reducing risks to ALARP. My sampling approach has been used to identify any weaknesses in the proposed facility design and operation, identify where improvements were considered and understand how risk assessments have been used to demonstrate that safety is not unduly reliant on a small set of particular safety features.

2.2.8 Consolidated Safety Case

27. I have therefore sampled several documents of the SI safety case, to ensure that improvements or conclusions drawn from lower-tier sources of evidence that have been modified or updated during GDA are fully represented in the higher-level safety case document.

2.3 Out of Scope Items

28. All components of the UK HPR1000 primary circuit pressure boundary, and some sections of the secondary circuit pressure boundary located within the Nuclear Island (NI) are considered to be within scope, as is the notion of design for inspectability. This means that GDA is expected to provide assurance that the RP's approach and design

facilitates pre-service and in-service inspections, and I have assessed these at a principles level.

29. However, the detailed scope of in-service inspection, maintenance of systems through life and detailed record keeping are typically addressed during site-specific phases, and as such, in-depth assessment is out of the scope of GDA.

30. Integrity of the secondary circuit steam or feedwater systems associated with the Conventional Island (CI) structures have not been explicitly considered on the grounds of reduced nuclear safety significance. A few exceptions to this exist where the risk of consequences of failure (direct and indirect) may challenge integrity claims for SSCs located in the in the Nuclear Island (NI).

2.4 Standards and Criteria

31. The relevant standards and criteria adopted within this assessment are principally the ONR SAPs (Ref. 2), TAGs (Ref. 5), relevant national and international standards, and relevant good practice informed from existing practices adopted on nuclear licensed sites in Great Britain. The key SAPs and any relevant TAGs, national and international standards and guidance are detailed within this section. Relevant good practice (RGP), where applicable, is cited within the body of the assessment.

2.4.1 Safety Assessment Principles

32. The ONR SAPs (Ref. 2) constitute the regulatory principles against which ONR judge the adequacy of safety cases. The SAPs applicable to Structural Integrity are included within Annex 1 of this report.

33. The key ONR SAPs applied within my assessment were EMC.1-34 and those from the ECS, EKP, EAD and SC series, as applicable and referenced within the body of my assessment.

2.4.2 Technical Assessment Guides

34. The following TAGs were used as part of this assessment (Ref. 5):

- NS-TAST-GD-005, Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable).
- NS-TAST-GD-051, The Purpose, Scope and Content of Safety Cases.
- NS-TAST-GD-016, Integrity of Metal Structures, Systems and Components.
- NS-TAST-GD-094 Categorisation of Safety Functions and Classification of Structures, Systems and Components.

2.4.3 National and International Standards and Guidance

35. The following standards and guidance were used as part of this assessment:

- Relevant International Atomic Energy Agency (IAEA) standards:
    - IAEA, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, No.SSG-30, May 2014 (Ref. 9).
    - IAEA, Safety of Nuclear Power Plants: Design. Safety Requirements, Safety Standards Series NS-R-1, 2000, (Ref. 10).
    - The relevant guidance from IAEA standards as discussed in Appendix A2 of ONR-TAST-GD-016 (Ref. 5).
- Western European Nuclear Regulators' Association (WENRA) references:

- The relevant guidance from WENRA (Ref. 11) reference levels as discussed in Appendix A1 of ONR-TAST-GD-016 (Ref. 5).

■ Other national standards:

- R6 – Assessment of the Integrity of Structures Containing Defects, Revision 4, EDF Energy Nuclear Generation Ltd. (Ref. 12).

■ Other international standards:

- The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Sections III and XI (Ref. 13).
- RCC-M. Design and Construction Rules for Mechanical Components of PWR Nuclear Islands. 2007 Edition. Published by the French Association for Design, Construction and In-Service Inspection Rules for Nuclear Island Components – AFCEN, Paris (Ref. 14).
- RSE-M. In-Service Inspection Rules for Mechanical Components of PWR Nuclear Islands, RSE-M, 2010 edition+2012 addendum, 2010, 2012, AFCEN (Ref. 15).
- European Methodology for Qualification of Non-Destructive Testing. Third Issue. ENIQ Report No. 31 EUR 22906 EN. August 2007 (Ref. 16).
- ENIQ Recommended Practice 2. Strategy and Recommended Contents for Technical Justifications, Issue 2. ENIQ Report No.39. EUR 24111EN-2010. June 2010 (Ref. 17).

2.5 Use of Technical Support Contractors

36. It is usual in GDA for ONR to use Technical Support Contractors (TSCs) to provide access to independent advice and experience, analysis techniques and models, and to enable ONR‘s inspectors to focus on regulatory decision making.

37. Table 1 below sets out the areas in which I used TSCs to support my assessment. I required this support to provide additional assessment capacity and access to independent specialist advice and experience on technical topics fundamental to the assessment of the RP's safety case and supporting evidence. The TSC was required to provide an independent technical review of a sample of the claims, arguments and evidence presented by the RP in relation to the structural integrity of the UK HPR1000. For the information sampled, the TSC provided a view on whether any assumptions made by the RP were appropriate, and whether the SI provisions were commensurate with reducing risks to ALARP. For the three technical areas described in Table 1, the TSC was required to make recommendations to ONR where the RP's approach may or may not satisfy RGP and/or where further justification may be required by the RP.

Table 1: Work Packages Undertaken by the TSC

Number	Description
1	Review of Compliance with Design Requirements of Recognised Codes. (Ref. 18).
2	Review of Defect Tolerance Assessment & Undertake Comparative R6 Calculations (Ref. 19).

Number	Description
3	Review of Component Safety Reports and Supporting Documentation (Ref. 20).

38. Whilst the TSC undertook detailed technical reviews, this was done under my direction and close supervision. The regulatory judgment on the adequacy, or otherwise, of the generic UK HPR1000 safety case in this report has been made exclusively by ONR.

2.6 Integration with Other Assessment Topics

39. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot be carried out in isolation as there are often aspects of the assessment that span multiple disciplines. I have therefore worked closely with a number of other ONR inspectors to inform my assessment. The key interactions were:

- Mechanical Engineering – I have engaged with the Mechanical Engineering specialists to assess the completeness of the safety case documentation as presented in the system design manuals which relate to structural integrity claims.
- Fault Studies – I have engaged with the Fault Studies specialists to discuss and assess the severity of the consequences of failure for key SSCs as part of the structural integrity classification work and ALARP basis of key design features.
- Fuel and Core – I have engaged with the Fuel & Core specialists to discuss and assess the severity of the consequences of failure for key SSCs as part of the structural integrity classification work and the ALARP basis of key design features.
- Internal Hazards – I have engaged with the Internal Hazards specialists to discuss and assess the severity of the indirect consequences of failure for key SSCs as part of the structural integrity classification work and the ALARP basis of key design features.
- Civil Engineering – I have engaged with Civil Engineering specialists to discuss the integrity claims for civil structures and the subsequent SI classification of certain SSCs along with the materials selection for the containment liner.
- Chemistry – I have engaged with the Chemistry specialists to discuss and assess the RP’s development of material selection strategies and the ALARP basis for material selection.
- Management for Safety & Quality Assurance (MSQA) – I have engaged with the MSQA specialists to discuss, where appropriate, compliance with SI related procedures and the principles for control of design and manufacture.
- Severe Accident Analyses (SAA) - I have engaged with the SAA specialist to discuss the RP’s integrity claims for the RPV under severe accident conditions, particularly the in-vessel retention (IVR) safety feature of the UK HPR1000.
- Electrical Engineering – I have engaged with the Electrical Engineering specialist to determine what impact the RP’s design for meeting UK Grid Code compliance would have on the input criteria for structural integrity assessments, particularly regarding load transient frequencies and magnitudes used in demonstrating avoidance of fracture arguments.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Introduction to the Generic UK HPR1000 Design

40. The generic UK HPR1000 design is described in detail in the PCSR (Ref. 3). It is a three-loop PWR designed by CGN using the Chinese Hualong technology. The generic UK HPR1000 design has evolved from reactors that have been constructed and operated in China since the late 1980s, including the M310 design used at Daya Bay and Ling'ao (Units 1 and 2), the CPR1000, the CPR1000$^+$ and the more recent ACPR1000. The first two units of CGN's HPR1000, Fangchenggang Nuclear Power Plant (NPP) Units 3 and 4, are under construction in China and Unit 3 is the reference plant for the generic UK HPR1000 design. The design is claimed to have a lifetime of at least 60 years and has a nominal electric output of 1,180 Megawatts (MW).

41. The reactor core contains zirconium clad uranium dioxide (UO_2) fuel assemblies and reactivity is controlled by a combination of control rods, soluble boron in the coolant and burnable poisons within the fuel. The core is contained within a steel Reactor Pressure Vessel (RPV) which is connected to the key primary circuit components by the Main Coolant Lines (MCL), including the Reactor Coolant Pumps (RCP), Steam Generators (SG) and Pressuriser (PZR), in a three-loop configuration. The secondary side of the SGs is connected to the power turbines via the Main Steam Lines (MSL), which pass through the containment structure and safeguards building to the steam consumers (power turbines). The design also includes a number of auxiliary systems that allow normal operation of the plant, as well as active and passive safety systems to provide protection in the case of faults, all contained within a number of dedicated buildings.

42. The reactor building houses the reactor and primary circuit and is based on a double-walled containment with a large free volume. Three separate safeguard buildings surround the reactor building and house key safety systems and the main control room. The fuel building is also adjacent to the reactor, and contains the fuel handling and short-term storage facilities. Finally, the nuclear auxiliary building contains a number of systems that support operation of the reactor. In combination with the diesel, personnel access and equipment access buildings, these constitute the nuclear island for the generic UK HPR1000 design.

43. Figure 1 below shows some of the key primary circuit components of the generic UK HPR1000 that are housed within the Containment Building.

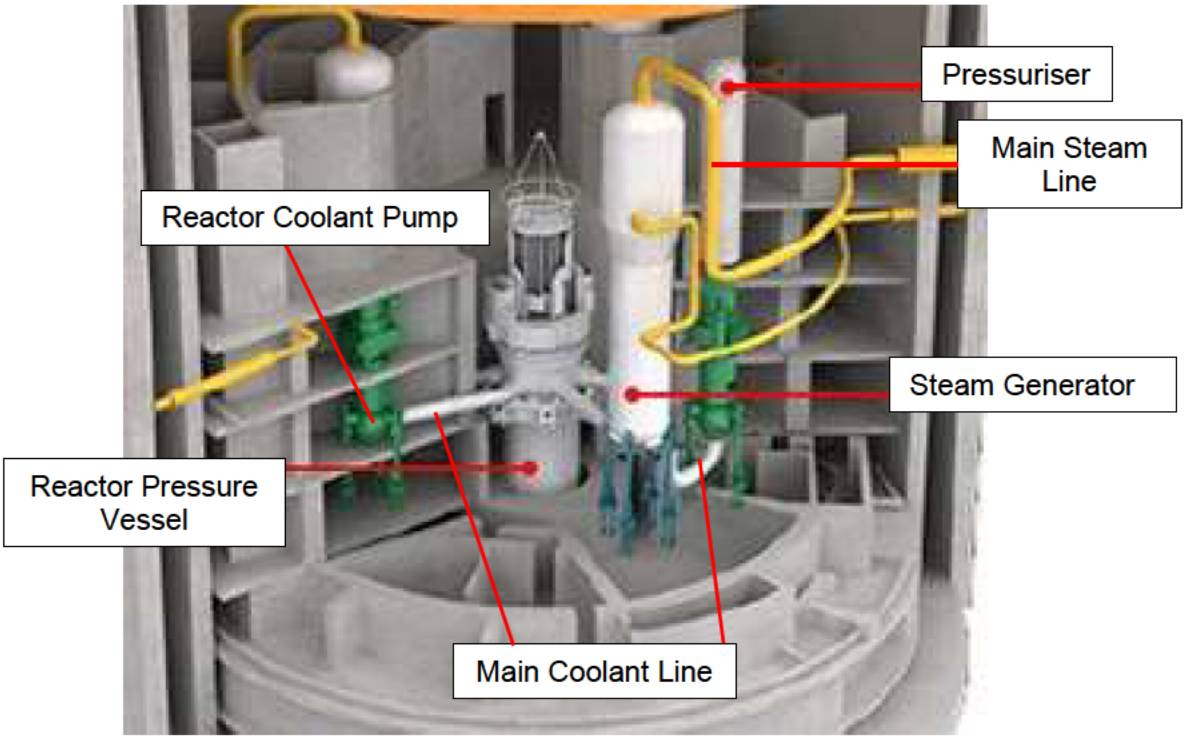

Figure 1: Diagram showing some of the key UK HPR1000 primary circuit metal components housed within the Containment Building (Ref. 6).

3.2 The Generic UK HPR1000 Structural Integrity Safety Case

44. This section provides an overview of the Structural Integrity aspects of the generic UK HPR1000 safety case as provided by the RP during GDA. Details of the technical content of the documentation and my assessment of its adequacy are reported in the subsequent sections of my report.

45. The main sections of the PCSR that are of interest to Structural Integrity are as follows:

- Pre Construction Safety Report - Chapter 17 Structural Integrity (Ref. 3)
- Pre Construction Safety Report - Chapter 6 - Reactor Coolant System (Ref. 21)
- Pre Construction Safety Report - Chapter 7 - Safety Systems (Ref. 22)
- Pre Construction Safety Report - Chapter 11 – Steam and Power Conversion System (Ref. 23)

46. These documents form the top level of the safety demonstration (tier one) under which there are two more tiers. In tier two, overarching documentation provides the link between the claims and arguments within the PCSR chapter and the documents presenting the evidence in tier three.

47. At the beginning of Step 3 of GDA, the RP submitted a PCSR (Ref. 24) and supporting references, which outline the nuclear safety case for the UK HPR1000. This was supplemented during my assessment with further submissions, including responses to my regulatory questions.

48. This section presents a high-level summary of the RP's case and identifies the main documents which formed the basis of my assessment.

49. The structural integrity claims that are presented in the PCSR (Ref. 24) are as follows:

- 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.

50. A summary of the main SSCs that rely on structural integrity justification are documented in PSCR Chapters 6 (Ref. 21), 7 (Ref. 22) and 11 (Ref. 23). A summary of the structural integrity justification is provided in PSCR Chapter 17 (Ref. 24). It is stated in PCSR Chapter 17 that the structural integrity justification supports the following level 1 claim:

- Claim 3: 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).

51. Specifically, the structural integrity justification supports the following level 2 claims:

- 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.

52. The RP has divided these level 2 claims further in to level 3 claims and those that are relevant to the structural integrity technical areas are as follows:

- 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.

53. The overarching Claim of PSCR Chapter 17 is as follows:

"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)."

54. The RP's methodology is informed by the UK Technical Advisory Group on Structural Integrity (TAGSI) approach to demonstrating structural integrity of SSCs, based on offering arguments in four legs, underpinned with appropriate and relevant evidence. The requirement of each TAGSI leg is presented as below.

- Leg 1: Design and manufacture based on the existing engineering (manufacturing, construction and operation) experience and proven track records. For this leg the RP intends to provide arguments and evidence to demonstrate the high reliability and avoidance of defects to support the claim of good design and manufacture.
- Leg 2: Functional testing. For this leg, the RP intends to provide arguments and evidence to demonstrate the functionality of the component through testing such as the proof pressure test.
- Leg 3: Failure Analysis. For this leg, the RP intends to provide arguments and evidence to demonstrate tolerance to defects and through-life degradation mechanisms over the design life of the plant using the current best scientific understanding.
- Leg 4: Forewarning of Failure. For this leg, the RP intends to provide arguments and evidence to demonstrate that in-service inspection, leakage monitoring, transient monitoring, irradiation surveillance etc. will be in place,

and commit to take future actions on gaining new information to effectively forewarn of failure.

55. This four leg approach has been considered by the RP and used to construct the basis of the RP's SI safety cases, specifically for pressure vessels/pressure boundary components.

56. The guidance focuses on the development of the central document for the RP's SSC-specific SI safety case, the component safety reports (CSRs). Information is provided on how the structure of these CSRs links to the overall UK HPR1000 safety case, from the fundamental objective, through to the sub-claims and evidence needed for a robust SI safety case.

57. These are presented in a series of comprehensive hierarchical document structure diagrams, clearly differentiating between the requirements for a highest reliability safety case. The documents provide a brief description of each sub-claim, argument and evidence required and how these align with the TAGSI recommended approach of presenting a multi-leg safety case to satisfy SI safety expectations. I note that the RP's approach has adopted a three-leg structure, using the following subclaims below to substantiate the Chapter 17 claim (see Paragraph 53-54 above):

- 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.

58. The RP's approach is similar to the TAGSI approach referenced and is noted to have combined legs 1 and 2.

4 ONR ASSESSMENT

59. In completing my assessment, I have followed guidance on the identification and management of GDA issues, assessment findings and minor shortfalls (Ref. 25).

4.1 Traceability of Claims, Arguments and Evidence within the Structural Integrity Safety Case.

60. In Step 3 of GDA, the ONR SI assessor explored the strategy and structure of the SI safety case. This was considered appropriate in advance of the presentation of detailed evidence during Step 4 of GDA, in particular to understand the linkage between the top tier PCSR Chapter 17 and the supporting SI documentation.

61. The RP was acknowledged as being responsive to challenges presented and took action to progress development of the safety case in accordance with UK expectations. Several matters were identified for follow up during Step 4 of GDA, including the need to sample the structure and content of the safety case in terms of claims, arguments and the development of evidence to underwrite both highest reliability and other structural integrity classifications (Ref. 8). System Design Manuals (SDMs) were also identified as aspects of the safety case for further sampling, to ensure that structural integrity requirements are clearly identified and referenced.

62. The RP agreed to improve the traceability in document updates, with ONR identifying further work necessary in Step 4 of GDA, to confirm that the updates addressed the concerns raised.

4.1.1 Assessment

63. It is important that the SI safety case is easily interrogated to understand how claims, arguments and evidence work together to ensure the risks of UK HPR1000 operation have been reduced to ALARP.

64. ONR's expectations on the structure of a safety case are outlined in the ONR SAPs paragraphs 96 & 97 (Ref. 2):

> "The process for producing safety cases should take into account the needs of those who will use the safety case to ensure safe operations. It is essential that the safety case documentation is clear and logically structured so that the information is easily accessible to those who need to use it (paragraph 87, Ref. 2). This includes designers, operations and maintenance staff, technical personnel and managers who are accountable for safety.
>
> The safety case process should also take into account how the different levels and types of documentation fit together to cover the full scope and content of the safety case. The needs of users should be addressed by ensuring that all descriptions and terms are easy to understand by the prime audience, all arguments are cogent and coherently developed, all references are easily accessible, and that all conclusions are fully supported, and follow logically from the arguments. The trail from claims through argument to evidence should be clear."

65. Within this section of my assessment, I have focused on the traceability of evidence within the SI safety case, whereas Sub-section 4.2 addresses the classification aspects.

4.1.2 Assessment of Structural Integrity SI Safety Case Structure

66. ONR concluded in Step 3 of GDA that the RP has developed an appropriate structure for the SI safety case presented in Chapter 17 of the PCSR. However, the linkage to the supporting documentation required further explanation. A good structure and

traceability are essential for evolving documentation. If the traceability is clear, then the impact of modifications to the design or supporting analyses can be readily understood and fully evaluated.

67. A description of the RP's SI safety case structure is discussed in Sub-section 3.2 above. The scope of Chapter 17 is to substantiate the SI of all metal components and structures that are important to nuclear safety during the design lifetime for all conditions within the design basis. Chapter 17 is structured and signposted in an appropriate way, referencing supporting documents in a way that is understandable.

68. Within GDA, and recognising the significance of the claims, the focus of Chapter 17 has been upon the components of the highest reliability, designated as HIC by the RP. For structural integrity classes below HIC, a sampling approach has been taken, with the assessment informed by their nuclear safety function. I judged that this is appropriate.

69. The linkages from Chapter 17 to other chapters of the PCSR have been made in a clear, tabulated form.

70. Chapter 17 clearly identifies the codes and standards selected by the RP that are relevant for demonstrating the structural integrity of the safety significant components. Chapter 17 provides a clear link to show that two separate versions of the ASME boiler and pressure vessel code have been used: 2007 for the steam generators and 2017 for the reactor coolant pump and steel containment liner. ONR's views on the appropriateness of using different versions of codes, noting especially that the main design of the nuclear steam raising plant is to RCC-M, are discussed under Sub-section 4.4 below. For the purposes of this section of my assessment, I consider that the codes and standards used are clearly referenced and traceable in the safety case.

71. The links back to the safety functional requirements are made in Sub-section 17.4 of Chapter 17 (Ref. 3). This is presented at a high level only, with consideration of safety functional requirements of individual SSCs accounted for within the structural integrity classification process, which is presented in Sub-section 17.5 of Chapter 17. A description of the SI classifications is presented under Table 11.5-1 (Ref. 3). These range from HIC (highest SI classification) through to SIC-3 (lowest SI classification, see Sub-section 4.2 below) and are clearly referenced with respect to the methodology used and the supporting evidence for each of the components presented within GDA. I was satisfied that there was a clear link between Chapter 17 and the supporting classification methodology, with traceability to the assessment evidence used to determine the SI classifications of the key components. The adequacy of the SI classification process is discussed in Sub-section 4.2 below.

72. A high-level overview of the construction of a structural integrity safety case for components has been presented within the PCSR Chapter 17. This is briefly explained and linked back to the RP's 'Safety Case Methodology for HIC and SIC Components' (Ref. 26). The methodology presented (Ref. 26) details how the relevant Claims, Arguments and Evidence presented in Chapter 17 are addressed on a component-by-component basis, the content of which is commensurate with SI classification. The adequacy of the supporting component-specific safety documents (Component Safety Reports – CSRs) with respect to traceability of evidence is discussed below (Sub-section 4.1.2). For the perspective of linking the top-tier SI safety case (Chapter 17) to the next tier of safety demonstration (CSRs), I was satisfied that the RP has demonstrated clear traceability.

73. Overall, I judge that the structure of the SI safety case enables the user to clearly trace the safety claims between the top tier (Chapter 17) of the SI safety case and the supporting documents. This reflects the hierarchical structure of the SI safety case

and does not prejudice any discussions elsewhere in this report which relate to more detailed evidence (Sub-section 4.1.4).

4.1.3 Assessment of Consideration and Traceability of ALARP in Chapter 17

74. The ALARP Assessment for SI has been presented in Sub-section 17.8 of Chapter 17 (Ref. 3). This references Chapter 33 of the PCSR to provide the RP's generic approach and structure of a safety case for demonstration of ALARP.

75. The RP describes how the ALARP positions of each of the highest reliability components have been considered. This is at a high level, and links back to the individual considerations presented in subordinate reports, such as inspection qualification, materials selection and use of applicable design codes. Chapter 17, Sub-section 17.8 links to the supporting document 'ALARP Demonstration Report of PCSR Chapter 17' (Ref. 27). Within this, the reader can trace the process completed by the RP to identify a number of 'gaps' against OPEX and RGP, for which further work has been completed to demonstrate ALARP. This has resulted in the RP proposing a number of modifications to the design of the generic UK HPR1000.

76. The catalogue of reasonable modifications made to the reactor plant to meet the UK expectations of maintaining risks ALARP is presented in Table 11.8-1 (Ref. 3). This provides a clear link to the evidence used to support risk reduction resulting in design modifications. I judge this to be an adequate demonstration of traceability for the review and implementation of findings from the RP's ALARP assessment. The individual ALARP assessments made on a component-by-component basis have been assessed as a separate consideration in Sub-section 4.7 of my report.

77. In RQ-UKHPR1000-0661 (Ref. 28), I sought clarification from the RP on how traceability is achieved through the SI claims presented in PCSR Chapter 17 and the subsequent arguments and evidence presented within supporting documentation.

78. The RP provided the submission 'Safety Case Methodology for HIC and SIC Components' (Ref. 26), which contains details on how the structure of SI safety cases would be presented and how the RP has developed a series of corresponding supporting Tier 2 and 3 documents to provide the arguments and evidence for key safety significant SSCs. The hierarchy of supporting SI documents, i.e. the safety case route map in the format of Claims, Arguments and Evidence (CAE), was presented in the submission 'Production Strategy for Structural Integrity' (Ref. 29).

79. The scope and depth of evidence presented is commensurate with the level of safety demonstration being claimed. The main difference being that, for SIC claims, there is no requirement for the RP to perform an avoidance of fracture demonstration. This means that achieving compliance with a relevant nuclear design code is generally considered by the RP as being sufficient for SICs.

80. In my opinion from my review of the RP's 'Safety Case Methodology for HIC and SIC Components' (Ref. 26), the RP has developed an adequate approach to constructing a robust SI safety case. The methodology developed encompasses the key SI expectations presented within ONR SAPs EMC 1-34, which expresses ONR's expectations for claims of the highest reliability. The RP's approach also recognises and makes provision for adopting a depth of assessment proportionate to the safety significance of the component, which is also an expectation of the ONR.

4.1.4 Assessment of Implementation and Traceability of the Safety Case Methodology

81. Whilst I am broadly satisfied that the RP's methodology for constructing an SI safety case aligns with ONR expectations, I considered it prudent to sample how this methodology was implemented for the UK HPR1000. I initially engaged with the RP

during Step 3 of GDA to identify SI significant SSCs, for which evidence supporting the SI safety case would be produced within GDA.

82. The RP provided a sample of Tier 2 and 3 documents to demonstrate how its 'Safety Case Methodology for SI Components' (Ref. 26) is applied to the UK HPR1000. These included submissions for all of the identified UK HPR1000 HICs and a selection of SIC-1, -2 and -3 components. Details of these components and the respective SI classification is provided in Table 2 below.

Table 2: List of SSCs selected by the RP to demonstrate SI CAE safety case structure and associated SI safety classification within GDA.

SSC	SI Classification
RPV	HIC
PZR	
SG	
RCP	
MCL	
MSL	
Reactor Vessel Internals (RVI)	SIC-1
Surge Line (SL)	SIC-1
Accumulator (ACC)	SIC-2
Surge Tank (ST)	SIC-3

83. During Step 3 of GDA, I sampled the information presented for the RPV to understand the extent of traceability and visibility of safety claims and arguments within lower tier documents. I raised RQ-UKHPR1000-0218 (Ref. 28), which contained a number of questions related to a specific CSR, namely the RPV because this represented a HIC.

84. The RP stated that the route map for the claims and arguments is presented within PCSR Chapter 17. Whilst I noted that a route map was presented in this top tier document, it was my opinion that the traceability and accountability within lower tier documentation back up to the top tier document could be improved. I noted in subsequent samples taken from Tier 2 and 3 documents for other SSCs, that there was a lack of clear identification of which claims, arguments and supporting information are supported by the documents presented. The RP acknowledged my observation and stated that as part of the general updates to the safety case Tier 2 and 3 documents, it would look to improve the traceability of safety case claims, namely within the CSRs.

85. To ensure that the areas for improvement identified in my review had been addressed, I sampled a number of the components listed in Table 2 above. This was mostly undertaken with TSC support under Work Package 3 (WP3), as described in Sub-section 2.5 above. In addition to the scope of WP3 (Ref. 20), I chose to review the

traceability of claims, arguments and evidence presented for the accumulators (ACCs) and the surge tanks (STs).

86. In addition, I also chose to sample the traceability of evidence presented for SI claims in the UK HPR1000 system design manuals.

4.1.5 Review of Component Safety Reports

87. The purpose of the CSR is to provide a high-level statement of the safety claims relating to structural integrity, articulating the supporting safety arguments and summarising the available evidence to support those claims. The scope of the CSR should address all aspects of the SI safety case, including design, manufacture, inspection and operation, though the completeness of the evidence in the individual sections should reflect the stage in the development of the safety case. The CSRs should also address design basis loads, including faults and hazards, defect tolerance (for HICs) and degradation mechanisms.

88. I consider the CSRs to be an integral part of the SI safety case, which I chose to sample for several safety significant SSCs. In addition to reviewing individual CSRs, I assessed some of the processes, procedures and ways of working that underpin the RP's assessments and safety case. A sample of the CSRs were reviewed under WP3 with the results presented in a summary report (Ref. 20).

89. The following SSCs were subject to an initial review:

- RPV (Ref. 30)
- PZR (Ref. 31) and (Ref. 32)
- RCP (Ref. 33)
- MSL (Ref. 34)
- Surge Line (Ref. 35)
- Accumulators (Ref. 36)
- Surge Tank (Ref. 37)

90. A brief overview and summary of the key points for each of the sampled CSRs is provided below.

4.1.5.1 RPV Assessment Summary

91. From the review of the RPV CSR, the main questions and clarifications raised were associated with derivation/provenance of design loads and derivation of safety functional requirements with relevance to structural integrity. All of the questions raised in RQ-UKHPR1000-1368 and RQ-UKHPR1000-1642 (Ref. 28) were satisfactorily resolved by the RP, however some generic aspects relevant to the review of the CSRs were raised and explored further through more focused reviews, which are discussed below. One notable finding was that that the Safety Functional Requirements (SFRs) listed in the CSR are not clearly linked to their derivation. ONR guidance SC.6 (Ref. 2) states that "the safety case for a facility or site should identify the important aspects of operation and management required for maintaining safety and how these will be implemented". Information related to the safety functional requirements is presented later in the CSR under the classification of components, however the reader has to make this link back to understand the SFRs.

92. I consider the absence of a direct link from the safety functional requirements directing the reader to the importance of maintaining integrity as a minor shortfall. This is indicative of a generic shortfall against the expectations of ONR SAP SC.2 (Ref. 2), that the safety case documentation should be clear and logically structured so that the information is easily accessible to those who need to use it.

4.1.5.2 PZR Assessment Summary

93. From the review of the PZR CSR, a number of questions and clarifications were raised in RQ-UKHPR1000-1313, RQ-UKHPR1000-1464, RQ-UKHPR1000-1465 and RQ-UKHPR1000-1731, the majority of which were satisfactorily resolved by the RP (Ref. 28).

94. One of the key lines of questioning related to the classification of the PZR manway cover and retaining studs, for which the traceability of evidence supporting the classification wasn't immediately clear. This was explored further under Sub-section 4.2 'SI Classification' below.

95. Another observation made is that the CSR lacked a coherent narrative of how the design basis loads were derived. The Design Specification only lists load values, with no clear link to the Transient Specification. The design load values or their frequencies of occurrence are not being disputed, the concern relates to the traceability of their origin so that the significance of a change and any implications for the safety case can be readily understood. This point was identified as a generic theme.

4.1.5.3 RCP Assessment Summary

96. From the review of the RCP CSR, several questions and clarifications were raised in line with generic themes identified for other CSRs sampled, which related to traceability of design loads and safety functional requirements. All of the questions raised in RQ-UKHPR1000-1515 were satisfactorily resolved by the RP (Ref. 28).

4.1.5.4 Surge Line Assessment Summary

97. Questions and clarifications raised for the SL CSR in RQ-UKHPR1000-1314 were satisfactorily resolved by the RP (Ref. 28). While the SL is designated a SIC-1 classification, the CSR is similar in structure and content to that of a HIC (noting that defect tolerance assessments are not provided for the SL as it is a SIC-1 component). As per previous CSR reviews, a number of emerging generic topics were identified, including traceability of design loads and safety functional requirements relevant to structural integrity.

4.1.5.5 Accumulator Assessment Summary

98. From my review of the ACC CSR, the CAE are linked to the structure expected for a SIC component. I was broadly satisfied that the methodology presented in Ref. 26 had been followed for the ACC CSR, however I identified an area of ambiguity that required further clarification regarding traceability of evidence.

99. I was unable to identify why the ACC had been classified as SIC-2 items, as the RP's 'Method and Requirements of Structural Integrity Classification' had indicated a SIC-3 classification. I raised RQ-UKHPR1000-1620 (Ref. 28) to pursue this query. The RP's response explained this was due to the consideration of OPEX, which has affected the classification. I was satisfied that from the perspective of traceability of evidence, this was not a shortfall. However, the relationship between SI classification and the design code class (pressure equipment) required further assessment and is discussed in more detail under Sub-section 4.2 below.

4.1.5.6 Surge Tank Assessment Summary

100. I reviewed the surge tank CSR (Ref. 37) to determine whether the RP's approach for a lower safety classification component was consistent with the methodology presented in Ref. 26.

101. Sub-section 2.3 of the CSR identifies the safety functional requirements as being to provide “supplementary water volume changes during the Component Cooling Water System (CCWS) operation”. I did not consider this provided much information as to how significant this role was for safety, so I reviewed the relevant CCWS SDM to seek further information on the surge tank safety functional requirements.

102. I reviewed ‘CCWS Design Manual Chapter 3 System Functions and Design Bases’ (Ref. 38), where the safety functions are listed under Sub-section 3.1. From here, I was able to deduce that the surge tanks do not provide a direct safety function. However, they do appear to be necessary for normal CCWS operation, which provides ‘extra supporting functions’ for backup safety systems such as cooling the safety injection pumps, containment heat removal system and primary pump thermal barriers.

103. Sub-section 3.2.4 of the CCWS SDM states that “the structural integrity classification shall also be considered in CCWS, the components whose failure is intolerable and for which no protection is provided or protection provision is not reasonably practicable are assigned to High Integrity Component (HIC). According to the effect of equipment and structure failure on the core, the equipment and structure can be assigned to Structural Integrity Class 1, Class 2 and Class 3. Currently, the structural integrity classification of CCWS is still in progress and the information will be supplemented later”. There does not appear to have been a revised version of this document since June 2019, which I consider to be a minor shortfall. The licensee should ensure that there is a clear and coherent safety case, with all relevant SI safety case documents updated, post GDA.

104. Taking cognisance of the above, I note that the CSR was produced in November 2020, and so is likely to contain more up to date information than the CCWS SDM. The CSR details the safety classification as SIC-3, in accordance with the ‘Method and Requirements of SI Classification’ (Ref. 39). For SIC-3, this document declares that “SIC-2 or SIC-3 are assigned to the components and structures whose failure does not result in core damage. These should be at least two lines of protection with diversity. The integrity claim for such components is mainly based on the compliance with appropriate design codes and standards”.

105. Sub-section 3.3.2 of Ref. 39 also states that:

> “…. an assessment on the failure consequences and the plant protection design against the failure can be performed, and in most cases the results of fault and hazard study can be used for the assessment. Nevertheless, as for those failures that have not been considered in the scope of fault and hazard study, specific assessment should be carried out to indicate the severity of failure consequences and to justify the effectiveness of protection. The level of detail of the assessment varies for different scenarios, for instance the consequences of RPV gross failure can be feasibly determined by engineering judgment while failure consequences of some tanks and pipes have to be calculated by appropriate computer codes. Once the assessment for an individual component is completed, the component can be classified according to specific criteria based on the assessment result.”

106. From my review of documents referenced in the CSR, I was unable to find any link to relevant consequence analysis to demonstrate that failure of the surge tanks was tolerable and would not result in core damage.

107. During a technical exchange meeting with the RP (Ref. 40), information was presented to explain the scope of consequence analysis that had been completed for the surge tank, with reference to a series of events that have been analysed accordingly that bound direct and indirect consequences of failure of the surge tanks. Whilst I am satisfied that the RP has followed due process, this has not been reflected in the safety case. This is an example of where the RP’s process has not been fully demonstrated within the safety case.

108. I do not consider this to be a significant burden for the licensee to address. I therefore consider it to be a minor shortfall that there is no reference within the safety case to an appropriate consequence analysis that demonstrates how direct or indirect consequences of failure are tolerable for a SIC classified system. Inclusion of this information within the CSR would help clearly define the level of safety significance for this system and the reasoning for the SIC-3 classification to the licensee/key stakeholders. By comparison, the CSR for the SIC-2 classified Accumulators (Ref. 36) clearly identifies and references the consequence analyses undertaken to substantiate the classification.

4.1.5.7 Initial Generic Points from CSR Assessments

109. From my initial sampling of the CSRs, I identified a number of generic points relating to the traceability of the supporting documentation. I consider the generic traceability of evidence specifically relating to design loads, claims, arguments and evidence and component alignment below, with generic points relating to classification and fatigue usage factors (FUF) addressed in Sub-section 4.4.

Design Loads

110. ONR SAP EMC.3 (Ref. 2) states that evidence should be provided to demonstrate that the necessary level of integrity has been achieved for the most demanding situations identified in the safety case. To satisfy this expectation, there should be evidence of a detailed loading specification covering normal operations, faults and accident conditions, which should include plant transients and internal and external hazards (Ref. 2, para 295.b).

111. ONR guidance under EMC.7 (Ref. 2) also states that “the schedule of design loadings (including combinations of loadings) for components and structures, together with conservative estimates of their frequency of occurrence should be used as the basis for design against normal operation, fault and accident conditions. This should include plant transients and tests together with internal and external hazards”. The calculation of the transients is not usually an area that SI form a view on, so I engaged with ONR Fault Studies specialists to understand if these are adequate. For SI, it is important that the RP implements adequate processes to ensure conservative estimates of loadings and combinations (ONR EMC 7, Ref. 2). I consulted the ONR Fault Studies Inspector who was content that the RP’s methods should be appropriate for determining the magnitudes used within the transient specification. In addition, ONR’s PSA Inspector was satisfied that the transient frequencies are conservative.

112. During my review of the methods for stress analyses during Step 3 of GDA, I reviewed the overall design transient specification, as presented by the RP within the ‘Design Transient Specification’ (Ref. 41). From the information that was presented, it was not clear why the number and occurrences of transients was appropriate or conservative. In addition, there was limited detail upon which to base subsequent analyses.

113. Within Step 4 of GDA, I consider this aspect of the RP’s safety case following the feedback received from the initial reviews. Several documents were reviewed to assess the derivation and provenance of design loads in the safety case. A series of queries and clarifications were raised (see Table 3)

Table 3: List of RQs raised to review design load traceability.

Identified Component	**RQ Raised** (Ref. 28)
Pressuriser CSR (Revision C)	RQ-UKHPR1000-1313 RQ-UKHPR1000-1464
Pressuriser CSR (Revision D)	RQ-UKHPR1000-1465

Identified Component	RQ Raised (Ref. 28)
Reactor Pressure Vessel CSR	RQ-UKHPR1000-1368 RQ-UKHPR1000-1642
Reactor Pressure Vessel Design Specification	RQ-UKHPR1000-1368
Reactor Cooling Pump CSR	RQ-UKHPR1000-1515
Main Coolant Line General Report of Mechanical Analysis	RQ-UKHPR1000-1148
Reactor Vessel Internals Dimension Report	RQ-UKHPR1000-1150
Generic Queries from Structural Integrity Reviews (WP1)	RQ-UKHPR1000-1548
CSR Deep-Dive	RQ-UKHPR1000-1641

114. To provide confidence that all loading scenarios have been considered in a logical and rational manner, the CSR and supporting documents should give a narrative or commentary of such, in order to minimise the potential for any unforeseen loading conditions, which may compromise the structural integrity of the plant. From the information reviewed and the RP's responses to the RQs raised in Table 3, it was observed that such a narrative is not readily presented.

115. Given the specific nature of this finding to the derivation and specification of design transients, I consider this further within my assessment of the RP's use of design code under Sub-section 4.4.2.2 below.

116. In general, I consider that there should be a clear and traceable path from the assessment of OPEX, through the identification of postulated initiating events, definition and quantification of the design transients, the derivation of inputs to the mechanical and thermo-fluid analyses and the collation and post-processing of the analysis outputs to the generation of the tabulated loads data in the relevant Design Specifications.

117. In line with guidance from the ONR SAPs under SC.2 and para 96-97 (Ref. 2), this process should be articulated by a clear narrative and supported by references to the detailed technical reports on the individual technical assessments and analyses. Only when this is in place can the reader or safety case user be confident that the process is robust and in accordance with ONR SAP SC.2 from a structural integrity perspective. I have therefore raised an assessment finding (AF) to address this topic.

AF-UKHPR1000-0006 – The licensee shall, as part of detailed design, demonstrate that the derivation of design loads is compliant with the provisions of the relevant design codes and standards. This should include, but not be limited to, the basis for design against normal operation, fault and accident conditions.

4.1.6 Claims, Arguments and Evidence

118. In Ref. 18, several queries were raised in relation to information presented in supporting documents and how this is traced to the pieces of evidence in the CSR for which they have been prepared.

119. One of the components I sampled was the PZR, for which the claim is made that "High quality is achieved through good design, manufacture and functional testing", which is supported by the arguments that "Design analysis will confirm integrity based on conservative assessment (Argument 1.2)" supplemented by the statement "Design analysis is performed according to RCC-M code to demonstrate that design assessment has been done using conservative methods and has adequate margin."

120. The Pressuriser CSR (Ref. 31) cites two pieces of evidence relating to structural assessment:

- The size for PZR complies with the limits defined by RCC-M (Evidence 1-2-3).
- The design analysis complies with the allowable stress limits as specified in the recognised code for Class 1 components and the fatigue usage factors at the end of life are below 1 (Evidence 1-2-4).

121. It follows from the description of these pieces of evidence that the sizing and the mechanical behaviour of the PZR are addressed separately. This approach is consistent with the organisation of the design process as presented in Clause B 3112.2 of RCC-M. RCC-M additionally states that these two steps are covered separately – for the PZR, sizing is dealt with in RCC-M B 3300 and mechanical behaviour is dealt with in RCC-M B 3200.

122. I sampled the supporting technical documents for Evidence 1-2-3 and 1-2-4 for the PZR, the Pressuriser Dimensioning Report (Ref. 42) and the Pressuriser General Report of Mechanical Analysis (Ref. 43) respectively.

123. A number of questions were raised in RQ-UKHPR1000-1151 to ascertain the origin of the information presented. From the RP's response (Ref. 28), further questions were raised to confirm compliance with the design code in RQ-UKHPR1000-1737 (Ref. 28). From this, shortfalls were identified in the traceability between evidential statements and the documents that have been presented in support of these statements. Specifically, the supporting references contain information that justifies the overall structural integrity of the plant, but the evidential statements, do not in some cases, bear direct relevance to them.

124. I made a similar finding for the RCP. For this reason, I do not consider this to be an isolated shortfall for the PZR, but indicative of a shortfall in the overall approach to presenting this key information in the SI safety case.

125. Thus, the current safety case does not allow the reader to readily trace the necessary information for design code compliance. In accordance with ONR SAP ECS.3, all structures, systems and components that are important to safety should be designed, constructed, commissioned, operated, maintained, tested and inspected to the appropriate codes and standards.

126. In the course of my assessment, it has been challenging to trace the sources of the underpinning information (e.g. design loading specification covering normal operation, faults and accident conditions, including plant transients). I also consider the absence of traceability of evidence to be a shortfall against expectations laid out in ONR SAP SC.4, para. 101(d) (Ref. 2):

"A safety case should provide sufficient information to demonstrate that engineering rules have been applied in an appropriate manner. (For example, it should be clearly demonstrated that all structures, systems and components have been designed, constructed, commissioned, operated and maintained in such a way as to enable them to fulfil their safety functions for their projected lifetimes)."

4.1.7 Traceability of Claims, Arguments and Evidence in System Design Manuals

127. The SI GDA Step 3 assessment report (Ref. 7) noted that the RP has provided a suite of SDMs that identify system functional requirements and form the basis of procurement activities. These were not sampled in detail within the structural integrity area within Step 3 of GDA, however it was noted that there was no clear link to relevant structural integrity safety submissions presented within the SDMs. I therefore

chose to sample a number of relevant SDMs to ascertain whether SI requirements for a range of safety significant components are captured and traceable, where relevant.

4.1.7.1 Reactor Coolant System Design Manual

128. The reactor coolant system (RCS) comprises of many different components, essential for the normal and emergency operation of the UK HPR1000. The majority of HICs for the UK HPR1000 make up the RCS, including the RPV, MCL, SG, PZR and RCP. I therefore consider this system as one relevant for sampling the traceability and linkage of SI claims.

129. Chapter 3 of the RCS Design Manual (Ref. 44) details the RCS functions and design bases. I reviewed this document and was able to identify the SI classifications for HIC and HIC candidates identified within GDA, as detailed within the RP's submission 'Equipment Structural Integrity Classification List' (Ref. 45). Within this document, there is clear traceability to the methodology followed and classification analyses completed to support the assigned SI classification of key components that make up the RCS.

130. In completing my review, I did note a minor inconsistency related to the classification of the SG secondary side shell. The SDM identifies the SG secondary side shell as being a BSC-2 design provision, which in accordance with the RP's methodology for HIC and SIC safety cases, constitutes a SIC-2 or RCC-M Class 2 code requirement. This in contrast to the information presented within Rev. G of the 'Equipment Structural Integrity Classification List', (Ref. 45) which clearly states the SG secondary shell as being HIC and as such, should warrant a BSC-1 design provision. Given the importance of HIC claims for nuclear safety, it is my expectation that the SDM for this system would have reflected the SI safety case necessary to provide the claimed safety function.

131. Whilst this error does not undermine the SI case for the SG, it provides a further example of the need for maintaining oversight of the developing safety case by the RP, to ensure clear and accurate traceability of SI safety claims linked to the SDMs. This error is similar to one I identified in the CCWS SDM when sampling the surge tank safety case. In my opinion, this shows that the shortfall may not be an isolated event. I consider the minor shortfall already raised encompasses this error.

4.1.7.2 Steam Generator Blowdown System Design Manual

132. The RP's 'Equipment Structural Integrity Classification List' (Ref. 46) identifies the steam blowdown line as a candidate HIC. This document provides a concise overview of the consequences analyses undertaken to explain how the final SI classification of SIC-2 for the SG Blowdown Line is reached. This is summarised in a referenced document 'SG Blowdown Lines Classification Conclusion' (Ref. 47) Where the SG Blowdown Lines are discussed in the two references above, the components are identified as being part of the APG-Steam Generator Blowdown System (SGBS) system.

133. I sampled the SDM for the SGBS and identified SI claims discussed under Sub-section 3.2.4 'APG-Steam Generator Blowdown System Design Manual Chapter 3 System Functions and Design Bases' (Ref. 48). As has been noticed for other SDMs, the SDM states that "currently, the structural integrity classification of APG [SGBS] is still in progress and the information will be supplemented later". Given the time that has elapsed since the last revision of the SDM (Rev. D, November 2019) and the completion of 'SG Blowdown Lines Classification Conclusion' (Ref. 47) Revision B, June 2020), it is my expectation that this would have been updated by this point of the GDA. I have therefore identified this as another example that supports the minor shortfall already identified for the RP to ensure that a clear and coherent safety case if

provided, by reviewing and updating relevant SI safety case documents are updated post GDA.

4.1.7.3 Main Steam Line Valves

134. Through assessment work conducted under RO-UKHPR1000-0058 (Ref. 49), I noted a lack of traceability for the safety classification of the Main Steam Safety Valve (MSSV) and Main Steam Relief Isolation Valve (MSRIV), as presented within the MSL CSR. The Main Steam System (MSS) was identified by the RP as being a HIC, however two major valves located within the safeguards buildings were classified as SIC-2. As discussed in Sub-section 4.2 (Classification), RO-UKHPR1000-0058 was raised to better understand the RP's approach to justify classification of the MSL and major valves in the safeguard building. The RP's 'MSL SI Classification Analysis Report' (Ref. 50) identified that the consequences of the MSRIV and MSSV gross failure were tolerable, and as such a SIC classification was appropriate. The MSL CSR identified the MSSV and MSRIV as being SIC-2, however there was no traceability to the evidence provided to support the SIC-2 classification, in preference to a SIC-1 or SIC-3 classification.

135. Nonetheless, I also noted that the MSRIV and MSSV are designated as SIC-2 components (Ref. 51). I did not consider there to be a clear link between how the RP progressed from identifying the SIC classification range from Ref. 40, to assignment of the SIC-2 SI class (Ref. 51).

136. In RQ-UKHPR1000-1727 (Ref. 28) I sought further clarification from the RP. The RP explained that the classification was based on safety functional requirements, which are contained within the relevant System Design Manuals for the Main Steam System (MSS) (Ref. 52) and Atmospheric Steam Dump System (ASDS) (Ref. 53).

137. The RP recognised that the MSRIV and MSSV components are the interface between the MSL and ASDS/MSS vent. As a boundary interface, there should be a clear description of where the evidence supporting classification is presented. The RP made a commitment to update the documents to improve traceability between the interacting system safety cases. I have reviewed these changes and can confirm that the SDM for the MSS does reflect the HIC status of the MSL and MSIV. However, classification of the MSSV as SIC-2 is not clearly stated, with only limited reference made to the barrier classification of BSC-2 and code classification of RCC-M2. This is similarly noted for the SDM of the ASDS, which covers the safety classification of the MSRIV (again, B-SC2 and RCC-M2).

138. The RP subsequently revised the CSR (Ref. 54) to ensure that there is clear traceability for the basis of safety classification for these major valves, which I consider to be satisfactory to address my queries (ONR SAP ECS.2, Ref. 2).

139. Whilst it can be inferred from the RP's 'Methodology for HIC-SIC' document (Ref. 26) that a B-SC2 and RCC-M2 component must be SIC-2 component, I do not consider that this constitutes a clear link to relevant structural integrity safety submissions (such as the SDMs). I note from my sample that this is only an issue for non-HIC examples, and that HICs are clearly defined in the SDMs. I therefore consider this to be a minor shortfall, and that the RP should consider reviewing the SDMs post-GDA to identify any relevant reliance on SI safety submissions to further improve traceability of SI claims for the licensee.

4.1.7.4 Alignment of Critical Components

140. A number of components on the UK HPR1000 are alignment-critical for the correct functioning of the reactor plant during normal operations and adverse conditions. By

alignment critical, I am referring to the importance of dimensional tolerances and geometric positioning relevant to adjacent components.

141. To support my targeted sample of alignment-critical components, I requested in RQ-UKHPR1000-1641 that the RP provide a high-level summary of those components considered to be alignment-critical and what assessments are performed in order to ensure that they operate as intended during all plant conditions, faults and external hazard events, within the design basis (Ref. 28).

142. The RP's response to RQ-UKHPR1000-1641 (Ref. 28) listed the RPV, Reactor Vessel Internals (RVI) and RCP as being the most SI safety significant alignment sensitive components. The RP's response to the questions raised focussed on the RCP as an example, with minimal information provided for either the RPV or RVI. As such, it has not been possible within GDA to develop any informed view on the requirements for alignment placed on the RPV and RVI and how they are demonstrated to be met, through the limited information provided in the CSRs.

143. Following review of the RP's response to RQ-UKHPR1000-1641, I raised RQ-UKHPR1000-1752 (Ref. 28) to get further information on the response provided for the RCP. From the RCP CSR, there is no specific mention of alignment as a consideration for fulfilling the four safety functional requirements, three of which are related to provision of adequate flow (which in turn are presumably dependent on maintaining correct alignment of the relevant RCP components). It is therefore not clear where the detailed design requirements for alignment are derived, to ensure that the three high-level SFRs related to flow will be satisfied (Ref. 55).

144. It is evident that work has been conducted as detailed in the RCP code assessment report to support the substantiation of the RCP in terms of alignment and function of the RCP. However, a well-defined link to where the evidence will be used is lacking. Hence, it has not been possible to clearly establish that the correct analysis has been done and that it is consistent with the requirements, noting that the standards used by the RP are not quoted within the 'Pump Equipment Specification' (Ref. 56).

145. Reviewing the CSR for the RVI (Ref. 57) several SFRs refer to the requirement of the RVI sub-components to "locate, support, restrain, protect and guide". Whilst some arguments provided within the CSR are related to structural integrity, these are based around design code compliance and allowable stress limits. The CSR does not explicitly address the issue of providing adequate tolerances and stiffnesses to ensure alignment is achieved and maintained.

146. The RP's response to RQ-UKHPR1000-1641 (Ref. 28) states that "in order to ensure the alignment of components, the requirements of alignment are mainly controlled during the manufacturing and installation". In my opinion, if alignment of a component is critical in it fulfilling its SFRs, then the required performance needs to be defined and substantiated (where reasonable) during the design stage, prior to manufacturing and installation.

147. The RP's response to RQ-UKHPR1000-1752 (Ref. 28) also included some additional (albeit brief) discussion on the alignment of the RPV and the upper/lower RVI, which fulfil SFRs related to aligning and limiting the horizontal displacement of the RVI during operation and guiding the Rod Cluster Control Assembly (RCCA) to support the control of reactivity. This includes RCCA drop time analysis, which accounts for the effect of RPV and RVI deformation during normal operation and hazard conditions. According to this theoretical analysis, the RP claims that the RCCA drop time can meet the criteria required by the safety case. However, unlike the RCP, regarding deflections and where clearance is required, it is noted that the design code assessment reports make no clear statements regarding deflections, whether clearances are maintained, etc.

148. From the information sampled, there is insufficient evidence available to demonstrate that the RP's approach meets ONR expectations for highest reliability components, such that sound design concepts and proven design features have been used (ONR EMC.3, Ref. 2). I consider the lack of clear evidence for how geometric alignment is demonstrated at the design stage is a matter that warrants tracking during site specific stages.

AF-UKHPR1000-0116 – The licensee shall, as part of detailed design, demonstrate the impact of geometric alignment on fulfilling safety functions for components in situations where misalignment cannot be discounted within the safety case.

4.1.8 Strengths

149. Based on the information that I have sampled, the RP has developed a safety case methodology structure that recognises and addresses ONR SI expectations. The RP has engaged openly and implemented changes to reflect the UK context within the SI safety case for the UK HPR1000.

150. The safety case developed within GDA has covered a broad range of the UK HPR1000 SSCs, demonstrating how the arguments and underpinning evidence has been developed and presented proportionately for SI safety claims, ranging from the highest (HIC) to lowest (SIC-3) classifications.

4.1.9 Outcomes

151. From the information that I have sampled, the traceability of evidence in the CSRs does not allow the reader to drill easily down into the detail. The CSR should link the main claims in the safety case and the underpinning detailed assessments. As such, it would normally be expected to provide a clear signpost and comprehensive list of all the technical assessment reports. To be clear, the issue is not with the technical assessments themselves, rather the ability to be able to identify all relevant documentation from within the CSR and ensure there are no gaps.

152. The CSRs provide a description of the relevant safety functional requirements (SFRs), however there is no description of their derivation. Most of these SFRs have no stated connection to the integrity of the structures, which in my opinion, is an important feature for the CSRs and the overall SI safety case.

153. For instances where SI classified SSCs are claimed to be 'alignment critical', the safety case is lacking a clear link to evidence of how this is demonstrated through design, manufacture and operation. The CSRs sampled do not explicitly address the issue of providing adequate tolerances and stiffnesses to ensure alignment is achieved and maintained. In my opinion, if alignment of a component is critical in it fulfilling its SFRs, then the required performance needs to be defined and substantiated (where reasonable) during the design stage, prior to manufacturing and installation I have raised AF-UKHPR1000-0116 to address this shortfall.

154. It has become apparent during my assessment that there are several instances where the safety case requires updating to reflect changes implemented by the RP in response to ONR queries. This has resulted in an inconsistent safety case, for which traceability of claims, arguments and evidence is lacking. Whilst each incident in isolation appears minor and does not undermine the technical basis of the safety claims being made, when amalgamated, I consider this to be indicative of a generic shortfall that requires resolution during site specific stages. I therefore raise the following assessment finding:

AF-UKHPR1000-0186 – The licensee shall, as part of detailed design, demonstrate that the evidence necessary to underpin the claims and arguments in the safety case is robust, traceable and supported by a clear narrative. This should include, but not be limited to, the provision of structural integrity classification to reflect its importance within the system design manuals and clear linkage within component safety reports to the consequence analysis undertaken for structural integrity classified components.

4.1.10 Conclusion

155. Overall, I am satisfied that the RP has developed an adequate safety case structure for the UK HPR1000 to demonstrate how the risks associated with structural integrity of the plant are identified, assessed and managed. From my assessment, I have identified three assessment findings (AF-UKHPR1000-0006, AF-UKHPR1000-0116 and AF-UKHPR1000-00186), associated with the traceability and derivation of design loads, impact of geometric alignment and general traceability of claims, arguments and evidence within the safety case. I have also identified a number of minor shortfalls that I consider to be important and require resolution post-GDA to ensure that the licensee is fully aware of the level of SI requirements and demonstration necessary to meet ONR expectations. These minor shortfalls include the need to update extant safety case documents to reflect the consequence analysis completed for the surge tanks and ensure SI classification is referenced appropriately within the relevant system design manuals. It should be noted that whilst these items are important, my observations are not related to the adequacy of the evidence/technical assessments themselves, rather the ability for the safety case user to identify the relevant documentation to support the claims and arguments, to ensure there are no gaps.

4.2 Structural Integrity Classification

4.2.1 Assessment

156. During Step 3 of GDA, I assessed of the RP's proposals for SI classification, focussing on three aspects:

- methods development;
- linkage to the overall plant categorisation of safety functions; and
- classification of SSC, including the HIC candidate listing.

157. From my review of the developing methods and safety case, I was satisfied that the RP's approach to SI classification was suitable, and importantly, allowed the RP to identify those structures or components (including locations) that will require a higher Structural Integrity claim. The RP recognised the link between the UK HPR1000 plant categorisation of safety functions and classification of SSC and Structural Integrity classification.

158. Whilst I was satisfied with the progress that had been made in Step 3 of GDA, I identified a number of areas that required further consideration in Step 4 of GDA, in order to satisfy my review of the RP's arrangements and application of SI classification for the UK HPR1000 (Ref. 7).

159. To address the areas identified for further consideration during Step 3 of GDA, I have continued to sample the RP's proposals for SI classification focussed on the following four aspects:

- SSC and SI classification
- Linkage between the SSC safety, SI and code classes
- HIC Candidate SSC and their SI Classifications
- HIC Listing

4.2.1.1 SSC and SI Classification

UK HPR1000 SSC Classification

160. The RP's overall approach to categorisation of safety functions and classification of SSC for the UK HPR1000 is to use the UK HPR1000 reference design, Fangchenggang Nuclear Power Plant Unit 3 (Fangchenggang-3) plant classification as the initial basis, and then to identify and address 'gaps' in meeting ONR's expectations as per ONR SAP ECS.2 to ECS.3 (Ref. 2). For Fangchenggang-3, the plant categorisation of safety functions and classification of SSC is founded on IAEA SSG-30 (Ref. 9). This uses a combination of the functional classification (fault conditions) and design provision (or barrier class – effectively direct classification) under normal operation to classify SSC.

161. It is preferable that the plant categorisation and classification scheme is in place to inform the development of the SI classification of SSC. However, in the absence of the provision of the UK HPR1000 plant safety categorisation and classification of SSC, I was content for the SI classification to progress because the integrity provisions for the major vessels and piping tend to be determined by the design provision (or barrier class). Thus, the plant classification for several of the major vessels and piping were unlikely to change post implementation of proposals to meet ONR's expectations. However, this may not be the case for all SSC because ONR's expectations are different to Fanghenggang-3, such as the levels of redundancy and diversity, role of leak before break (LBB), consideration of postulated gross failure, along with the assumptions for indirect consequences such as pipe whip/missile methodologies as raised in RQ-UKHPR1000-0007 and RQ-UKHPR1000-0102 (Ref. 28).

162. The RP's SI classification of SSC was therefore dependent on the progression of consequences analyses work to meet ONR's expectations. In addition, ONR's expectations for the categorisation of safety functions and classification of SSC need to be addressed. The RP committed to reviewing the position for SI classification following the implementation of its proposals to meet ONR's expectations. The overall position with the development of the RP's categorisation of safety functions and classification of SSC is a topic that was progressed by ONR's Fault Studies Inspector and is discussed in ONR-NR-AR-21-014, (Ref. 58) and concludes that ONR is content that the overall cat and class scheme proposed for the UK HPR1000 is consistent with meeting ONR expectations and international guidance.

Multi-discipline Input to SI Classification

163. To meet ONR's expectations relating to SI classification under ONR SAPS EMC.1-3, ECS.2-3 (Ref. 2), the RP developed several methodologies. ONR's SI discipline led the assessment of the SI classification of SSC, in particular, the development of the RP's listing of HIC components (HIC listing). However, consequences analyses (direct and indirect) are an essential input to inform the SI classifications. Thus, to ensure coherency in ONR's assessment, I consulted ONR's Fault Studies (direct) and Internal Hazards/Civil Engineering (indirect) inspectors on the consequences of the postulated gross failure of SSC.

164. ONR's internal hazards inspector, supported by ONR's civil engineering inspector, led on the assessment of the RP's pipe whip and missile methodologies. ONR's SI discipline provided advice on the assessment criteria which needed to take cognisance of the low frequency of the loading. The RP's methods were adequate to progress beyond GDA Step 2, but in GDA Step 3, I was made aware of difficulties in meeting ONR's expectations during their application as per RQs RQ-UKHPR1000-0137, -0138 & -0492 (Ref. 28). I followed-up the adequacy of the RP's application of methods for pipe whip and missiles from an SI perspective in GDA Step 4. The final position is

discussed in the Internal Hazards (IH) assessment report, ONR-NR-AR-21-012 (Ref. 59).

165. For the HIC and the Candidate HIC SSC, I also needed to gain assurance that their structural integrity would not be undermined by the postulated gross failure of other SSC. I therefore supported ONR's IH specialist on the assessment of the RP's approaches to protecting HIC components from the indirect consequences of the postulated failure of SSC through the progression of RO-UKHPR1000-0046 (Ref. 49). In summary, the IH inspector was satisfied that the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is discussed in the ONR IH assessment report, ONR-NR-AR-21-012 (Ref. 59).

166. A further point relating to the multi-discipline aspects of the SI classification related to improving my understanding of the RP's QA arrangements for Fangchenggang-3 and their relationship to the SI classes for the UK HPR1000. The RP provided 'Quality Assurance Grading Method and Associated Management Requirements' (Ref. 60), which outlines a graded approach based on IAEA standards (Ref. 9).The relationship between the plant classes and QA provisions in Fanghenggang-3 is described. I had assumed that design activities/analyses featured in this graded QA approach. However, the RP uses a uniform approach to QA for design activities. The veracity of the RP's QA arrangements is within the remit of ONR MSQA Inspector and is discussed in the Management for Safety and Quality Assurance assessment report, ONR-NR-AR-21-003 (Ref. 61). Regarding SI, I followed-up the RP's QA arrangements in relation to its DTA with ONR's MSQA inspector in GDA Step 4. This is discussed under the avoidance of fracture demonstration Sub-section 4.3 of this assessment report.

SI Classification

167. In the reference design for the UK HPR1000, LBB arguments and break preclusion type arguments are applied to several of the main piping systems including the MCL, MSL and containment penetration locations. This effectively precludes the need to consider the consequences of postulated gross failure. However, to meet ONR's expectations for the UK HPR1000, the RP has developed an approach to SI classification founded on a systematic consideration of the direct and indirect consequences of postulated gross failures (Ref. 62). The SI classifications are also informed by the categorisation of functions and classification of SSCs (Ref. 63). A sub-set of structures and components is then identified within the plant standard class 1 SSC as Structural Integrity class 1 (SIC-1) structures and components, which require a higher reliability claim. These components are identified as 'high integrity components' (HIC) in the RP's safety case.

168. In contrast, where gross failure is not discounted, the SSC is assigned to one of three Structural Integrity classes: Standard Class 1 (SIC-1), Standard Class 2 (SIC-2) and Standard Class 3 (SIC-3), with the SI class dependent on the safety class, the consequences of postulated failures and the level of protection offered in the design as per the RP's response to RQ-UKHPR1000-0083 (Ref. 28). The subsequent allocation of the design code (pressure equipment) class is also important. The relationship between the different classification schemes is discussed further in Sub-section 4.2.1.2 below.

169. I was generally content with the RP's SI classification proposals, as they are similar to those adopted by other RPs. Nevertheless, in addition to assigning SI classifications of HIC and SIC above, the RP had identified potential requirements for 'significant' SIC-1 structures and components. This SI classification is not currently reflected within the SI classification methodology and so in Step 4 of GDA, I sought to establish the

RP's intent on developing this SI classification for inclusion within its current methodology for the UK HPR1000.

170. In subsequent technical exchange discussions, the RP confirmed that additional SI provisions would be applied to significant SIC-1 components e.g. RVI, as applied in Fanghenggang-3, as a means to reduce risk. However, the RP chose not to include a specific SI class of 'significant SIC-1' within the RP's SI classification process (Ref. 64). My sample review of the 'Methods and Requirements of Structural Integrity Classification' (Ref. 64) confirmed the RP's proposed SI classifications were as expected, with no mention of a significant SIC-1 classification. This notwithstanding, I also note that there is no indication within the Safety Case Methodology document for the UK HPR1000 (Ref. 26), to explain the existence or even criteria for deciding whether a component is a significant SIC-1 or to highlight that additional SI provisions are necessary. I am aware that the RP has identified and used similar concepts for Fangchenggang-3, for example, for the RVI. In general, ONR has a policy of not accepting a lower standard of safety demonstration than that required for the reference design plant in the country of origin. Understanding how additional SI provisions for Class 1 components on Fangchenggang-3 are addressed for the UK HPR1000 is an important point, which requires further consideration during the detailed design stage. I will therefore raise an assessment finding to capture this.

AF-UKHPR1000-0187 – The licensee shall, as part of detailed design, demonstrate how additional provisions applied for Structural Integrity Class-1 components for the UK HPR1000 reference design are addressed within the UK HPR1000 safety case. This should include, but is not limited to, justification of the associated criteria and additional provisions for the reactor vessel internals.

171. A further generic SI classification topic arose during the consideration of the classification of the fuel transfer tube (FTT). Although RQ-UKHPR1000-0479 included specific questions relating to the identification of the limiting conditions for the FTT, ONR highlighted a generic point that the RP's SI classification process should identify and consider the limiting faults during all plant states (Ref. 28). I have reviewed the extant version of the RP's SI classification document (Ref. 64) to establish whether the RP has addressed this point in the development of its SI classification process. The RP's 'Methods and Requirements of Structural Integrity Classification' (Ref. 64) states:

"All direct consequences (such as loss of coolant inventory, thermal transient effects, and radioactive releases) and indirect consequences (such as missiles, blast, pipe whip, flooding and steam jets, some effects caused by environmental condition changes or other subsequent hazard effects) should be taken into consideration."

172. However, the importance of the limiting conditions and plant states is not apparent, and at best, any inference is tenuous. It is therefore unclear whether the learning from the classification of the FTT has informed the development of the RP's SI classification process. I view this matter as significant to warrant tracking during site specific stages.

AF-UKHPR1000-0188 – The licensee shall, as part of detailed design, demonstrate that the structural integrity classification process identifies and confirms the limiting faults during all plant states.

Fangchenggang-3 Equipment List

173. At a workshop in China, the RP presented an extract from its Fangchenggang-3 Equipment List (Ref. 65). This was a useful document as it presented the linkage between the plant safety class, code class and other provisions (e.g. seismic and QA classes). I recognised that the underlying assumptions used may differ from those expected for UK application, but nonetheless consider it may provide useful insight to inform my sampling approach for the UK HPR1000. In particular, it would assist with

the identification of SSC with lower SI classifications e.g. SIC-2 and SIC-3. It would also inform my targeting of a sample of SSC where the designation of SI provisions is not usually straightforward e.g. SSC with passive safety features and non-safety classified SSC.

174. I raised RQ-UKHPR1000-0260 to request a copy of the 'Fanghenggang-3 Equipment List' (Ref. 28). A translated version of the Fanghenggang-3 Equipment list (Ref. 65) arrived late in GDA Step 3 and so I followed-up this topic in GDA Step 4. From my sample of this (Ref. 65), I identified the pressuriser surge line, the accumulators, MSRIV, MSSV, FTT, ST and MSL beyond the NI, as a representative sample of components to inform a view on the SI classification outputs, the safety case provisions and traceability of the safety class and code classes, which I discuss below.

Equipment SI Classification List

175. In addition, a key output from the RP's SI classification work was the RP's 'Equipment SI Classification List' (Ref. 66). In Step 3 of GDA, the RP updated the Equipment SI Classification list to capture the developments in the HIC candidate listing (Ref. 66). The list was further updated in Step 4 of GDA to reflect changes to the plant categorisation of safety functions and classification of SSC; the completion of the consequence analyses work; and the implementation of any design changes. In Step 4 of GDA, I sampled the outputs for a range of SI classifications to ensure these meet ONR's expectations and that there is coherency in the safety case. The traceability aspects are the subject of Sub-section 4.1 of my report. In this section, my focus is on sampling the RP's SI classification outputs, in particular, the linkage between the SSC safety, SI and code classes along with their listing of HIC components. I discuss these next.

4.2.1.2 Linkage Between the SSC Safety, SI and Code Class

176. Nuclear pressure vessel design and construction codes, such as RCC-M (Ref. 14) and ASME III (Ref. 13) have sets of requirements for the design and construction of pressure vessels and associated pressure retaining components such as pipework and valves. The requirements are graded according to which of three construction classes, Class 1, Class 2 and Class 3 is specified for the components. Construction Class 1 components are designed, constructed and inspected to higher standards than construction Class 2 and likewise construction Class 3. The construction class specified for the component also determines the through life inspection regime for the component.

177. This is a well-established approach for nuclear pressure equipment and gives a varying quality level according to the construction class specified for the component. In addition to these three construction classes for pressure equipment designed to nuclear standards there may be pressure equipment on the plant that does not need any controls over and above normal industrial standards.

178. Whilst the pressure vessel design codes provide rules for design and construction against these different pressure equipment construction classes, they do not provide the criteria for allocating the constriction class that should be specified for a particular component. This is undertaken on a plant specific basis according to separate rules and guidance. Thus, the allocation of an appropriate construction class to pressure retaining equipment is an important underpin to the structural integrity case for the plant.

179. As part of my assessment of the RP's SI classification proposals I sampled how the RP's safety categorisation and classification of SSC was linked to the SI class, pressure equipment class (i.e. the SI provisions within the RP's safety case). ONR's

expectations in this area are captured under ONR SAP ECS.2 and Paragraphs 160 and 166 along with ONR SAP ECS.3 and Paragraphs 170 and 171 (Ref. 2).

180. The RP has developed a generic 'Methodology of Safety Categorisation and Classification' (Ref. 62). The generic SSC classification process is based on IAEA SSG-30 (Ref. 9) with the safety class of SSC determined from consideration of the function category (active SSC for fault protection, F-SC) and design provision/class (passive – containment role, B-SC). If an SSC delivers both a safety fulfils multiple functions and has a design provision its safety classification is the higher of the F-SC and B-SC categories depends on the function with the highest function categorisation, as per Section 6 of the RP's 'Methodology of Safety Categorisation and Classification' (Ref. 62). This approach has some similarities with that adopted for the UK EPR™ (Ref. 67).

181. However, for the UK HPR1000, there are three function and three design provision categories and the classification of equipment delivering the function/design provision matches the categorisation. Therefore, equipment that deliver a category 1 safety function are assigned safety class 1, that which deliver a category 2 function are assigned safety class 2 and that which deliver a category 3 function are safety class 3. The UK HPR1000 safety case uses the notation F-SC1, F-SC2 and F-SC3 to indicate safety class 1, 2 and 3 respectively.

182. SSCs can also be classified directly based on the consequences of failure (loss of integrity). The potential consequences are graded as High, Medium or Low (Ref. 58).These SSCs are referred to as 'Design Provisions'. The UK HPR1000 safety case uses B-SC1, B-SC2 and B-SC3 to indicate Design Provisions Class 1, 2, and 3 which informs the allocation of the design code (pressure equipment) class and the structural integrity class of components.

183. To meet UK expectations, the high-level principle is that safety Class 1 and Class 2 SSC in accordance with the ONR SAPs (Ref. 2) are designed in accordance with nuclear specific codes and standards, where those are available. Safety Class 3 safety systems may be designed in accordance with nuclear specific or industrial codes and standards as per Sub-section 7.2 of 'Methodology of Safety Categorisation and Classification' (Ref. 62). The categorisation of safety functions and the classification of SSC is discussed in the Fault Studies assessment report (Ref. 58). I will suffice to say that overall, ONR is content that the scheme adopted by the RP for the UK HPR1000 is consistent with ONR expectations laid down in NS-TAST-GD-094 (Ref. 5) and international guidance (Ref. 9).

184. This notwithstanding, the design code (pressure equipment) class that follow from the applied SSC safety classification are matters for the structural integrity discipline. The linkage between the RP's SSC safety classification and the design code (pressure equipment) class is via the assignment of a mechanical equipment class (M1 to M3 or NC). The M Class then determines the design code (pressure equipment) class, with the selection of the M class informed by rules based on the importance of the functional requirements and the design provision (F-SC and B-SC). In accordance with IAEA guidance, the class of SSCs forming the reactor coolant pressure boundary (RCPB) is increased over the default that would be given by the direct classification method. Thus, design provisions forming part of the RCPB whose failure cannot be compensated by the normal makeup are specified as Class 1, whereas design provisions forming part of the RCPB whose failure can be compensated by normal makeup, are specified as Class 2 in Sub-section 6.2 of the RP's 'Methodology of Safety Categorisation and Classification' (Ref. 62).

185. In addition, and in parallel, the RP has developed a specific approach for structural integrity in 'Methods and Requirements of Structural Integrity Classification' (Ref. 39), which explains how the safety function class or design code (pressure equipment)

provision class is are linked via through consequence analyses to the RP's SI classes including HIC, as per Sub-sections 3.2-3.5 of 'Methods and Requirements of Structural Integrity Classification' (Ref. 39).

186. As the SI classification of the major vessels and piping systems were in general appropriate, I sampled a selection of lower SI classification SSC (non-HICs) as a means to confirm that the outcome of the RP's classification process within SI would meet ONR's expectations. I was content that the design code (pressure equipment) class outputs, and the RP's SI classification of SIC-1 and SIC-3 for the pressuriser surge line and surge tank respectively, were appropriate and aligned with the protection claims and the safety functional requirements.

187. The position for the UK HPR1000 ACC was not as straightforward. Notably, in Sub-section 2.6 of the ACC CSR (Ref. 36) the RP states that:

"According to the 'Methodology of Safety Categorisation and Classification' (Ref. 68), the safety function class of the ACC is F-SC1, the barrier class is NC, the code design class (pressure equipment design class) is RCC-M2."

Sub-section 2.6 of the ACC CSR later states that the "Based on the ACCs Failure Consequence Analysis Report, (Ref. 68) the ACC is classified as SIC-2."

188. Using Tables T-3.2-1 and T-3.3-1 of the generic categorisation and classification methodology document (Ref. 39), the safety function class and SI class do not align with the SIC-2 classification. In particular, the delivery of an F-SC1 safety function class relates to a Safety Class 1 relates to a mechanical class of M3, and hence a design code (pressure equipment class) class of RCC-M Class 3, which in accordance with the structural integrity classification document (Ref. 39) is commensurate with a standard (or safety) Class 3 and hence an SI classification of SIC-3, and not SIC 2.

189. Furthermore, in terms of the consequence analyses, the only specific difference between a SIC-2 and SIC-3 component is given in Table T-3.3-1 of Ref. 39, with SIC-3 assigned to "protected potential failure consequence of components", whereas a SIC-2 component can result in an "offsite minor radioactive release".

190. I raised this with the RP, who confirmed during a technical exchange meeting (Ref. 40) that, according to the consequences of failure analyses performed in the 'Methodology of Safety Categorisation and Classification' (Ref. 68), no Design Basis Condition (DBC) or Design Extension Condition (DEC) will be raised and no safety functions to deal with this PIE will be affected by missile effects. As such, failure of the ACCs would not result in an offsite radiological release.

191. With regard to the allocation of a design code (pressure equipment) class of RCC-M Class 2 and a SI classification of SIC-2 for the ACC, the RP explained that during its review of OPEX, it was found that the ACCs are classified as SIC-2 for other reactor designs, which had informed the design of the UK HPR1000. I had no objection to the RP assigning a SIC-2 SI classification, with commensurate RCC-M Class 2 code provisions to the ACCs, because this was more akin to what I would expect based on the safety functional requirements of the ACC.

192. Nevertheless, I considered there to be a lack of clarity in the classification process, and it was also unclear as to how the intent of ECS.2 and ONR TAG NS-TAST-GD-094 (Ref. 5) has been satisfied in the design of the UK HPR1000 ACCs and potentially other SSC. I raised RQ-UKHPR1000-1620 (Ref. 28) to gain further clarification of how the SI class and pressure equipment class are linked, and for the RP to identify any further examples where SI class did not align with the RP's 'Methods and Requirements of Structural Integrity Classification' document (Ref. 39).

193. In response the RP explained that, as for Fangchenggang-3, OPEX from previous engineering practices was considered and so RCC-M 2 (or RCC-M Class 2) was selected for design and construction of the ACC. Thus, for UK HPR1000, if RCC-M Class 2 (or RCC-M 2) is taken as an input for SI classification, the SI class of ACC should be SIC-2 according to T-3.3-1 of the RP's 'Methods and Requirements of Structural Integrity Classification' (Ref. 39).

194. I noticed that for the ACC, the OPEX considered from other 'similar' PWR design reactors was focused on reactor technology deployed within the UK HPR1000 country of origin, i.e. the People's Republic of China. I consider that where the technology is being applied within the UK, a broader range of OPEX should have been considered, taking into account precedent within the UK (e.g. AP1000 and UK EPR™ GDA, Sizewell B).

195. I was therefore not satisfied that the RP had satisfactorily explained why a SIC-3 classification was initially selected and subsequently raised to SIC-2 for the ACC for the UK HPR1000. The classification of the ACCs may differ depending on the reactor-specific functional requirements and the consideration of direct/indirect consequences of postulated gross failure. However, in my opinion the safety function of the ACC for the UK HPR1000 is not significantly different from previous reactor designs. Thus, the rationale for changing the classification of SSC needs to be understood, to establish its relevance to a plant design. A change in an SSC classification compared to accepted norms (i.e. the expected design code standards to deliver the safety functions) may relate to a change in the safety functional requirements (up or down), or alternatively, the change in the SSC classification may be an unintended output from the rules linking the M class to the categorisation of safety functions and design provision. A change to the M class rules should not in itself lead to a lower classification.

196. For example, for the UK EPR™, the RP's initial assignment of a M3 mechanical requirement and hence an RRC-M Class 3 design code (pressure equipment) class provision for the ACC was queried by ONR (Ref. 67). This was because the ACC in the UK EPR™ are Safety Class 1 components delivering a Category A safety function. This was subsequently pursued as cross-cutting GDA Issue GI-UKEPR-CC-01.A4. An RCC-M design code (pressure equipment) class of RCC-M Class 2 for the ACC was subsequently implemented for the UK EPR™ to ensure that the design code (pressure equipment) class was commensurate with the safety functional requirements (Ref. 69).

197. In response to RQ-UKHPR1000-1620 (Ref. 28), the RP reviewed the position for all of the other HIC candidates, and based on its review of OPEX, no other SSC with an initial SIC-3 SI classification was identified, where an upgrade to the SI class from application of the SI classification methodology was identified. The position for non-HIC candidates will need to be addressed at the site-specific stage.

198. I welcome the RP's consideration of OPEX in developing the safety case (SC.7 and paragraphs 99 and 100 of Ref. 2), for the UK HPR1000. I also note that for RCPB SSC i.e. the most safety significant SSC from a structural integrity perspective, the design provision class governs the classification and that the mapping through the M class to the design code (pressure equipment) class meets my expectations. This notwithstanding, for lower classes of SSC, the basis for any change in the classification of SSC in comparison with expected norms should be clearly understandable (SC.8, para 111-112 of Ref. 2). In my opinion, adopting a component classification applied from another plant design without fully understanding the reasoning or context for that classification may lead to either over or under classification of SSC. I consider that from an SI perspective, this is indicative of a shortfall in the rules governing the linkage the safety class (F-SC and B-SC) and the M class for the UK HPR1000. This shortfall requires resolution during site specific stages. I therefore raise the following assessment finding:

AF-UKHPR1000-0189 – The licensee shall, as part of detailed design, compare the design code (pressure equipment) class assigned via the M class with relevant operational experience. For structural integrity, if this comparison results in the assignment of a different design code (pressure equipment) class for the component, the licensee shall either upgrade the nuclear pressure equipment design class or justify the use of a lower design class.

199. The RP indicated that the output from its SI classification process was an initial classification/design code allocation and that there is provision to raise the design code (pressure equipment) class if appropriate. I sought assurance in RQ-UKHPR1000-1620 (Ref. 28), whether there would be a specific provision to allow a change in the design code class if the RP designates the component as HIC.

200. The RP cited the MSL as an example, where under the RP's categorisation of safety functions and consideration of the design provisions the safety class is F-SC1/B-SC2 from Fangchenggang-3, and the design code class is RCC-M Class 2. However, the original failure consequences and protection lines of MSL are not in line with the SIC-2 criteria, and so a higher SI class of F-SC1/B-SC1 and hence SIC-1 was assigned by the RP based on ALARP considerations (See RP's response to RQ-UKHPR1000-1620, Ref. 28). In my opinion there is a need for a minimum level of provision in the assignment of the design code (pressure equipment) class via the M class to ensure that the provisions for structural integrity are compatible with a HIC classification.

201. A provision to raise the SI classification for all SIC classifications subject to OPEX and ALARP considerations, was also included in the revised SI classification document (Ref. 63). In my opinion these are not ALARP matters and the need for this amendment is indicative of further difficulties in the M class rules for translating the RP's safety class to the design code (pressure equipment) class. The shortfall is captured within AF-UKHPR1000-0189.

202. A further point from the RP's design code (pressure equipment) class assignment rules relates to the potential to use industrial (non-nuclear) standards/codes to deliver categorised safety functions (F-SC). I initially questioned the proposed use of non-nuclear codes for functional class 3 (SAPs equivalent Category C safety function) for SSC safety class 3. The SAPs indicate that appropriate nuclear industry-specific, national or international codes and standards should be adopted for Class 1 and 2 SSC. For Class 3, if there is no appropriate nuclear industry-specific code or standard, an appropriate non-nuclear codes or standard should be applied instead (SAP ECS.3 Paragraph 171).

203. I asked the RP to clarify under what circumstances from a structural integrity perspective, non-nuclear codes or industrial standards would be invoked. The RP explained that the UK HPR1000 Categorisation and Classification approach follows the IAEA SSG-30 (Ref. 9) method and guidance in IAEA-TECDOC-1787 (Ref. 70). In particular, Table 18 of IAEA TECDOC-1787, specifies that for a functional class 3 component:

- if its failure leads to low consequence, or if it provides cat.3 barrier function, then its design provision class will be B-SC3, and it will meet RCC-M class 3 or equivalent; and
- if not, then its design provision class will be NC, and it will meet conventional codes.

204. As a result, for the UK HPR1000, a F-SC3 equipment shall at minimum meet an industrial code, unless its failure leads to low consequence (equivalent to a cat 3 barrier function) or indeed higher safety classes, in which case the higher design provision class (RCC-M class 3 or higher) will be adopted. For example, the functional class for RCV3220BA (Volume control tank) is F-SC3. Its design code class should be:

- RCC-M2 if its design provision class is B-SC2;
- RCC-M3 if its design provision class is B-SC3; or
- industrial code if its design provision class is NC.

205. The code class for RCV3220BA (Volume control tank) is actually RCC-M2 since its safety class is F-SC3/B-SC2.

206. Furthermore, for Fangchenggang Unit 3, mechanical pressure components classified NC mechanical class, nuclear codes are selected, if available, in preference to industrial standards (Ref. 28).

207. In my opinion from a structural integrity perspective, the RP's approach for to assigning non-nuclear codes to SSC that deliver F-SC3 functions accords with meeting the intent of SAP (ECS.3 Paragraph 171).

208. In RQ-UKHPR1000-1620 (Ref. 28), I also sought clarification as to whether the pressure equipment class for SSC proposed for the UK HPR1000 had took account of any changes in classification arising from the implementation of the ESPN order as identified for the UK EPR™ (Ref. 69). In response the RP reaffirmed its commitment to implement the ESPN Order, ASN Guide 8 and PE(S)R 2016 requirements for the UK HPR1000 (Ref. 28, RP's response to RQ-UKHPR1000-1620). The ESPN order does not apply in the UK, but it is a national practice which, taken together with the nuclear pressure vessel design code, is considered in France and China, to lead to a suitable quality of nuclear pressure equipment. A licensee will therefore need to review the ESPN order to see whether additional requirements need to be applied to the UK HPR1000 as a result of this practice in China (Sub-section 4.4.3.3). In addition, the licensee will need to establish whether by complying with the ESPN Order an upgrade to the class to which nuclear pressure equipment is designed and manufactured is appropriate. I consider this a shortfall that warrants tracking to completion during the site-specific stage. I therefore raise the following assessment finding:

AF-UKHPR1000-0190 – The licensee shall, as part of implementing the ESPN Order during detailed design, review any resultant changes in the classification of structures, systems and components and either upgrade or justify the allocated design code (pressure equipment) class.

4.2.1.3 HIC Candidate SSC and SI Classifications

HIC Candidates

209. It is ONR's expectation that safety cases should not rely on claims of highest reliability, if reasonably practicable (ONR SAP EMC.2 Paragraph 293, Ref. 2). This is because it is out with the achievement of physical defence-in-depth in the plant design (ONR SAP EKP.3, Ref. 2). Furthermore, it is an onerous route to a safety justification with the expectations of measures beyond normal practice and extensive commitments to maintain structural integrity through-life.

210. Thus, the identification and justification of HIC structures and components are important aspects when considering the 'safety claim' relating to SI. The identification of an initial HIC candidate listing of those structures and components requiring a higher SI claim was completed by the RP towards the latter stages of GDA Step 2 (Table 4 below):

Table 4: HIC Candidate Listing for UK HPR1000

Identified Component	Location	Structural Integrity Classification	Consequence
Reactor Pressure Vessel	Reactor Building (BRX)	HIC	break/missile
Pressuriser	BRX	HIC	break/missile
Steam Generator (primary and secondary shell & tube sheet)	BRX	HIC	break/beyond design basis/missile
Main Coolant Line	BRX	HIC[(1)]	LBLOCA break (pipe whip)
Main Steam Line	BRX	Candidate HIC[(1)]	break (pipe whip),
Main Steam Line	Safeguards Buildings A and B (BSA/BSB)	Candidate HIC[(1)]	break (pipe whip)
Main Steam Valves (MSIV)	BSA/BSB	Candidate HIC[(1)]	break/missile
Reactor Coolant Pump (Casing & Flywheel)	BRX	HIC[(1)]	break/missile
Pressuriser Surge Line	BRX	SIC-1[(1)]	break (pipe whip)
Reactor Vessel Internals	BRX	SIC-1[(1)]	break/beyond design
Steam Generator Blowdown Lines	BSA	SIC-2[1)]	break (pipe whip)
Fuel Transfer Tube	BRX-Fuel Building (BFX)	SIC-2[(1)]	break/beyond design
Accumulator	BRX	SIC-2[(1)]	break/missile

[(1)] These were SSC classifications targeted for ONR assessment during Step 3 of GDA.

211. The RP's HIC candidate listing was primarily based on a read-across from other GDAs. However, the rationale for the selection of all HIC candidates and how these relate to the defence in depth provisions afforded on the UK HPR1000 design was not explained. Notably, whilst the RP acknowledged the linkage to the progression of its assessments of the consequences of postulated direct and indirect failures of SSC, the consequence analyses methods were under development and their application was on-going.

212. In GDA Step 2, I was generally content with the RP's proposals for SI classification. However, the RP's initial proposal was to limit its consideration of HIC candidates to SIC-1 SSC. I was aware that lower classes of SCC (e.g. the accumulators) may also be HIC candidates (Ref. 71). I raised this point with the RP and they subsequently expanded consideration of potential candidate SCC to include SI classes below SIC-1 (see Table 4).

213. With regard to the RP's HIC candidates, my main aim was to seek, where appropriate, suitable and sufficient evidence to justify the SI classifications i.e. that there had been a reasonable consideration of whether a HIC claim could be avoided and/or risks reduced to ALARP.

214. The RP subsequently undertook consequence analyses and ALARP assessment work to inform its view on the SI classifications of several HIC candidate SSC including: the main coolant line (MCL), main steam line and associated valves (MSL), reactor coolant

pump (casing & flywheel) (RCP), the pressuriser surge line (SL), SG blowdown lines and components, reactor vessel internals (RVI), fuel transfer tube (FTT) and accumulators (ACC) (see Table 4). I discuss the HIC candidate SSC and its final SI classifications below.

SI Classification of the Main Coolant Line (MCL) (RO-UKHPR1000-0008)

RO-UKHPR1000-0008

215. During the latter stages of GDA Step 2 the RP identified the MCL as a 'definite' HIC (Ref. 66) this was governed by the indirect consequences. At that time the direct consequences e.g. large break loss of coolant accident (LBLOCA) were held to be within the design basis, though in the Preliminary Safety Report (PSR), the design basis for a LBLOCA is limited to a gross failure of the pressuriser surge line (Ref. 72). In addition, the underlying consequence analyses, which inform the structural integrity classification of the MCL, were not available.

216. I therefore concluded that there were important gaps in the RP's case to adequately justify the structural integrity classification of the MCL. Notably, there was insufficient information to form a judgement on whether a HIC case could be avoided and whether the structural integrity classification of the MCL was appropriate and commensurate with reducing risks to ALARP. I raised RO-UKHPR1000-0008: 'Justification of the Structural Integrity Classification of the Main Coolant Loop' (Ref. 49) to outline ONR's expectations and to gain further evidence to address the gaps in the safety case.

217. The RP responded positively and developed a resolution plan to address four actions (Ref. 49):

- ROA1 – Process to establish the Structural Integrity Classification of the MCL.
- ROA2 – MCL Consequence Analyses, Design Optioneering and Identification of Measures to Reduce Risk.
- ROA3 – Justification that the Structural Integrity Classification of the MCL is Commensurate with Reducing Risks ALARP.
- ROA4 – Demonstration of the Adequacy of the MCL Structural Integrity Safety Case.

218. ONR's SI discipline led the assessment of RO-UKHPR1000-0008 (Ref. 49), but with the support of several ONR disciplines: Fault Studies; Internal Hazards; and Civil Engineering. The key ONR SAPs considered within my assessment of the RP's SI MCL classification therefore included EKP.3, ECS.2-3, EMC.1-3, the EHA series (Ref. 2) and the relevant TAG, NS-TAST-GD-016 (Ref. 5).

219. A summary of how the RP progressed RO-UKHPR1000-0008 (Ref. 49) and my assessment of the RP's submissions is provided below with a detailed record available in ONR-NR-AN-20-023 (Ref. 73).

Review of ROA1-4 of RO-UKHRP1000-0008

220. In response to ROA.1, the RP needed to explain the approach it will develop and implement to establish the MCL SI classification in GDA. To address the requirements of this action, the RP provided a number of submissions detailing the SI classification approach for the MCL (Ref. 74) (Ref. 75) (Ref. 76) (Ref. 77). The key submission was 'MCL SI Classification Approach' (Ref. 77), which presented the process control requirements and explained the strategy, key steps, work management, timescale and quality assurance/control for the MCL SI classification.

221. In my opinion, the information presented demonstrated the RP had developed an adequate process to establish the SI classification of the MCL. This aligns with the

expectations of ONR SAP ECS.2, such that the approach presented is sufficient to explain how the MCL has been classified from an SI perspective, on the basis of its significance to safety.

222. Nonetheless, in my opinion, the RP's presentation of information within Ref. 72 could be improved to highlight which documents underpin the key decision-making steps in the process of the MCL classification. I consider this matter to be a minor shortfall as it is a matter of presentation, rather than as a result of any technical shortfall.

223. To address ROA.2, the RP needed to demonstrate how adequate consequence analyses (direct and indirect) informs the SI classification of the MCL. In response, the RP provided the 'MCL Consequence Analysis Report' (Ref. 74), which detailed the failure modes, postulated break locations and analysis of direct and indirect consequences of MCL failure. The consequence analyses for the fuel assembly, primary circuit components (RPV, RCP, SG and associated supports) and civil structure were also considered.

224. The RP also considered how failure of the MCL may give rise to internal hazards such as missiles, steam or hot gas release, collisions, pipe whip, which could potentially compromise other safety related structures and equipment. This approach aligns with ONR expectations (Ref. 5), that appropriate consideration has been given to the effects of internal hazards on safety related structures, and of the secondary effects of structural failure. The RP's analysis concluded that the consequences of failure to these SSCs as a result of a LB-LOCA are intolerable (EHA. 6).

225. I sought feedback from ONR Fault Studies (FS), Internal Hazards (IH) and Civil Engineering (CE) specialists on the scope, analyses and conclusions presented within the 'MCLs Failure Consequence Analysis Report' (Ref. 74). The feedback given was that the approach taken, and the conclusions drawn by the RP (Ref. 74) are reasonable, and so the consequences of a MCL failure are intolerable. I consider this shows alignment with ONR's expectations under ONR SAPs EHA.1 and EHA.6 (Ref. 2), that the RP has developed an effective process to identify and characterise all hazards and consequences associated with failure of the MCL that could affect the safety of the UK HPR1000.

226. The RP's conclusion that the component is a HIC meant that further assessment is required to determine whether any further measures, despite the HIC classification, were reasonably practicable to implement to reduce risk. The driver here was a concern that by assigning a HIC classification, the UK HPR1000 plant may have limited design provision to afford protection against LOCAs in excess of the size of the SL, i.e. a cliff-edge effect with limited defence in depth.

227. In addition, the RP needed to demonstrate the risks to the MCL, as a HIC, from adjacent component failure were reduced to ALARP. This is a generic topic that affects all HICs and is the subject of RO-UKHPR-1000-0046 (Ref. 49). Assessment of this topic was led by the ONR IH Inspector and is discussed in the Internal Hazards assessment report, ONR-NR-AR-21-012 (Ref. 59).

228. For ROA.3, the RP needed to provide a demonstration that the SI classification of the MCL is commensurate with reducing risks to ALARP, with a balanced consideration of the benefits, detriments and application of gross disproportion i.e. ALARP optioneering taking cognisance of the HIC classification. The RP provided a High-level ALARP Assessment for the MCL SI classification with details the identified risks from MCL failure and the feasibility of potential design modifications to either reduce the likelihood of the MCL's gross failure or mitigate the consequences (Ref. 75).

229. The potential engineering design options included barriers, restraints, layout changes and removal of welds. These mostly require civil structural changes to minimise the

indirect consequences of MCL failure, which influence whether implementation is considered reasonably practicable. I sought advice from the ONR IH and CE inspectors on whether the options identified, approach taken, and conclusions drawn were reasonable. In general, the ONR specialists from the IH and CE were satisfied that, albeit at a high-level, that it was reasonable to demonstrate that the SI classification of the MCL as HIC is ALARP, which aligns with ONR's expectations under the ONR SAP EMC.2, para. 293 (Ref. 2).

230. A question was raised relating to RP's consideration of risk reducing measures for the indirect consequences of over-pressurisation (e.g. blow out panels/venting) in specific rooms, to mitigate the generation of failed concrete missiles because of MCL failure. These were responded to in RQ-UKHPR1000-0894 (Ref. 28). A decision on whether to undertake any further assessment of the RP's approach to justifying measures to reduce risks related to compartment over-pressurisation and the resultant consequences to the civil structure as a result of MCL failure, are within the remit of ONR's Civil Engineering (Ref. 78) and IH (Ref. 59) disciplines, respectively.

231. For ROA.4 the RP needed to demonstrate an adequate SI safety case for the MCL, which is informed by the structural integrity classification. The RP provided a revised SI safety case methodology (Ref. 26), with amended versions of the key MCL safety case submissions, including Chapter 17 of the PCSR (Ref. 79), MCL Component Safety Report (Ref. 80) and a document summarising the conclusions of the SI classification assessment (Ref. 76).

232. My focus for ROA 4 was on the adequacy of how the MCL SI classification fits within the overall SI safety case for the UK HPR1000. The highest-ranking PCSR document Chapter 17 – Structural Integrity (Ref. 79) identifies the MCL as a HIC classified item and makes reference to the specific SI safety case of the MCL as a HIC being presented in the CSR (Ref. 80).

233. The CSR makes reference to the 'Safety Case Methodology for HIC and SIC Components' (Ref. 26) which provides a clear link to how the CSR has been structured, and the provisions necessary to underpin an SI safety case, including ONR expectations under ONR SAP ECS. 3 (Ref. 2) for the use of appropriate design codes and standards. The structure of the MCL CSR (Ref. 80) aligns with the methodology presented in the RP's 'Safety Case Methodology for HIC and SIC Components' (Ref. 26) for a HIC structured safety case. In my opinion, the RP's 'Safety Case Methodology for HIC and SIC Components' (Ref. 26) contains sufficient detail on how the RP will provide evidence to demonstrate how the necessary level of integrity will be achieved for the MCL SI safety case, which aligns with ONR expectations under ONR SAP EMC. 3 (Ref 2).

234. This notwithstanding, there is no mention within the CSR of the overall 'MCL Classification Conclusion' document (Ref. 76), which consolidates the findings of the other three reports (Ref. 77) (Ref. 74) (Ref. 81).This effectively presents the judgement of HIC and the need for it to be reflected in the PCSR (Ref. 79) and CSR (Ref. 80). Given that the link to the three supporting documents for the HIC classification is presented within the CSR, I consider this matter to be a minor shortfall.

235. Provisions for consideration of the HIC classification on the fault/hazard schedule are mentioned in the 'MCL SI Classification Approach' (Ref. 77), which identifies the need to check and update related documents and PCSR chapters as a result of the MCL SI classification conclusion. These include the following PCSR Chapters: 12 for FS; 16 for CE; 19 for IH; and 6 for the Reactor Coolant System.

236. The reflection of the HIC status in the fault schedule has been identified as an area of discrepancy with respect to the treatment of LBLOCA as a design basis event. This does not demonstrate consistency between the SI and FS safety cases, with a

potential gap related to ONR expectations under FA.5 paragraph 628 (b), (Ref. 2) as to how failures of structures, systems or components, for which appropriate specific arguments for preventing the initiating fault have been made (i.e. given it is now HIC). In my opinion, this bears no impact on the intent of RO-UKHPR1000-008 - ROA 4, with respect to the demonstration of an adequate MCL SI safety case. The overall position on LBLOCA is addressed by ONR's FS specialist (Ref. 58). From an SI perspective the position for the GDA is that a LBLOCA is not within the design basis, and it is currently not proven whether a deterministic LBLOCA case can be made using best estimate assumptions. Thus, a HIC claim is warranted. This is addressed by ONR's FS specialist in ONR-NR-AR-21-014 (Ref. 58).

Conclusions on the SI Classification of the MCL (RO-UKHPR1000-0008)

237. I was satisfied that the HIC classification for the MCL could not be avoided. This is based on my assessment of the RP's response to RO-UKHPR1000-0008 (Ref. 49), notably:

- The RP developed an adequate process to establish the structural integrity classification for the MCL (ROA 1).
- The RP demonstrated that the process developed under Action 1 of this RO has been appropriately implemented for the MCL, providing necessary evidence that analyses of direct and indirect consequences of MCL failure have been used to inform the SI classification process (ROA 2).
- I am satisfied that the implementation of the process developed under action 1 for the MCL has been carried out appropriately, and that the RP has demonstrated that the final classification of the MCL as a HIC is commensurate with reducing risks to ALARP (ROA 3).
- The RP has produced an adequate strategy for providing a structural integrity safety case for the MCL that is informed by the SI classification (RO Action 4).

238. There is a need to demonstrate that the MCL is adequately protected from the postulated failure of SSC. Through closure of RO-UKHPR1000-0046 (Ref. 49), ONR's IH inspector was satisfied that the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is dependent on internal hazards assessments and is discussed in ONR-NR-AR-21-012 (Ref. 59).

SI Classification of the Main Steam Line (MSL) and Major Valves (MSIV, MSRIV, MSSV) (RO-UKHPR1000-0058)

239. During my GDA Step 3 assessment (Ref. 7), further work was identified to fully understand the RP's proposed classification for the MSL and to ensure suitable and sufficient evidence is available to demonstrate that the classification is reasonable. The RP completed its analysis of the direct and indirect consequences of failure for the MSL, which concluded that the section of MSL within containment would be classified as HIC, whereas the section of MSL outside of containment, but within the Safeguards Buildings, would be classified as SIC-2.

240. ONR's preliminary review of the information provided to underpin the SIC-2 classification for the MSL outside containment concluded that there was insufficient information provided, to demonstrate that a robust consequences analysis had been completed. A robust consequence case is expected to justify non-HIC SI classifications (ONR SAP EMC. 3, Paragraphs 298-300, EHA.5-6, Ref. 2. In RQ-UKHPR100-0970 (Ref. 28), I sought further clarification and reference to the evidence underpinning the consequences analyse, the HIC boundaries for the MSL and whether additional measures were proposed for the break preclusion areas e.g. the containment penetrations in the Safeguards Buildings. The RP clarified some aspects of the HIC boundary but indicated that analysis work was on-going. However, the use

of the break preclusion measures for the MSL (referred to as 'super-piping') would be invoked if the SI classification of the MSL was non-HIC, otherwise the additional measures for a HIC SI classification would be implemented.

241. The RP subsequently completed further consequence analyses and identified that the MSL located within the Nuclear Island (NI) will now be classified as HIC, specifically between the SG steam outlet nozzle weld and the first weld outside the Safeguard Buildings (BSA and BSB), leading to the Conventional Island (CI).

242. This was a significant change from the RP's previous position and prompted some uncertainty in relation to the reasoning and implementation of the RP's approach to justify the SI classification of the MSL and major valves in the Safeguard Buildings. In addition, there would be a significant increase in the HIC boundary for the MSL, including HIC classified welds and subcomponents (including the MSIVs). It was also important to confirm the HIC boundary outside the Safeguard Buildings, and this was dependent on the indirect consequences of a pipe whip. This was the subject of RQ-UKHPR1000-0925 raised by ONR's Internal Hazards discipline (Ref. 28).

243. I considered there was a lack of robust information presented to justify the RP's HIC SI classification and raised RO-UKHPR1000-0058 (Ref. 49). The main purpose of the RO was to ensure that the SI classification of the MSL and associated major valves in the Safeguards Buildings were clearly defined, justified and aligned with the plant classification of SSCs, commensurate with reducing risk ALARP. Several actions were raised:

- ROA1 – MSL Safeguard Buildings Consequence Analyses, Design Optioneering and Identification of Measures to Reduce Risk.
- ROA2 – Justification that the Structural Integrity Classification of the MSL and Major Valves in the Safeguard Buildings is Commensurate with Reducing Risks ALARP.
- ROA3 – Demonstration of the Adequacy of the Structural Integrity Safety Case for the MSL and major valves within the Safeguards Buildings.

244. The RP subsequently developed a resolution plan, which cited a number of submissions that would provide the necessary evidence to underpin the safety classification of the MSL and associated valves within the Safeguards Buildings.

245. ONR's SI discipline led the assessment of RO-UKHPR1000-0058 (Ref. 49), but with the support of ONR's Fault Studies and Internal Hazards disciplines. The key ONR SAPs considered within my assessment of the RP's SI MSL classification included EKP.3, ECS.2-3, EMC.1-3, the EHA series (Ref. 2) and the relevant TAG NS-TAST-GD-016 (Ref. 5).

246. A summary of how the RP progressed RO-UKHPR1000-0058 and my assessment of the RP's submissions is provided below with a detailed record available in 'Assessment of the Response to RO-UKHPR1000-0058' (Ref. 82). The RP adopted a similar approach to the SI classification of the MSL, as for that provided for the MCL (paragraphs 215 to 238 of this AR). As I was generally content with the process developed for the MCL, my focus was on the RP's application of this approach for the MSL, within the context of RO-UKHPR1000-0058 (Ref. 49).

Review of ROA1-3 of RO-UKHRP1000-0058

247. In response to ROA1 the RP provided the 'MSLs SI Classification Analysis Report' (Ref. 83), which defined the scope of the consequence analysis, the key assumptions made, and the bounding cases selected to underpin the classification of the MSL. This submission also includes details of the failure modes, postulated break locations and

analysis of direct and indirect consequences of postulated MSL or associated major valve failures within the Safeguards Buildings.

248. In brief the direct consequences of failure of the MSLs and MSIV inside the Safeguards Buildings are unacceptable from a core damage perspective. Similarly, the indirect consequences of failure are unacceptable for the MSL, MSIV and containment penetration from a pipe whip, steam jet impingement, and combined hazards perspective.

249. To establish whether there was alignment with claims made elsewhere in the UK HPR1000 generic safety case, I sought feedback from ONR FS and IH specialists. The feedback confirmed that these were in line with ONR expectations and claims made elsewhere in the generic safety case (ONR SAP EMC.3, Ref. 2). The direct and indirect consequences analyses for the MSL are based on selected bounding cases and the FS and IH specialists considered the principles and bounding cases selected to be reasonable for the purposes of determining the classification of the MSL and major valves within the Safeguards Buildings.

250. Nonetheless, in several instances there appeared to be a lack of evidence, or insufficient referencing to the analyses undertaken to substantiate the conclusions drawn (ONR SAP EHA.6, Ref. 2). These shortfalls were predominantly associated with the conclusion for the MSL major valves located within the Safeguards Buildings, specifically the MSRIV and MSSV, where the consequences of failure were judged to be 'acceptable'. Several of these concerns had previously been raised by the ONR IH specialist in RQ-UKHPR1000-0925 (Ref. 28).

251. I raised RQ-UKHPR1000-1727 (Ref. 28) and the RP responded with further evidence to demonstrate that failure of the MSRIV/MSSVs were tolerable with a commitment to include this information in an update of the 'MSLs SI Classification Analysis Report'.

252. Following review of the RP's draft response to RQ-UKHPR1000-1727, the ONR IH specialist considered the full hazard profile from the MSRIV and MSSV perspective had not been adequality presented, with respect to avoiding damage to the MSL or MSIV (which have both been identified as HIC in accordance with the RP's classification approach for the MSL - (Ref. 84). This topic falls within the scope of RO-UKHPR1000-0046 – 'Demonstration that the Risks to HIC Components from Internal Hazards are Reduced to ALARP' (Ref. 49), and is discussed in the internal hazards GDA Step 4 assessment report AR, (Ref. 59).

253. For ROA.2, the RP needed to demonstrate that the SI classification of the MSL and major valves located within the Safeguards Buildings is commensurate with reducing risks to ALARP, with a balanced consideration of the benefits, detriments and application of gross disproportion i.e. ALARP optioneering. To address the requirements of this action, the RP provided a 'High-level ALARP Assessment for the MSL SI Classification' (Ref. 81). This report reviewed the identified risks from (Ref. 83) and the feasibility of introducing potential design modifications to reduce the likelihood of the MSL gross failure or mitigate consequences caused in order to avoid a HIC claim. This was a multi-discipline activity that took account of the impact of design modifications on other safety case aspects and SSCs.

254. A companion submission for ROA2 (and ROA3) was the 'MSL SI Classification Approach' (Ref. 85) which presented the approach and process control requirements for SI classification of the MSL and major valves, explaining the strategy, key steps, work management, timescale and quality assurance/control arrangements applied.

255. The RP's selection criteria 'screens out' certain options, allowing others to be taken forward for further consideration. One of the options initially screened out are those that rely on high reliability/integrity claims. I raised RQ-UKHPR1000-0925 (Ref. 28), to

seek clarification on this point. The RP explained that a claim of high reliability is intended as a 'last resort' and is initially removed from the list of available options, so that the process can fully consider all other options first. In my opinion, this has shown that the RP's process recognises the burden associated with demonstrating high reliability claims (para. 286, Ref. 2 and, Sub-section 5.18, Ref. 5) and has adopted an approach to avoid this, where reasonable to do so.

256. The 'MCL Classification Conclusion' document (Ref. 76) contains evidence of a multi-discipline optioneering workshop that was held to discuss the design options available to reduce risk for the 'unacceptable' scenarios. A broad range of disciplines (including safety, engineering and manufacturing experts) contributed to the workshop, which aligns with my expectations for ensuring a comprehensive and broad spectrum of technical scrutiny has been applied.

257. A range of engineering design options (such as barriers, restraints, reactor system layout changes and strengthening of the Safeguards Buildings civil structure) that could reduce the risks associate with failure of the MSL or associated valves were identified (ONR SAP EKP.3, ELO.4, Ref.2). These mostly require radical layout changes for the MSL, or civil structural changes to minimise the indirect consequences of MSL or major valve failure, which influenced whether implementation is considered reasonably practicable. Overall, the RP recognises that, whilst some modifications identified could reduce the risk associated with several of the indirect consequences of MSL or MSIV failure, the overall requirement for a HIC claim was still present due to the direct consequences of failure.

258. I sought advice from the ONR IH and FS inspectors on whether the options identified, approach taken, and conclusions drawn were reasonable (Ref. 86). Feedback from the FS and IH inspectors confirmed that the direct consequences of MSL or MSIV failure remained bounding, and therefore the HIC classification was reasonable. I was satisfied that the HIC claim was justified, and that the RP had taken a proportionate approach to assessing the benefits of adopting design changes to mitigate the indirect consequences of failure, knowing that failure was still intolerable.

259. This notwithstanding, the RP's high-level ALARP review identified two design modifications that it considers could reduce the risk of indirect consequences of MSL failure, to ALARP (ONR SAP EKP.3, Ref. 2). These measures have been implemented into the UK HPR1000 generic design, in accordance with the RP's GDA design modification process, as Category 3 modifications under entry number M86. This has been accepted for inclusion within GDA assessment by the ONR IH inspector and is reported under RO-UKHPR1000-0046 'Demonstration that the Risks to HIC Components from Internal Hazards are Reduced to ALARP' (Ref. 49). In addition, as the MSRIV and MSSV are non-HIC components (see Sub-section 4.2, above) in close proximity to the HIC-classified MSL and MSIV, they also fall within the scope of RO-UKHPR1000-0046 and are discussed in ONR-NR-AR-21-012 (Ref. 59). In summary, the IH inspector was satisfied that for RO-UKHPR1000-0046, the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is discussed in the ONR-NR-AR-21-012, (Ref. 59).

260. Informed by the SI classification, ROA.3 covered the need for the RP to demonstrate that an adequate 'structural integrity safety case' for the MSL and major valves located within the Safeguards Buildings could be developed. In response the RP provided the 'SI Classification Approach for the MSLs' (Ref. 85), which identifies the need to undertake the consequence analysis completed under (Ref. 83). The RP has also recognised the need to amend versions of the key MSL safety case submissions, including Chapter 17 of the PCSR (Ref. 79) and the 'MSL Component Safety Report' (Ref. 54).

261. The MSL CSR (Ref. 54) is the key link between the highest level of the hierarchical generic safety case Pre-Construction Safety Report - PCSR Chapter 17 – (Ref. 79), through to the supporting technical submissions provided for the MSL and major valves in the Safeguards Buildings.

262. A summary for the MSLs 'SI safety case' is reported in the 'Main Steam Lines Component Safety Report' (Ref. 54), which clearly identifies the processes followed and technical documents containing the key supporting 'legs' of the MSL safety case. The CSR makes reference to the RP's Safety Case Methodology for HIC and SIC Components (Ref. 26), which provides a clear link to how the CSR has been structured, and the provisions necessary to underpin an SI safety case. The structure of Ref. 51 aligns with the methodology presented in Ref. 26 for HIC and SIC structured safety cases. In addition, the traceability of the analysis, optioneering and conclusions drawn throughout the MSL SI classification process is present, including steps taken to revise key documents in accordance with the methodology and strategy documents referenced (ONR SAP EMC.3, Ref. 2).

263. Provisions for consideration of the SI classification on the fault/hazard schedule are mentioned in Ref. 80, which also identifies the need to check and update related documents and PCSR chapters as a result of the MSL SI classification conclusion. These include PCSR Chapters: 11 (Mechanical Engineering), 12 (Fault Studies) 16 (Civil Engineering), and 19 (Internal Hazards). I discuss this topic further as part of the consolidation of the safety case (Sub-section 4.8 of this AR).

264. Overall, based on the information I have sampled, the RP's responses to queries raised in RQ-UKHPR1000-1727 (Ref. 28) and feedback provided by ONR IH and FS specialists, I am satisfied that the information provided under ROA 3 is sufficient to demonstrate that the RP has an adequate MSL SI safety case.

265. In response to RQ-UKHPR1000-1727 (Ref. 28) the RP clearly defined the boundary of the MSL with respect to the classification analysis; the section of MSL within the CI is defined as 'NC' or 'non-categorised classification', which according to Ref. 39 means it will conform to conventional industrial requirements (i.e. no specific nuclear code needs to be used for design or manufacture). Whilst the focus of this RO is on the process and outcome of MSL classification within the Safeguards Building, the RP has recognised the boundary between the NI and CI sits at the Safeguards Buildings penetrations, and as such, should consider indirect consequences of MSL pipework failure within the CI that is immediately downstream of the boundary. This accords with meeting the intent of ONR guidance provided in ECS.2 and paragraph 167 of the ONR SAPs (Ref. 2), which states "appropriately designed interfaces should be provided between (or within) structures, systems and components of different classes to ensure that any failure in a lower-class item will not propagate to an item of a higher class".

Conclusions on the SI classification of the MSL and Major Valves (RO-UKHPR1000-0058)

266. I am satisfied that the RP has provided sufficient information to justify the classification of the MSL and MSIV within the safeguards buildings as HIC. I am also satisfied that the SI classification of the MSRIV and MSSV (SIC-2) is appropriate, based on the RP's demonstration that the consequences of postulated gross failure (direct and indirect) are 'tolerable'. This is based on my assessment of the RP's response to RO-UKHPR1000-0058 (Ref. 49), notably:

- The RP developed an adequate process to establish the SI classification for the MSL and associated major valves in the Safeguards Buildings (ROA 1).
- I am satisfied that the implementation of the process developed under ROA 1 has been carried out appropriately, and that the RP has demonstrated that the

final SI classification of the MSL and major valves in the Safeguards Buildings, is commensurate with reducing risks to ALARP (ROA 2).

- The RP produced an adequate strategy for providing a 'structural integrity safety case' for the MSL and major valves in the Safeguard Buildings, that is informed by their SI classification (ROA3).

267. This notwithstanding, there is a need to demonstrate that the MSL and MSIV are adequately protected from the postulated failure of SSC. In summary, the IH inspector was satisfied that for the requirements of RO-UKHPR1000-0046 (Ref. 49), the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is discussed in the ONR IG assessment report, ONR-NR-AR-21-012 (Ref. 59).

SI Classification of the RCP (Casing and Flywheel)

268. I looked at the reasoning behind the SI classification of the RCP (casing and flywheel). The RP identified that a postulated rupture or fracture of the casing body is more serious than a rupture or fracture of the inlet and outlet of the casing. The direct consequences include a LBLOCA equivalent to that resulting from a double-ended rupture of the MCL, with unacceptable consequences. The LBLOCA load is also significant such that the consequences for the fuel assembly, primary equipment and civil structure were deemed unacceptable (Ref. 46).

269. The indirect consequences include pipe whip, jet impingement, internal flooding, missile, mass and energy release in the compartment. The internal flooding consequences fall within the design basis, whereas the consequences associated with the remaining internal hazards give rise to unacceptable damage to the core, compartment and containment for which it is not reasonably practicable to afford protection (Ref. 46) (Ref. 87).

270. In my opinion as the RCP is connected to the MCL and is integral to the primary circuit a gross failure of the RCP is likely to impact the MCL, with in broad terms the same level of consequences. Thus, I was satisfied that SI classification of HIC for the RCP casing was appropriate.

271. The consequences of a gross failure of the RCP flywheel are also considered by the RP. The direct consequences include additional vibration, RCP overspeed or RCP seizure. The RP claims that the former can be detected, and the RCPs can be tripped to prevent excessive vibration that could challenge the RCP integrity. The latter two effects can have direct consequences for heat removal from the core (Ref. 46) (Ref. 88). ONR's FS inspector has provided me with assurance that the consequences of both RCP overspeed and RCP seizure have been considered within the design basis analysis, which I consider to be acceptable.

272. However, the indirect consequences involve large, energetic missiles with energies of up to circa 64755kJ, which will rupture the wall of the RCP compartment and with the attendant potential to damage the MCL and other components e.g. the RPV and SG, which are classified as HIC. The RP considered whether it was reasonably practicable to afford protection by increasing the compartment wall thickness (a factor of at least 2.2 would be needed), but the subsequent implications for space in the SG compartment, RPV internals storage area and Examination, Maintenance, Inspection and Testing (EMIT) activities and overall plant layout were deemed unacceptable (Ref. 46) (Ref. 87).

273. I was content that the RP had provided sufficient evidence to justify its SI classification of HIC for the RCP (flywheel).

Conclusions on SI Classification of RCP (Casing and Flywheel)

274. I was satisfied that the consequences of a postulated gross failure of the RCP (casing and flywheel) were unacceptable, it was not reasonably practicable to afford protection, and hence that a HIC classification was appropriate.

275. As the RCP casing is designated HIC, its protection falls within the consideration of RO-UKHPR1000-0046 (Ref. 49). A specific regulatory query, RQ-UKHPR1000-1503 was raised by ONR's IH inspector to seek clarification of the protection claims for the RCP (casing and flywheel) (Ref. 28). In summary, the IH inspector was satisfied that for the requirements of RO-UKHPR1000-0046, the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is discussed in ONR-NR-AR-21-012 (Ref. 59).

SI Classification of the Pressuriser Surge Line (SL) (Ref. 89)

276. In my opinion the RP's identification of the SL as a potential HIC candidate is indicative of its cautious approach to the identification of HIC components. In particular, if the position was uncertain, awaiting consequence analyses work or the introduction of a potential design modification, the RP's default position was to assume that a HIC classification was needed. This approach enabled the development of a HIC safety case to proceed with an understanding of the risks by both organisations.

277. Nonetheless, I was surprised to see the pressuriser surge line (SL) as a HIC candidate. In consultation with ONR's fault studies inspector, I assessed the RP's direct consequence analyses and judged that they were acceptable. In addition, the RP confirmed there were no 'cliff-edge' effects for leaks in excess of a full-bore guillotine failure of the SL. The RP also assessed the indirect consequences (jets, pipe whip mass energy release, internal flooding) and concluded that these are tolerable without design modifications to either SSC or civil structures (Ref. 90) and that the risks were reduced to ALARP (Ref. 91).

278. With the support of the IH and CE disciplines, I assessed the indirect consequences of failure on the PZR or MCL (HIC components) and the surrounding civil structure.

279. The ONR CE inspector reviewed the claims made for the resilience of the civil structure to withstand SL failure. The CE inspector concluded that the information presented was at a high level and not sufficiently detailed to undertake a thorough assessment. This was considered to be a minor shortfall, but acceptable, on the expectation that more detailed information would be made available during safety case development required to support site-specific phases. It was therefore concluded that, for the purposes of GDA, the information presented to support tolerability of SL failure on the civil structure was adequate.

280. Within the assessment of RO-UKHPR1000-0046 (Ref. 49), the risks of SSC failure causing damage to adjacent HICs, such as the PZR and MCL, have been considered by the RP in 'Substantiation of Pressuriser Against IH' (Ref. 92) and 'Substantiation of Main Coolant Lines Against IH' (Ref. 93), respectively. In these documents, risk of damage from explosion, high energy pipe failure (HEPF) and jetting has been considered as a result of SL failure. The IH inspector raised RQ-UKHPR1000-1502 and RQ-UKHPR1000-1504 related to these components (Ref. 28) to determine whether the RP had undertaken sufficient and robust analyses to underpin the conclusions drawn.

281. Overall, the IH inspector was satisfied that for GDA, the RP has demonstrated that it has undertaken sufficient identification, screening, and analysis of hazards and their impact on the HICs in-line with ONR expectations. The RP has also demonstrated that

where hazards have not been eliminated, the integrity of the HICs can be demonstrated. I therefore consider that the indirect consequence of failure of the SL is tolerable from the perspective of inducing damage to adjacent HICs.

282. The IH assessment identified a generic shortfall related to assessment methods and substantiation of the modifications once at detailed design. However, the IH inspector judged that these shortfalls did not undermine the conclusions of the RO-UKHPR1000-0046 (Ref. 49) assessment and was satisfied that the RO could be closed. These shortfalls are a matter for the IH report and will be managed within ONR-NR-AR-21-012 (Ref. 59).

Conclusions on SI Classification of the SL

283. From the feedback received from the ONR CE inspector and closure of RO-UKHPR1000-0046 (Ref. 49) by the ONR IH inspector, I am satisfied that the RP has provided adequate reasoning and justification to support the classification of the surge line as a non-HIC component. The adequacy of consequences analyses undertaken by the RP to support this outcome is considered further within the IH assessment, ONR-NR-AR-21-012 (Ref. 59) , from which no conclusions are drawn that undermine the outcome of my judgement.

SI Classification of SG Components

284. As with the SL it was unusual for the RP to identify the SG blowdown lines as HIC Candidates. However, the RP identified the potential for a common mode failure arising from the layout of the SG blowdown lines in Safeguards Building, which could lead to the blow down all three SGs with unacceptable consequences.

285. The potential vulnerability of the SG blowdown lines to common mode failure was as a result of the RP considering the consequences of postulated gross failure and in this case, it was the indirect consequences, from a pipe whip, which had the potential to negate the engineered redundancy. The RP subsequently developed two potential design modifications that would prevent a common mode failure of the SG blowdown lines. I noted that each option presented different means of reducing the risk posed by failure. The first option appeared to eliminate the risk by changing the piping layout whilst the other appeared to prevent the risk by introducing isolation. It is ONR's view that eliminating the risk presents the lower level of risk and hence is, where practical, preferential. I raised RQ-UKHPR1000-0670 (Ref. 28) with my key questions covering the RP's approach to selecting the ALARP options and its understanding of the risk hierarchy in selecting the ALARP option. This is discussed in Sub-section 4.7 of this assessment report. It will suffice to say that on balance, the first option introduced several adverse consequences for the plant layout, which the RP judged to outweigh the benefits. The second design option was therefore proposed by the RP to avoid a HIC claim.

286. ONR's IH Inspector raised RQ-UKHPR1000-1338 (Ref. 28) to seek clarification on the classification of the modification, timescales of the modification and the implications for the IH safety case. These topics are addressed in the ONR GDA Step 4 IH assessment report ONR-NR-AR-21-012 (Ref. 59).

287. The primary and secondary shells along with the tubesheet were confirmed as HIC components early in the GDA. However, I raised RQ-UKHPR1000-0371 (Ref. 28) for the RP to clarify the SI classification of certain internal and external components in the UK HPR1000 SG, taking cognisance of ONR SAPs ECS2 and ECS.3 (Ref. 2). Specifically, in RQ-UKHPR1000-0371, I queried: the SI classification of the SG supports; the SG manway and studs; the divider plate; and the steam nozzle venturi. The first two components feature redundant sub-components and are discussed under

related SI classification topics (paragraph 316-332 below). My focus in this part is therefore on the divider plate and steam nozzle venturi.

288. The RP assigned a preliminary SI classification of SIC-1 for the postulated failure of the divider plate. The SI classification took account of the loss of the heat removal capacity and redundancy afforded by the two unaffected SGs. An SI classification claim of SIC-1 for postulated failure of a divider plate was reaffirmed by the RP (Ref. 94). No indirect consequences were identified, but I discussed the direct consequences of a divider plate failure with ONR's FS assessor and confirmed that an SI classification of SIC-1 was appropriate, and that the RP's arguments are sensible.

289. Steam nozzle venturi is a Quality Class 1 component, but no SI classification was identified. To justify the absence of an SI classification it is important to understand the claims made against any steam flow restrictions and the consequences if this requirement is not satisfied when demanded. The RP subsequently confirmed that the steam nozzle venturi was assigned an SI classification of SIC-1 (Ref. 94), on the basis that the consequences were bounded by steam line pipe break. I have consulted ONR's FS inspector on this matter, who considers that the RP's arguments are sensible.

Conclusions on SI Classification of SG Components

290. From my discussion with and the feedback provided by the FS inspector, I am satisfied that the SIC-1 classification for the SG components identified is reasonable and justified with appropriate analysis. The use of bounding cases in both instances is sufficient and I shall consider the SI case presented for these components commensurate with the assigned classification in conjunction with my broader sample of the SGs.

SI Classification of Reactor Vessel Internals (RVI)

291. In RQ-UKHPR1000-0247 and RQ-UKHPR1000-0279 (Ref. 28), I sought clarification of the consequences of postulated gross failures of the welds in the RVI and FTT, respectively (Ref. 28). My queries also covered whether it was reasonably practicable to avoid welds, in particular, longitudinal welds (ONR SAPs EMC.8 and EMC.9, Ref. 2). The design aspects of the RVI and FTT are addressed in Sub-section 4.7 of this assessment report.

292. With regard to the SI classification of the RVI, my focus was on the most safety significant shell welds in the core barrel. I questioned the scope of the RP's consideration of the consequences of failure for the core barrel, which appeared to be limited to the circumferential welds. I also queried how the RP intended to present the safety case for the core barrel.

293. The RP explained that the failure of the upper circumferential weld between the core barrel flange and upper core barrel shell (weld 1) would result in the most serious consequences for the radial support keys and the RPV bottom head. This was therefore the bounding weld for the consideration of the consequences of failure.

294. A failure of weld 1 will result in the downward movement of the core and barrel. However, the drop is limited by the radial support keys that are welded to the RPV lower head. The downward movement of the core barrel will be about 20mm (cold condition) and about 9.8mm (hot condition). These distances are less than the engaged length of fuel assembly upper alignment pins, so that an acceptable flow channel geometry and control rod entry/drop provision are maintained.

295. The indirect consequences included the effects on the radial support keys and RPV bottom head. The core drop analysis shows that the radial support keys can absorb the

core drop energy so that the effects on radial support keys and RPV bottom head were tolerable. The RP also considered partial failure of weld 1, where the main concern was tilting of the core. However, the design includes as small gap between the radial support keys and keyways on the lower support plate, to limits any inclination. I specifically consider the scope of RP claims against alignment critical SSCs for the UK HPR1000, which is discussed under Sub-section 4.1.2.2 of this report.

296. The RP also committed to updating its safety case by providing 'SI Classification Report for Reactor Vessel Internals', (Ref. 95). In consultation with ONR's Fault Studies, Fuel & Core and Internal Hazards Inspectors, I subsequently confirmed the consequences of failure of the core barrel were tolerable, with the radial support keys providing a line of protection. I was therefore satisfied that an SI classification of SIC-1 for the RVI was appropriate, on the understanding that adequate heat removal can be maintained following failure of the core barrel and that adequate margin to fuel failure is predicted (Ref. 95) (Ref. 46).

Conclusions on SI Classification of the RVI

297. Overall, I am satisfied that the RP has provided sufficient information to substantiate the SIC-1 classification for the RVIs. My judgement is based on the information I have sampled with the provided consequences analysis and classification documents and the discussions I've held with ONR FS, F&C and IH inspectors.

Fuel Transfer Tube (FTT)

298. Drainage from the flooded compartments within the reactor and fuel buildings that are not isolable (e.g. FTT, drain line penetrations or man access points within flooded compartments) could lead to various consequences. Nonetheless, the basis of the justification for the structural integrity classification of sections of pipework that are not isolable was not clear from the safety case (Ref. 96).

299. I raised RQ-UKHPR1000-0279 (Ref. 28) to gain further clarification of the design and construction of the FTT, the penetrations and drain lines in the flooded compartments in the reactor and fuel buildings. I also sought to understand the unmitigated consequences of postulated gross failures, and which failure locations had been considered to inform the SI classifications.

300. The RP explained that during the refuelling stage, the flooded compartments in the reactor and fuel buildings include reactor pool, transfer compartment and spent fuel pool. The FTT connects the reactor pool and transfer compartment. The FTT consists of four sections, which are rolled from stainless steel plates and joined using longitudinal and circumferential welds. For Fangchenggang-3, LBB or limited leakage concepts are applied, such that for the FTT and drain lines (which cannot be isolated), design and construction is to RCC-M2 (equivalent to SIC-2). Whereas failure of other drain lines that will not result in the loss of cooling function or uncovering of fuel assemblies are designed and constructed to RCC-M3 (equivalent to SIC-3).

301. For the UK HPR1000, the RP recognised the need to consider the consequences of gross failure and to inform the SI classification and committed to deliver the 'Fuel Transfer Tube SI Classification Report', which would include a bounding analysis, mitigation measures i.e. to avoid a HIC claim if reasonably practicable and the final SI classification (Ref. 97). In advance of the production of Ref. 93, the RP responded to RQ-UKHPR1000-0479, providing an indication of the consequences of FTT failure along with potential design modification proposals (Ref. 28).

302. The RP's analyses subsequently showed the consequences of a postulated gross failure of the FTT were unacceptable with the potential for significant implications for the integrity of the fuel in transit, the reactor core and the functionality of SSC in the

Fuel Building. The RP considered design modifications to prevent the unacceptable consequences of FTT failure, and after undertaking the ALARP optioneering process, the RP proposed the following two modifications:

- To reduce the consequences of failure, the RP proposed the introduction of watertight doors and water-stops to make the room around the FTT watertight. This will significantly reduce the rate of water flow out of the FTT.
- To reduce the likelihood of failure of the FTT, the RP proposed a modification to the FTT design itself to avoid the number and length of welds needed to fabricate the component. This involved reducing the number of welds from five (two circumferential, three longitudinal) to one single circumferential weld between two-cylinder forgings.

303. I assessed the ‘Fuel Transfer Tube SI Classification Report’ (Ref. 97) in consultation with ONR’s Fault Studies, Fuel & Core, Internal Hazards and Civil Engineering Inspectors. I raised RQ-UKHPR1000-0895 (Ref. 28) to gain a better understanding of the consequence analysis undertaken and the impact the proposed solutions would have on the wider civil engineering safety case. The RP provided a response to address the queries raised, which I made available to ONR specialists in Civil Engineering and Internal Hazards for consideration. From an SI perspective, I was content that the RP had identified reasonably practicable measures to avoid a HIC claim for the FTT. I am therefore satisfied that, through the introduction of the design modifications to the component manufacture and the surrounding structure, an SI classification of SIC-2 for the FTT is appropriate. The RP subsequently presented the FTT modification for consideration in Step 4 of GDA.

304. Assessment of the civil structure and aspects associated with modifications proposed by the RP are not within the scope of my assessment. From my discussions on this topic with the ONR CE inspector, the impact of the proposed modifications on the civil structure have not been specifically assessed within the scope of the ONR GDA CE assessment. This notwithstanding, from a high level review the CE inspector considered that the proposed modifications to the civil structure proposed to mitigate the consequences of FTT failure seemed reasonable. As such, this was judged to be acceptable for the purposes of GDA on the expectation that a more detailed demonstration of safety will be produced as part of normal business within the site-specific phase of safety case development.

Conclusions on SI Classification of the FTT

305. From the information I have sampled and from my discussions with other ONR specialists, I am satisfied that the RP has assigned an appropriate SI classification for the FTT. The RP has demonstrated the use of consequence analyses and ALARP optioneering to avoid the need for presenting a HIC safety claim, based on design optimisation and implementation of engineered protection measures.

SI Classification of Accumulators

306. The Reactor Building (BRX) contains three separate trains of the Safety Injection System (SIS) which include three separate accumulators (ACC). In accordance with RP’s safety categorisation and classifications scheme (Ref. 66), the ACCs deliver a F-SC1 function class and their design provision class is NC (Ref. 98). The SIS is designed to inject borated water to the cold leg in response to RPV pressure drop. The RP undertook a systematic consideration of the direct and indirect consequences of a postulated gross failure of the ACC and reported their conclusions in the ‘ACCs Failure Consequence Analysis Report’ (Ref. 68). The RP concluded that the direct and indirect consequences of a postulated gross failure of the ACC were acceptable and informed by OPEX an SI classification of SIC-2 was appropriate.

307. I assessed the ‘ACCs Failure Consequence Analysis Report’ (Ref. 68) with the support of ONR’s Fault Studies, Internal Hazards and Civil Engineering Inspectors. As there are three ACC and only one is required for delivery of the RIS, the direct consequences were acceptable. The principal hazards from the ACC therefore relate to the indirect consequences, in particular, missiles and flooding, for which I sought the views of ONR’s Internal Hazards Inspector. Notably, two of the three ACC share a dividing wall (Barrier BRE2113VB), I questioned whether in the event of a gross failure of one of the two adjacent ACC, the second ACC would be lost, and if so, whether it was reasonably practicable to prevent the loss of both ACC. The question was subsequently pursued by ONR’s IH inspector. The RP confirmed that the barrier in question would not survive the increased loading and therefore the loss of both accumulators could not be discounted. In addition, given that one further ACC was available to deliver the RIS function, the RP concluded that it was not reasonably practicable to introduce a design modification to prevent the potential loss of the two ACC in the event of a postulated gross failure of the adjacent ACC, as explained in the RP’s response to RQ-UKHPR1000-1766 (Ref. 28).

308. The missile assessment utilises methods described in the R3 Procedure, but it was unclear what assumptions had been made to specify the missile characteristics (e.g. nose shape factor) and why the local damage formulae are relevant in this application. This notwithstanding, the RP’s assessment indicated that failure of one of the two ACC would damage the slab (BRE2604DB) that supports the IVR water tank, which would lead to a loss of the IVR inventory. ONR’s IH Inspector therefore questioned whether the loss of the IVR in this situation was acceptable and also sought assurances as to whether the ACC were sufficiently segregated from the MCL.

309. In response to RO-UKHPR1000-0046 (Ref. 49), the RP produced several reports to assess the risk to HICs from indirect consequences of failure. These included an assessment of the MCL itself (Ref. 93), which identifies the SL as the bounding case for explosion risk to the MCL. According to the RP’s assessment of explosion risk in the reactor building from ‘Internal Explosion Safety Assessment Report for Reactor Building’ (Ref. 99), the RP claims:

“At the data collection step, the explosion sources near the HICs are identified. The pressurised tanks (accumulators) are arranged away from the HICs and segregated with the claimed barriers”.

310. The RP also claims this in the general review of review of plant layout in ‘ALARP Demonstration on Plant Layout in Respect to HIC’ (Ref. 100), which identifies the risk from ACC failure on the MCL and states:

“The ACCs are located away from the MCLs and segregated by shielding walls. According to the ACCs Failure Consequence Analysis Report, their failure cannot impact the MCLs”

311. The RP has therefore concluded that an SI classification of SIC-2 for the ACC was appropriate on the basis that neither the direct nor indirect consequences of failure would cause intolerable damage to a HIC, that in turn could result in core damage with only a minor offsite radiological release.

312. Additional assessment of the RP’s classification of this component has been conducted, which identified shortfalls with the RP’s approach to demonstrating consideration of OPEX and its influence on classification and traceability of evidence within the safety case. These have been which has been discussed under Sub-sections 4.1.2.1 and 4.2.1.2 above.

313. The broader topic of the risk posed to HICs from hazards has been considered and addressed by the IH inspector through RO-UKHPR1000-0046 (Ref. 49). In brief, the IH

inspector was satisfied that the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The final position is discussed in the IH assessment report, ONR-NR-AR-21-012 (Ref. 59).

Conclusions on SI Classification of the Accumulators

314. With closure of RO-UKHPR1000-0046 (Ref. 49) by the ONR IH inspector, I am satisfied that the RP has provided adequate reasoning and justification to support the classification of the surge line as a non-HIC component. The adequacy of consequences analyses undertaken by the RP to support this outcome is considered further by the IH assessment, ONR-NR-AR-21-012 (Ref. 59), with no findings raised that undermine the outcome of this judgement.

315. From an SI perspective, I am satisfied that the SIC-2 classification is reasonable on the grounds of the consequence analysis undertaken, that demonstrates gross failure is tolerable. My judgement has taken account of the assessment undertaken elsewhere in my report related to the accumulators.

Classification of Closure Components and Supports for HIC

316. The safety class of individual components or parts of an SSC is provided in the corresponding CSR. With regard to closure components, I raised RQ-UKHPR1000-0371 (Ref. 28) to query the SI classification of the SG supports and manway closure components, including closure plates and studs. I expected the SI classification to consider the consequences of both single and multiple failures of redundant components. The preliminary classification of the SG manway was HIC, and this was subsequently confirmed in the Parts and Welds SI Classification Report for the SG (Ref. 94) and the SG CSR (Ref. 101). Whereas the SI classification of the supports was not determined and a preliminary SI classification of SIC-1 was assigned for the manway studs with the final SI classifications available in a ‘SI Classification Methodology for the SG’ (Ref. 102), and ‘Parts and Welds SI Classification Report for the SG Parts and Welds’ (Ref. 94). I sampled these documents and the redundant components were classified as SIC-1, but without consideration of the consequences of multiple failures.

317. I progressed the topic further with my TSC and investigated the execution of the RP’s SI classification process by sampling parts of the PZR (Ref. 20).

318. The failure of one of the circumferential welds on the PZR shell would lead to gross failure of the pressure boundary, the potential for energetic missiles, loss of a large volume of reactor coolant, depressurisation of the primary circuit, and ultimately loss of one or more of its critical safety functions. The potential consequences are therefore catastrophic and component failure is intolerable. The RP’s assignment of a HIC SI classification is therefore appropriate, since it is not reasonably practicable to provide protection against such consequences.

319. Similarly, the pressuriser manway cover, studs and nuts were claimed by the RP to be non-HIC (SIC-1) based on information provided in response to RQ-UKHPR1000-1313, and RQ-UKHPR1000-1464, (Ref. 28),which states:

- “The sealing requirement is met even if one stud/nut is failed, and the details are shown in Sealing Analysis Report (Ref. 103).
- The weakest region of manway cover is near bolt holes, so the failure results of manway cover are the same as that of bolt/nut.”
- The manway cover is classified as RCC-M 1 (code class) and Q1 (quality assurance), and strict manufacture will be achieved to ensure high quality. Therefore, the failure probability of manway cover is very low.

- According to analysis, the result margin of stress and fatigue analysis of manway cover is larger than 67.9% and that of bolt is large than 24.8%.
- The containment affords protection, which ensures the leakage confinement and prevents the radioactivity release to the environment."

320. The SIC-1 classification appeared to require discounting the failure mechanism of gross cover plate rupture on the basis that there is an assessment that demonstrates a significant factor of safety against failure. This represents a circular argument and fails to take fully into account all potential failure mechanisms.

321. The direct consequence of a pressuriser manway cover failure are likely to be bound by those of a SL failure. However, the indirect consequences associated with the missile warrant further consideration. Similarly, the potential consequence of multiple stud/ bolt failures in closures for HIC components need to be assessed.

322. Bolted closures are not unique to the pressuriser, and hence this issue is also relevant to the structural integrity classification of similar design features in other HIC vessels and valves of the UK HPR1000. I view this matter as significant to warrant tracking during the site-specific stages.

AF-UKHPR1000-0191 – The licensee shall, as part of detailed design, demonstrate that closure components are appropriately classified and underpinned by the safety case. This should include, but not be limited to, the consequence of failure of covers, studs and bolts for high integrity components.

323. Similarly, the pressuriser support brackets and the welds to the lower cylindrical shell are assigned by the RP as SIC-1 on the grounds that "failure consequences of these welds are the same as the consequences of the bracket support failure. According to the calculation and analysis, when a failure happens to one of these welds, it does not cause the PZR to collapse. Since there is no subversive influence on the PZR, the evaluation is acceptable. According to SI classification method, the welding is classified as SIC-1*"* (Ref. 31) (Ref. 32).

324. The support arrangements for the pressuriser have changed from a continuous skirt to three distinct brackets. Such a tripod arrangement can easily lead to instability if one bracket were to fail.

325. Subsequently, the RP confirmed in response to RQ-UKHPR1000-1464 (Ref. 28):

"There are 8 radial limiting stoppers around the upper section of PZR. Under normal condition, assuming one bracket missing, the PZR topples towards the upper radial limiting stopper on the failure side. At this moment, the upper radial limiting stoppers function as a horizontal support and maintain the PZR in an approximately vertical state. Mechanical analysis is carried out under condition that only two brackets remain, and the upper radial limiting stoppers carry the load, the evaluation result is acceptable."

326. The RP's explanation suggests that failure of a single bracket or its welds is broadly tolerable. It is, however, unclear whether the load transfer to the radial stoppers would be smooth or whether there is the potential for a gap to exist between the shell and the radial stoppers that would allow the pressuriser to topple slightly gaining some momentum before impacting with the stopper. Such an impact, if severe, may compromise the integrity of the pressuriser shell.

327. Analysis may show that the dynamic load transfer between a failed support bracket to the corresponding radial stoppers and remaining brackets is acceptable. The RP has not described whether the postulated failure of the bracket would be immediately detected and various mitigations put in place. If undetected, it would be necessary for

the remaining brackets and the radial stoppers to be able to support all subsequent operating loads and fault / hazard conditions. It has not been confirmed what post-failure assessments have been performed. On this basis, I consider that there is insufficient evidence to show that a classification of SIC-1 for the pressuriser support brackets and the welds to the lower cylindrical shell, is appropriate.

328. The more generic point is that alternative load paths need to be assessed where there is a lack of redundancy in the primary load path for vessel and piping supports in HIC components i.e. the consequences of both single and common-cause failure should inform the SI classification of supports in HIC components. In my opinion the RP has not fully considered these points, but as the RP has recognised the need to consider these postulated failures, I judge this matter as a minor shortfall.

329. The pressuriser manway studs are also redundant components and are assigned a SIC-1 classification by the RP, based on the consequences of a single stud failure. The potential for common cause failure is not considered, i.e. a defective batch of studs, or cascade failure of the bolted connection by virtue of the dynamic transference of load from a failed stud. The RP's reasoning was similar to that offered for the pressuriser supports, RQ-UKHPR1000-1313 and RQ-UKHPR1000-1464, (Ref. 28) with the addition of the following points:

- "ISI, VT and UT will be performed to identify the degradation before component failure according to RSE-M, and the bolt and nut will be replaced if any defect is detected.
- The failure of two or more bolts is expected to cause leakage, and operator will take proper actions to deal with this situation according to the alarm sheet and operating procedure that will be issued in license stage.
- The Requesting Party (RP) has designed and constructed more than 20 PWRs in China, and failure of single bolt has never been encountered in RPV\SG\PZR (including more than 3000 bolts)."

330. The RP therefore concluded: "As described above, only the failure of one single bolt needs to be considered, there is no common cause failure."

331. As noted above the lack of consideration of common cause failure in the studs and the absence of any analysis of dynamic load transfer between studs are of concern. There are also mitigations that could be invoked e.g. sourcing adjacent studs from separate batches or manufacturers and assessing the dynamic load transfer from a failed stud to adjacent studs to ensure that the present design margin is not completely eroded.

332. Bolted cover plates are not unique to the pressuriser, and hence this issue is also relevant to the structural integrity classification of similar design features in other HIC components of the UK HPR1000. I view this matter as significant to warrant tracking during the site-specific stages, which is captured under AF-UKHPR1000-0189 above.

4.2.1.4 HIC Listing

333. In GDA Steps 3 and 4, the RP undertook significant work to inform its view on the SI classification of a wide range of HIC candidate SCC. The RP concluded that the consequences of gross failure were unacceptable and it was not reasonably practicable to avoid a HIC claim for the RPV, PZR, SG, RCP, MCL and MSL (including MSIV). For the other HIC candidates, the RP either undertook further consequence analyses work or introduced design modifications to negate the need for a HIC claim for the SL, RVI, SG Blowdown lines, FTT and ACC. I was satisfied the RP has avoided HIC claims where reasonably practicable and reduce risks to ALARP for the purposes of the GDA. The final position on the RP's HIC candidate listing is compiled from the Equipment SI List (Ref. 46) and is detailed in Table 5 below:

Table 5: Final HIC Listing for UK HPR1000

Identified Component	Location	Structural Integrity Classification	Consequence
Reactor Pressure Vessel	BRX	HIC	break/missile
Pressuriser	BRX	HIC	break/missile
Steam Generator (primary and secondary shell & tube sheet)	BRX	HIC	break/beyond design basis/missile
Main Coolant Line	BRX	HIC	LBLOCA break (pipe whip)
Main Steam Line	BRX	HIC	break (pipe whip),
Main Steam Line	BSA/BSB	HIC	break (pipe whip)
Main Steam Valves (MSIV)	BSA/BSB	HIC	break/missile
Reactor Coolant Pump (Casing & Flywheel)	BRX	HIC	break/missile

334. The above listing of HIC components informed the subsequent development of the RP's SI safety case, and in particular, its avoidance of fracture demonstrations. The initial understanding was that the MCL, MSL, MSL valves and RCP would warrant a HIC claim. I took this approach to gain assurance that an adequate safety case could be developed, if necessary, for all the HIC components identified by the RP in the GDA and that a meaningful assessment could be undertaken to mitigate any significant risks e.g. the difficulties of underpinning a HIC claim for weld repairs and non-welded regions in cast components. Thus, the RP proceeded with the development of avoidance of fracture demonstrations for these HIC components. I welcomed the pragmatism shown by the RP. This was important because the design features (welds and non-welds) and material properties associated with these components (MCL, RCP, flywheel, MSL and MSL valves) differ significantly from the features assessed to cover the limiting welds and locations in the major vessels (RPV, PZR and SG).

4.2.2 Strengths

335. The RP has developed an approach to SI classification founded on systematic consideration of the direct and indirect consequences of postulated gross failures. The SI classifications are also informed by the categorisation of functions and classification of SSC.

336. The RP's process for SI classification promotes the consideration of OPEX and this has been applied in the design of the UK HPR1000.

337. The RP's approach to SI classification allows for the identification of those structures and components that require a highest reliability claim (HIC in the RP's SI classification system).

338. The RP has recognised that the level of SI demonstration should be commensurate with the importance of the SSC to maintaining nuclear safety. The SI responded to address my concerns with mapping of the safety class through to the SI and code classes.

339. The use of the Fangchenggang-3 SSC classifications allowed an initial SI classification for the UK HPR1000 to progress and the development of a HIC candidate list. The RP developed an awareness of the importance of adopting a multi-discipline approach to establishing the SI classifications of SSC.

340. The RP developed a systematic approach to establishing whether a HIC claim can be avoided and, if not, to determine whether risks are reduced to ALARP. The RP concluded that the consequence of gross failure weas unacceptable and it was not reasonably practicable to avoid HIC claims for the RPV, PZR, SG, RCP, MCL and MSL. For the other HIC candidates, the RP either undertook further consequence analyses work or introduced design modifications to negate the need for a HIC claim for the SL, RVI, SG Blowdown lines, FTT and ACC (Table 4).

4.2.3 Outcomes

341. The RP responded constructively to the issue of RO-UKHPR1000-0008 and RO-UKHPR1000-0058 (Ref. 49) by developing and implementing adequate processes to establish the SI classifications and provide the SI safety cases for the MCL and MSL respectively.

342. There is a need to demonstrate that the MCL and MSL are adequately protected from the postulated failure of SSC. The broader topic of the risk posed to HICs from hazards has been considered and addressed by the IH inspector through RO-UKHPR1000-0046 (Ref. 49). In brief, the IH inspector was satisfied that the RP had provided suitable and sufficient evidence to demonstrate that the hazards that could impact HICs have been adequately identified and addressed. The GDA position on this activity is dependent on the ONR internal hazards assessment and is discussed in ONR-NR-AR-21-012 (Ref. 59).

343. During my GDA Step 4 assessment of the Structural Integrity classification, I identified several minor shortfalls and AFs for follow-up during site-specific stages.

4.2.4 Conclusion

344. Based on the outcome of my assessment of Structural Integrity Classification, I have concluded that the RP's approach is suitable, and importantly, will allow the RP to identify those structures or components (including locations) that will require a higher Structural Integrity claim. The RP also recognises the link between the UK HPR1000 plant categorisation of safety functions and classification of SSC and Structural Integrity classification.

345. The RP undertook significant work to inform its view on the SI classification of a wide range of HIC candidate SCC. The application of these approaches led to the removal of five candidate components from the initial HIC listing.

346. I was satisfied the RP had avoided HIC claims where reasonably practicable and reduced risks to ALARP for the purposes of the GDA. The final position on the RP's HIC candidate listing is provided in Table 5 and should inform the development of the SI and UK HPR1000 safety cases.

347. I raised five AFs relating to SI classification (AF-UKHPR1000-0187 to AF-UKHPR1000-0191).

348. Overall, I am satisfied that the RP has demonstrated an adequate approach for classification of the UK HPR1000 SSCs important for safety. From the information that I have sampled within GDA, I consider this to be broadly in line with ONR structural integrity expectations (Ref. 2) (Ref. 5).

4.3 Avoidance of Fracture Demonstration

349. ONR’s assessment guidance (Refs. 2 and 5) identifies expectations for structures and components where the RP or duty holder invokes highest reliability claims. In such situations, the consequences of postulated gross failures are either deemed intolerable by the RP or its analysis to demonstrate tolerance is difficult or uncertain. In these instances, and since the RP has determined that it is not reasonably practicable to provide design provisions to prevent unacceptable consequences a case for discounting gross failure from the design basis is considered.

350. Discounting gross failure of a component or structure is an onerous approach to constructing an adequate safety case. Cases following this approach should provide an in-depth explanation of the measures over and above normal practice that support and justify the claim that gross failures can be discounted (ONR SAP EMC. 1-3, Ref. 2). In most cases where failure is discounted from the design basis it means no physical defence in depth can be introduced to eliminate, mitigate or protect against the consequences of failure. Instead, conceptual defence in depth is considered, with multiple robust safety arguments expected within the structural integrity case.

351. To achieve this aim, a key expectation informed by precedent in the United Kingdom (UK), relates to the integration of a defect tolerance assessment (DTA), qualified inspection and conservative material properties. In GDA this is referred to as an “avoidance of fracture demonstration” (AOFD). ONR seeks a balanced AOFD with adequately conservative assumptions. This should include a consideration of all the potential failure mechanisms of the component and the measures that have been taken to guard against them. Whilst design code compliance can provide a certain amount of assurance, there are certain areas which, informed by precedent, ONR expects to be further reinforced and integrated within the structural integrity case. For GDA these include:

- fracture analyses (DTA) to establish the sizes of defects of concern;
- reliable and readily qualified manufacturing inspections; and
- a basis for confidence in the achievement of material properties, especially fracture toughness.

352. Noting the expectation to infer a reliability beyond that which can be claimed by design code compliance, care needs to be taken to achieve appropriate balances between the three principal inputs to the avoidance of fracture demonstration i.e. DTA, inspection qualification, and material properties. For example, excessive conservatism in DTA can result in unrealistic demands for inspection qualification or in material properties, such as fracture toughness. This can be a challenging expectation for RPs and requires the exercise of sound judgements, the development of integrated approaches and adequate arrangements for reconciliation within the structural integrity discipline (Ref. 104).

353. In GDA Step 2, it was noted that the RP is developing an understanding of the UK expectations for the avoidance of fracture demonstration for UK HPR1000. However, ONR was not fully convinced that the RP understands the role and significance of the avoidance of fracture demonstration in underwriting the HIC claim for highest reliability structures and components (ONR SAP EMC.1 to EMC.3, Ref. 2). One area that required further development was the understanding of the integration of the fracture analyses, qualified inspection and material properties that will underwrite such cases. To address this perceived immaturity of the RP’s ability to construct an avoidance of fracture demonstration, I raised RO-UKHPR1000-0006 (Ref. 49).

4.3.1 RO-UKHPR1000-0006 - Avoidance of Fracture Demonstration

354. The aim of RO-UKHPR1000-0006 (Ref. 49) was to:

- Address the gaps identified during ONR's GDA Step 2 Structural Integrity assessment and clearly articulate ONR's regulatory expectations.
- Ensure that the avoidance of fracture demonstration considers the holistic avoidance of fracture demonstration and does not impart unrealistic burdens on the individual factors for HIC (e.g., DTA, material properties and inspection activities).
- Gain confidence that the RP understands the conditions for use of avoidance of fracture demonstration and that the RP has satisfactory processes to strike the required balance on the contributing elements of the demonstration.

355. In response to the aims of RO-UKHPR1000-0006 (Ref. 49) listed above, the RP developed a resolution plan, using the RPV as an example for an AOFD. This included a number of key technical documents relevant for the RPV. I assessed the adequacy of these submissions to address the aims of the RO, which are discussed further below.

4.3.1.1 Scope of RO-UKHPR1000-0006 Assessment

356. The role of the avoidance of fracture demonstration in the safety case is described in the 'Safety Case Methodology for HIC and SIC' (Ref. 26).

357. Within this document, the RP describes how it determines the end of life limiting defect size (ELLDS) via a DTA, which is the largest defect the component or structure can tolerate when exposed to the most onerous transients within the design basis. In parallel, the RP identifies the qualified examination defect size (QEDS), which is the defect the inspection techniques can detect and characterise with high reliability. To show that the component is defect tolerant (ONR SAP EMC.1, Ref. 2), and to demonstrate an adequate margin, the lifetime fatigue crack growth (LFCG) from the QEDS is then compared to the final defect size to the ELLDS to determine the defect size margin (DSM). It is RP's aim to demonstrate a DSM of 2 or above, which is consistent with UK experience. The RP has also developed partial NDE technical justifications (TJs) in parallel with its DTAs. The RP's approach is illustrated in Figure 2.

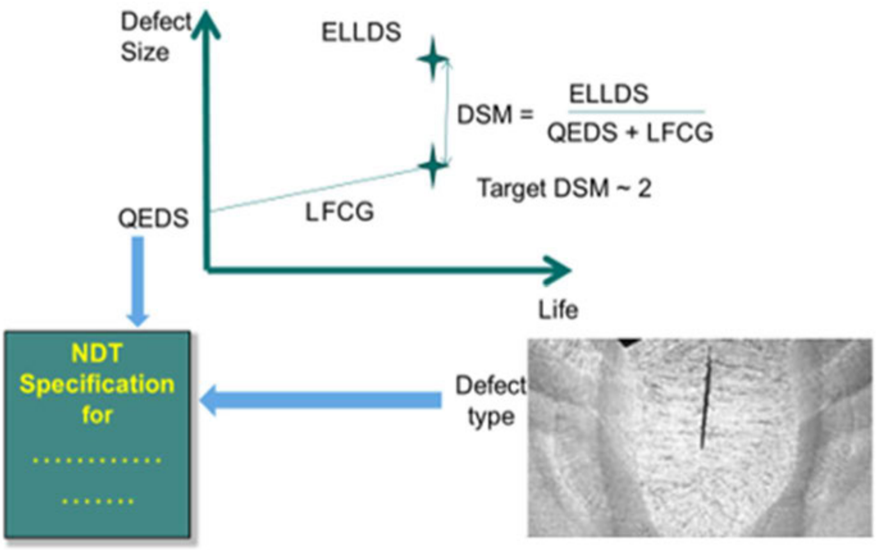

Figure 2: Schematic of Avoidance of Fracture Demonstration

358. To underpin the DTAs, it is ONR's expectation that sufficient evidence is presented to explain how the RP will demonstrate the achievement of conservative material properties, namely fracture toughness. The RP has presented how it will ensure conservative materials have been achieved for HICs within 'Supplementary Fracture Toughness Testing Requirements of Materials for HIC Components' (Ref. 105).

359. The RP provided avoidance of fracture demonstrations within GDA, for a limited set of cases to be selected using a 'weld ranking' process, which is presented in 'Weld Ranking Procedure' (Ref. 106).

360. Results from the application of the Weld Ranking Procedure for the UK HPR1000 HICs are presented across a number of references, based on the chronological development of the assessments during GDA. Details of the results for the RPV, SG, PZR and MCL are provided in the RP's 'Application of Weld Ranking Procedure' (Ref. 107). The results for the MSL (including MSIV) are demonstrated in the RP's 'Application of Weld Ranking Procedure for MSL' (Ref. 108), and the result for the RCP is demonstrated in 'Application of Weld Ranking Procedure for RCP' (Ref. 109).

361. For each of the bounding component locations identified above, three key submissions were provided by the RP:

- Defect Tolerance Assessment.
- Technical Justification.
- Avoidance of Fracture Reconciliation.

362. These were produced following the application of the RP's respective methodologies:

- Defect Tolerance Assessment Methodology for HIC Components (Ref. 110).
- Generic Design Assessment for UK HPR1000: Inspection Qualification for High Integrity Component: (Ref. 111).
- Avoidance of Fracture Reconciliation Strategy (Ref. 112).

363. The standards and criteria adopted within the assessment of RO-UKLHPR1000-0006 (Ref. 49) were principally the Safety Assessment Principles (SAPs) (Ref. 2), internal TAGs (Ref. 5), and relevant standards and RGP informed by existing practices adopted on GB nuclear licensed sites and in previous GDAs (Ref. 1) (Ref. 104). In particular the R6 Defect assessment procedure (Ref. 12) and the ENIQ methodology (Ref. 16).

4.3.2 Assessment

4.3.2.1 Weld Ranking Approach and Results

364. During Step 2 of GDA, the RP described how it would perform detailed avoidance of fracture demonstrations within GDA on a limited set of cases to be selected using a 'weld ranking' process (Ref. 106). The outcomes of the AOFD would be presented in a suite of reconciliation reports for each of the locations identified in HIC components for the GDA

365. I concluded at this stage of my assessment that this process provided a sound basis for selecting areas for further consideration. I was satisfied that the main application of the weld ranking methodology identified suitable and sufficient limiting cases to allow the RP to make an adequate avoidance of fracture demonstration within GDA for the UK HPR1000 HICs.

366. One exception was locations within non-welded regions, which was identified as an area of risk within previous GDAs. I therefore consider the development and application of this selection process within my assessment, by sampling the weld

ranking list and supporting evidence, including methodology documents and component-specific assessments.

367. In total, the RP identified 14 bounding locations for six HICs (Table 6):

Table 6: Identified Bounding Locations for AOFD for UK HPR1000 HICs

Component	Weld/AOFD
RPV	Flange-nozzle shell to core shell weld
	Inlet Nozzle to Flange-Nozzle Shell Weld
	Inlet Nozzle to Safe End Weld
	Core Shell
SG	Inlet Primary Nozzle Safe End to Primary Nozzle Buttering Weld
	Tube-Sheet to Primary Head Weld
	Main Feeder Water Nozzle to Steam Drum Can No.2 Weld
RCP	Casing Inlet Nozzle to Safe End Weld
	Flywheel
PZR	Man-Way Flange to Upper Cylindrical Shell Weld
	Upper Shell to Middle Shell Weld
MSL	MSIV Casing Crotch Repair Weld
	Connection to the Penetration Flange Weld
MCL	Hot leg to SG inlet nozzle safe end Weld

368. Initially, the RP intended to provide, for each of the limiting HIC welds, a detailed defect tolerance assessment and TJ. However, the RP introduced a further stage in the ranking process to reduce the number of TJ submissions for GDA. This additional process groups similar limiting welds according to specific features such as geometry, weld design, and Non-Destructive Examination (NDE) and applies only to the NDE aspects of the limiting HIC welds. I was broadly satisfied with this approach, which is discussed further in Sub-section 4.3.5 below.

369. The DTA reports provide details of the defect geometries assessed. For GDA the RP has focused on the tolerance to radial-circumferential defects. For the purposes of progressing RO-UKHPR1000-0006 (Ref. 49), which is to provide examples of the inputs and application of the reconciliation process, this was satisfactory. Nonetheless, an investigation of the RP's approach to demonstrating defect tolerance for a wider range of defects, e.g. radial-axial orientated, albeit these are less likely to arise from manufacture, formed part of my assessment.

370. One outcome of the RP's weld ranking I questioned was related to the selection of the limiting HIC weld for the SG tube sheet. Notably, I observed that in other GDAs, based on qualitative judgements relating to the likely defect tolerance and the inspectability aspects, highest reliability claims for shell welds on the secondary side of the SG may be limiting rather than on the primary side (Ref. 113). I therefore raised RQ-UKHPR1000-0892, for the RP to clarify an explain the basis of its judgements (Ref. 28).

371. The RP explained that it had followed its weld ranking process which included both its DTA and inspectability rankings. The SG tubesheet 'upper' (secondary shell) weld and the 'lower' (primary head) weld had the same DTA ranking. The deciding factor was the inspectability for which the 'lower' weld (primary head) was the limiting weld for the GDA.

372. In my opinion this was a close judgement, and to mitigate the potential risks prior to the site-specific stages, I asked the RP to consider whether given the reduced section

thickness the secondary side SG tube sheet weld warranted further consideration in the GDA. The RP recognised the potential risk and committed to consider the 'upper' secondary weld in GDA, as detailed in the response to RQ-UKHPR1000-0892 (Ref. 28).

373. The RP subsequently calculated the ELLDS values for both the primary and secondary SG tube sheet welds. The results for a 6:1 reference defect indicated that the SG tube sheet to primary head weld had an ELLDs of 54.2 mm, whereas the secondary side shell weld had an ELLDS of 60.2 mm (Ref. 114). My sampling of the DTAs did not include detailed assessment of the SG tube sheet welds, but noting these results, and the potential for relatively large DSMs, subject to confidence in the QEDS, I was satisfied that the RP has provided sufficient evidence to mitigate the risks for the purposes of the GDA.

374. I also raised RQ-UKHPR1000-0918 (Ref. 28) to gain clarification on several other points relating to application of the weld ranking procedure to the PZR, RPV, MSL and MSL. I also questioned the RP's proposals for its AOFD for the RPV and RCP. The RP clarified the HIC boundaries and explained their reasoning to satisfactorily address my queries relating to the selection of the limiting HIC welds for the PZR, RPV, MCL and MSL. The RP also explained using its weld ranking approach, the rationale for not providing a AOFD for the RCP casing in GDA. The RP also indicated that for the GDA the NDE claims for the non-weld regions in the RCP flywheel and RPV core shell would, as adopted in previous GDAs, be underpinned by a capability statement. I was satisfied with the RP responses to RQ-UKHPR1000-0918 (Ref. 28).

375. In RQ-UKHPR1000-1197 (Ref. 28), I also raised several points of clarification relating to the RP's application of its weld ranking procedure. These points were adequately addressed by the RP (Ref. 28).

376. Overall, I concluded that the RP's weld ranking process provided a reasonable and pragmatic means of selecting the HIC welds for detailed assessment during GDA.

4.3.3 Material properties

377. UK expectations for components of the highest reliability include sound material choices with validated and conservative material property values to infer that the component can deliver its safety functions throughout its entire volume. ONR does not specify or endorse any methodologies but has the expectation that the analyses are soundly based on recognised methods and input data are backed up by data from fully representative samples (as per the ONR SAPs, Paragraph 220) when the component or structure performs a principal role in ensuring nuclear safety.

378. According to the RP's 'Defect Tolerance Assessment Methodology for HIC Components' (Ref. 110), the material properties for the UK HPR1000 DTAs are taken as lower bound values from the appendices ZI & ZG of RCC-M or RSE-M Appendix 5. These material properties cover:

- Toughness K_{mat} or J-R curve.
- True strain-stress curve (tensile test).
- Crack growth rate (and Paris law coefficient).
- Thermal property, including λ, C, ρ, linear expansion coefficient.

379. The RP also indicated that the initiation fracture toughness used in its DTAs may be supplemented with fracture toughness values with ductile tearing invoked. I confirmed that the RP's intended use fracture toughness based on limited amounts of ductile tearing was restricted to extreme fault or hazard loading conditions, as per RQ-UKHPR1000-0452 (Ref. 28).

380. The RP's proposed use of ductile tearing in its DTA work accords with my expectations under SAP EMC. 34 (Ref. 2). However, a condition of its application is that there must be valid materials property data up to the limited extent of the tearing invoked. From my assessment of the RP's assumed tearing values, I conclude there is a basis of confidence in their achievement, but these will need confirmation during the site-specific stages as part of the implementation of the fracture toughness testing strategy. The main discussion of the materials properties is given in Sub-section 4.5. For the purposes of this section, I am content that the fracture toughness testing programme is capable of supporting any future analyses invoking stable tearing. This will need to be demonstrated during each site-specific phase, once the results of the materials-specific fracture toughness testing programme are known.

381. To underpin materials toughness values used in DTA, the RP set out additional fracture toughness testing requirements, which will be used in the DTA as a key part of the HIC safety case. These Supplementary Fracture Toughness Tests (SFTT) will be performed on base metal and representative welds of HICs during site-specific stages. The scope and strategy of the SFFT programme is described in the RP's document 'Supplementary Toughness Test Requirements of Materials for HIC Components' (Ref. 105). This document defines the strategy for supplementary SFTT for HIC components, including how targeted toughness data supports the DTA and the proposal of testing to obtain initial toughness directly from fully representative material for forging, casting and weld of HIC components. This supports the sub-claim "avoidance of fracture demonstration" of HICs in PCSR Chapter 17 (Structure Integrity) and to guide engineering practice of supplementary FTT during license stage.

4.3.3.1 Material Properties Assessment Strategy

382. Given the importance of using robust materials properties to underpin the DTA, I chose to sample The RP's 'Supplementary Toughness Test Requirements of Materials for HIC Components' (Ref. 105), to ensure that ONR expectations for the use of lower-bound materials properties determined from fully representative samples has been considered. To support my review of this document, I raised a number of technical queries and clarifications in RQ-UKHPR1000-1186 (Ref. 28).

4.3.3.2 Assessment of Material Properties and Testing Strategy

383. From my initial review of the RP's Defect Tolerance Assessment Methodology for HIC Components (Ref. 110), I noted that reference to the use of RCC-M and RSE-M for material properties data excluded any mention of the source data for those components where RCC-M is not the primary design code (such as the SG and RCP).

384. On further interrogation of the RP's safety case, I note that the latest version of the SFFM (Ref. 115) states that "lower bound materials toughness properties are required as input for the DTA of HIC. Material properties taken to DTA are referred to the RCC-M Appendix ZI & ZG, RSE-M Appendix 5.6 or ASME Section III Appendix G at present". I judge that this is appropriate within the scope of GDA, however I consider the statement provided in the SFTT should be reflected in the RP's DTA Methodology document (Ref. 110). I consider this to be a minor shortfall, which should be corrected to ensure a clear and consistent SI safety case for the UK HPR1000.

385. The RP's response to RQ-UKHPR1000-1186 (Ref. 28) provided assurances that the DTA calculations will be checked during the site-specific stages to ensure that no HIC materials, including weld materials, will have properties inferior to those used in the GDA calculations. If as-produced materials do indeed have properties inferior to those assumed within GDA, it will be incumbent upon the licensee to provide assurance that the components are either capable of adequately fulfilling their nuclear safety duty or ensure that the component does not enter that duty.

386. During the GDA phase, I have gained sufficient confidence that the materials proposed are capable of satisfying the safety case. This notwithstanding, to satisfy ONR's expectations (Ref. 2) regarding ONR SAPs EMC.3 (Evidence), EMC.13 (Materials) and EMC.20 (Records), the licensee should ensure that all relevant materials properties data, required to underpin the plant safety case, are available in an accessible manner; preferably a single reference document. This should be maintained throughout the lifetime of the plant and updated as appropriate, so as to support the defect tolerance assessments in the extant SI safety case.

387. The RP's response to RQ-UKHPR1000-1186 (Ref. 28) proposed that fracture toughness testing will be undertaken for all HIC components. This meets ONR's general expectations, but I sought further explanation on the extent of these tests through RQ-UKHPR1000-1415 (Ref. 28). This provided confirmation that the materials used in the fracture toughness testing program would be fully representative of the materials used in the forgings, filler metals, and where appropriate, Heat Affected Zone (HAZ) materials as well. The supplementary fracture toughness requirements documentation was subsequently updated to reflect this (Ref. 116).

388. Furthermore, in RQ-UKHPR1000-1415 (Ref. 28) the RP committed, during the build of highest reliability components, to using only welding parameters that have been tested and are backed up with direct fracture toughness testing data. This meets my expectations in this area and builds upon ONR's general experience in manufacture of highest reliability components. It is ONR's expectation that the welding performed on the plant will meet expectations, including any repair welds performed.

389. The RP stated in 'Supplementary Toughness Test Requirements of Materials for HIC Components', (Ref. 105) that there was no intention to test the fracture toughness of heat affected zones (HAZs) in welded components, as they considered that these would always be superior to the surrounding base materials and/or filler materials. I judged that this was not an acceptable position for a highest reliability component. In my opinion, where failure of a component could lead to a large release of radiation to the environment, direct testing should be performed where possible, unless a robust and scientifically based reason can be produced why there would be no safety benefit to so doing. Through the RP's response to RQ-UKHPR1000-1415 (Ref. 28), and discussions with the RP, I have gained assurances that the HAZs will be tested directly. This meets my expectations for weld materials testing.

4.3.4 Materials Properties Assessment Strengths

390. The RP has developed a material testing strategy that is commensurate with ONR expectations for data used to support a robust avoidance of fracture demonstration. The RP's approach encompasses 'above code' demands on fracture toughness testing, built on conservative, lower bound values specified in relevant nuclear design and constructions codes. During manufacture, these material properties will be proven through the application of a robust sampling and test programme, based on samples that are fully representative of the materials being produced.

4.3.5 Materials Properties Assessment Outcomes

391. In my opinion, the RP should ensure that the supporting safety case is reviewed to ensure that there is clear traceability to the evidence showing how the DTAs use conservative materials properties, underpinned by fully representative sampling during the manufacture of HICs.

392. I am aware that the toughness values used in the DTAs have been based on lower bound properties from the appropriate design codes, using allowances for irradiation embrittlement over time, where appropriate. I consider that this is suitable and sufficient for analyses performed during the GDA phase. This is based upon there

being a suitable and sufficient fracture toughness strategy for HIC components, and an associated materials irradiation surveillance programme. The materials surveillance programme is discussed in Subsection 4.5.1.7 of this report and has been judged to be acceptable. This should be maintained and developed through any licencing phase, to demonstrate that there is a clear and unambiguous link between fracture toughness measurements for the plant that meets the requirements of the safety case. This is of particular importance for, but is not necessarily restricted to, components of the Highest Reliability (HICs).

4.3.6 Materials Properties Assessment Conclusions

393. In conclusion, from the information I have sampled and engagements held with the RP, I am content that the materials toughness testing strategy for the UK HPR1000 will support the DTAs performed during GDA, and is adequate to guide ongoing work during the site-specific stages.

4.3.7 Defect Tolerance Assessment

394. The RP stated that it is using the R6 methodology (Ref. 12) for assessing the acceptability of a defect within a structure. ONR has experience of the R6 approach through assessments performed for the current Advanced Gas-cooled Reactors (AGR), Sizewell B and GDA (AP1000® and UK ABWR). I consider the choice of this procedure for the fracture mechanics assessment to be appropriate. R6 is an established and validated procedure for assessing the integrity of structures containing defects, or postulated defects, and is routinely used by Licensees in Great Britain to support nuclear safety cases.

395. The R6 procedure is based on a Failure Assessment Diagram (FAD), which illustrates proximity to failure, and indicates the predicted failure mode (Figure 3). The vertical axis of the FAD (Kr) represents the ratio of applied stress intensity factor to the fracture toughness of the material. This provides a measure of the proximity to failure by plastic collapse. The horizontal axis (Lr) represents the ratio of the applied load to the load required to cause plastic collapse of the section containing the postulated defect. This provides a measure of the proximity to failure by plastic collapse. The interaction between the two failure modes is represented by the failure assessment curve, which is established from the tensile properties. The proximity to failure for given defect size is represented by a locus of assessment points through to a limiting defect size.

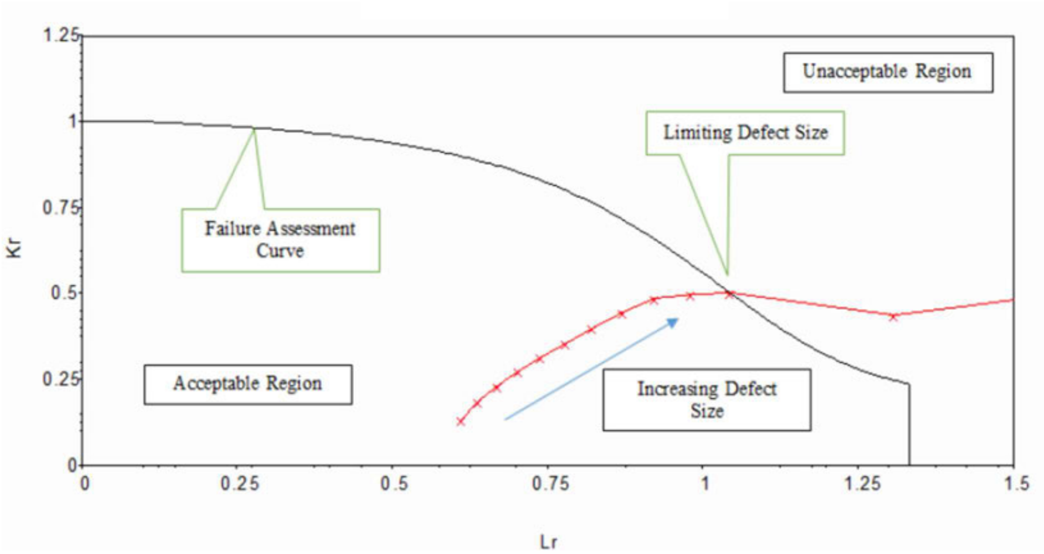

Figure 3: Schematic of R6 Failure Assessment Diagram

4.3.7.1 DTA Assessment Strategy

396. From my review of previous iterations of the RP's DTA methodology during Step 2 and 3 of GDA, I judged that the RP's approach to its DTA work appeared reasonable at a high-level (Ref. 6) (Ref. 7). However, in practice the adequacy of the RP's approach could only be determined through more detailed reviews and by undertaking comparative R6 calculations. In Step 4 of GDA, and with TSC support, I therefore undertook further reviews of the RP's DTA methodology and its implementation.

397. Firstly, I undertook broad brush reviews of several HIC welds in a number of HIC components. The aim of these initial reviews was to sample the DTAs, and to gain clarification of the RP's assumptions, approximations, and methods, in particular, their validity, scope, and compliance with the R6 procedure. The objectives of this initial review work were to gain an overall impression of the RP's DTA methodology and inform judgements on the conclusions being drawn. The initial review work subsequently informed the more detailed reviews and the comparative calculations.

398. In addition, as the R6 procedure requires several user-defined inputs, namely:

- loadings and stresses;
- defect location and characterisation; and
- fatigue crack growth laws.

399. I sampled the RP's derivation and application of several of these inputs for a range of HICs as part of my GDA Step 4 assessment.

400. The reconciliation between the inputs is key to the avoidance of fracture demonstration and so I also undertook assessment work to progress RO-UKHPR1000-0006 (Ref. 49) and my review of the RP's reconciliation process and its application. This was underpinned by my DTA work but also my sampling of the material properties (Sub-section 4.3.3) and TJs (Sub-section 4.3.5).

4.3.7.2 Assessment of DTA Methodology

401. The RP has documented its methodology for completing DTAs in 'Defect Tolerance Assessment Methodology for HIC' (Ref. 110).

402. During Step 3 of GDA the RP provided the first DTAs (Ref. 117) (Ref. 107) (Ref. 118), which I assessed at a high level to gauge the competency of the RP. In general, I was content with the approach taken. With the support of a TSC, I also undertook a more detailed review of the RP's DTA methodology in Step 4 of GDA. I raised RQ-UKHPR1000-1551, to gain clarification on several points to which the RP provided adequate responses (Ref. 28). Overall, my review of the RP's 'Defect Tolerance Assessment Methodology for HIC Components' (Ref. 110) concluded that the RP's DTA Methodology document could in principle provide an acceptable route to conservatively perform DTAs to the R6 procedure, but the guidance provided to the user was at a very high level and open to interpretation. For example, whilst the process of determining reference stresses and stress intensity factors is mentioned, it does not provide any detail on the different solutions available or guidance regarding consideration of out-of-plane stresses when considering plastic collapse. Had more information been provided, it would have acted as a more useful guide to the individuals performing the DTAs and could also have avoided some of the generic issues identified in my later review work.

403. In Step 3 of GDA, I also noted that the RP appeared to be taking a complex approach in its initial assessments. Notably, the RP had calculated assessment points at each interval in the design transient. This approach appears complex and not necessarily conservative. In particular, it was my understanding that there is a degree of

uncertainty in the design transient specification. This could impact aspects such as the pressure and temperature rise times, and timing of peak thermal gradients. To counter these uncertainties, it is my experience that bounding information is commonly extracted (e.g. design pressure, maximum and minimum temperature) and used to generate a single bounding assessment point on the failure assessment curve.

404. It was my opinion that this aspect of the RP's DTA's methodology warranted further consideration. In Step 4 of GDA and with TSC support, I investigated the RP's approach to establishing the ELLDS limiting case in more detail. This was achieved through review and undertaking comparative DTA calculations. I established that to determine the ELLDS across all transient conditions, the RP calculated the ELLDS for each time point in each transient and then took the overall limiting case (a full computation approach). The RP has used this methodology throughout the assessments reviewed in both the broad-brush and deep-dive reviews. The advantage of this approach is that it removes any uncertainty related to picking the limiting time points for assessment based on stress and temperature alone (manual approach). The disadvantage is that it requires more computation; however, this is often not a significant disadvantage as a large number of cases may still need to be considered, even when using the manual approach to ensure that the limiting time points are identified.

405. ONR understands that this approach is becoming more popular and may become accepted practice in due course. Nonetheless, although in principle the approach is more accurate, it is dependent on the veracity of the assumptions that underpin the transient definitions and so has the potential to remove some conservatism that may be included in the manual approach. I consider the overall level of conservatism in the RP's DTAs below.

4.3.7.3 Assessment of Loadings and Stresses

406. The specified loads are a key input in the DTAs. In RQ-UK HPR1000-0145, I sought clarification on the loads to be considered and the combination of those loads (ONR SAP EMC. 7, Ref. 2). Upon review of the RP's response to RQ-UK HPR1000-0145 (Ref. 28) and the systems and loadings for defect tolerance assessment document (Ref. 40) some follow-on questions were raised in RQ-UKHPR1000-0162 (Ref. 28). In the response to RQ-UKHPR1000-0162 (Ref. 28), the RP stated it considered pipe break loads which included pressure fluctuation, moments, forces, and pipe whip loads. The different sources of the loads stated by the RP appeared consistent with my expectations. I also note that for several of the RPV locations selected under RO-UKHPR1000-0006 (Ref. 49), the limiting loads as expected relate to overcooling transients. However, the traceability of the loads induced as a result of a design basis loss of coolant accident (LOCA) were not immediately visible. I consider the traceability of design loads and evidence in more detail in Sub-section 4.1.2.1 above, from which I have raised an assessment finding (AF-UKHPR1000-0186) that I consider captures this matter sufficiently.

407. I also noted that no assessment temperatures were listed for emergency or fault transients. It is important that the appropriate assessment temperatures are specified to ensure that appropriate material properties are used. The RP stated that the assessment temperatures for emergency and fault conditions were derived from the temperature transients and were available from the Design Transient Specification (Ref. 119). It was not clear that using the transient temperatures to determine material property and loads was demonstrably conservative. I later confirmed through my comparative DTA calculations for the RPV Flange Nozzle to Core Shell weld and RQ-UKHPR1000-1549 that the RP assesses both the deepest and surface tips of the cracks and the yield stress was corrected for the through wall temperature gradient accordingly (Ref. 28).

408. In RQ-UK HPR1000-0280, I sought further details of the process for identifying and combining transients and for ensuring the resulting loads are conservative in the design specification. The RP indicated that the loading assumptions included several conservatisms: overestimates of transient numbers, maximum heat-up and cooldown rates, the identification of bounding conditions with worst case assumptions under emergency and accident conditions (Ref. 28). On balance I was satisfied there was a basis for confidence that there was conservatism in the transient magnitudes and frequencies (ONR SAP EMC. 28, Ref.2). Nevertheless, the traceability of the loads is a generic topic for consideration in my GDA Step 4 assessment report.

409. With support of a TSC, I undertook a further review of the 'System and Component Loadings for Defect Tolerance Assessment' document (Ref. 120) and raised RQ-UK HPR1000-1203 and RQ-UK HPR1000-1550 (Ref. 28). The RP provided useful responses and, in particular, confirmed that expansion loads and thermal stresses under fault conditions were included in the DTA loadings. Additionally, my review, found that the RP did identify the relevant loadings requiring consideration, but limited guidance was provided to underpin them. For instance, whilst it states that WRS should be considered, it does not provide any detail regarding the stress distribution or derivation of the magnitude to be applied. More information would have acted as a more useful guide to the individuals performing the DTAs and could also have avoided some of the generic points identified later in my review.

410. The RP also indicated that simple estimates of the residual stress profiles (i.e. the most conservative) had been used in the DTAs. I was aware that the R6 Development Panel (Ref. 121) had altered its advice on the Level 1 residual stress for ferritic steels following post weld heat treatment. The RP initially stated that it was not its plan to change the value of residual stress within the current DTA as no formal guidance has been produced. It is my opinion this proposal was inconsistent with the aims of GDA (i.e. to mitigate risks) and was not commensurate with that expected of an informed customer.

411. I pursued the impact of the residual stress assumptions within Step 4 of GDA. In terms of the residual stress inputs used in the RPV locations selected for RO-UKHPR1000-0006 (Ref. 49), I also confirmed that the RP had used residual stress values based on lower bound yield stress values. The use of lower bound yield stress values was appropriate for judging the proximity to collapse in R6 (i.e. the Lr parameter). However, the guidance in R6 is to use residual stress based on mean yield stress values, as this is conservative for evaluation of the fracture parameter, Kr. The use of lower bound yield stress in the evaluation of residual stress was therefore non-conservative.

412. In response to RQ-UKHPR1000-0724, the RP acknowledged its understanding of the change in R6 guidance relating to the residual stress assumptions for ferritic welds subject to Post Weld Heat Treatment (PWHT) (Ref. 28). The RP also committed to update its DTAs as appropriate.

413. For the RPV and closure of RO-UKHPR1000-0006 (Ref. 49) the limiting case was the flange nozzle to core shell weld. However, the RP demonstrated a DSM of at least 2.1 in the updated assessment (Ref. 122). With the support of my TSC, I also undertook a sensitivity study to inform my judgement on the significance of the residual stress assumption. For the RPV flange nozzle to core shell welds this indicated a marginal reduction in the DSM to about 1.85 with a 25% increase in yield stress (Ref. 123). This was because as this weld is subject to PWHT, the Level 1 WRS profile is reduced to 0.3Sy, therefore the difference between using lower bound or mean yield stress is less significant in absolute terms in the calculation of the weld residual stress, as discussed in the RP's response to RQ-UKHPR1000-1200 (Ref. 28). I therefore judge that the small reduction in DSM in this case is indicative of adequate defect tolerance,

especially when the DSM is judged in the context of the additional conservative assumptions identified in the RP's DTA methodology (Ref. 112) (Ref. 110).

414. Nonetheless, I consider the overall position on the level of conservatism in the RP's DTAs in Sub-section 4.3.7.6 below. In addition, in reviewing the veracity of the RP's AOFD, I also take into account any potential non-conservative assumptions in the WRS values for the relevant HIC welds further on in my assessment.

4.3.7.4 Assessment of Fatigue Crack Growth

415. Within Step 2 of GDA, I identified that the RP had included a provision to invoke crack initiation assessment in the fatigue crack growth calculations. As this approach is not currently a feature of the R6 procedure, and appears to relate to provisions within the RSE-M code (Ref. 7), I was concerned that there needed to be adequate justification for combining different codes and standards, ONR SAP ECS.3 Paragraph 173 (Ref. 2) ONR states:

"The combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated".

416. Within RQ-UKHPR1000-0159 (Ref. 28), I asked the RP if it was its intention to use fatigue crack initiation as part of the DTAs for the UK HPR1000. In response, the RP stated that it is not its intent to include fatigue crack initiation in the DTA process. The RP stated that it would consider the use of fatigue crack initiation as part of a sensitivity analysis. I was therefore content that fatigue crack initiation is not intended to underpin the avoidance of fracture demonstration within the UK HPR1000 safety case.

417. To estimate the level of fatigue crack growth that can occur from the assumed QEDS the RP proposed a simple Paris Law which is consistent with the R6 methodology. In addition, the RP also proposed a methodology used within RSE-M to take account of plasticity within the stresses induced by transients. It is not clear that this approach is consistent with the R6 methodology.

418. Noting ONR's position relating to the combining of codes outlined in the ONR SAPs Paragraph 173 (Ref. 2) above, I raised RQ-UKHPR1000-0248 (Ref. 28) and reviewed the work completed by the Independent Expert Working Group (IEWG). This group was tasked with reviewing the 'UK methodology' for the UK EPR™ reactor design that is being constructed at Hinkley Point 'C', though the approach proposed by the RP was similar. The IEWG drew the following conclusion (Ref. 28)*:* "Assessments of FCG according to RSE-M give results identical to following R6 advice if the material remains elastic and the same FCG rates are used. Note, however, that RSE-M suggests more conservative crack growth rates".

419. Based on this review, and in the assumptions that the stresses induced by the transients used for the fatigue calculations should remain elastic, I was content with the RP's approach to fatigue crack growth (ONR SAP EMC. 34, Ref. 2).

420. In response to RQ-UKHPR1000-0452 (Ref. 28), the RP stated that, as part of the refinement of the DTA input and assumptions, there were several potential areas for consideration. Some of the suggestions put forward by the RP were in line with UK expectations, namely, interrogation of the transient combinations and invoking ductile tearing arguments for extreme fault or hazard loading conditions. However, some suggestions were not reasonable avenues to reduce conservatism. Specifically, it is my opinion that the use of mean fatigue crack growth laws and allowing for the strength of the cladding were not consistent with ONR's expectations (ONR SAP EMC. 33, Ref. 2).

421. I subsequently sampled the DTA reports referenced under RO-UKHPR1000-0006 (Ref. 49) and confirmed that conservative upper bound fatigue crack growth laws had been used (Ref. 122) (Ref. 124) (Ref. 125). This aligned with the RP's preference in its DTA methodology document (Ref. 110). In addition, I confirmed that the RP had not taken account of the strength of the cladding in the primary stress calculations when applying the R6 defect assessment procedure.

422. Late in Step 4 of the GDA, my TSC made me aware that the RP had appeared to have invoked intra-transient rather than inter-transient load combinations in some of the LFCG calculations (Ref. 126). Inter-transient pairing is conservative, because it reflects the worst possible (and usually unrealistic) sequence of fatigue cycles that could happen, these are unlikely to occur in reality. In contrast, intra-transient pairing, whilst not capturing the most favourable sequence of fatigue cycles, attempts to represent the plant operation and by doing so is not as conservative as inter-transient method in calculating the LFCG.

423. This was illustrated by the RP through a sensitivity study undertaken in response to RQ-UKHPR1000-1711, which showed a significant difference with the QEDS + FCG at end of life reduced from 16.7mm using inter-transient pairing to 10.8mm using intra-transient pairing, (Ref. 28). Thus, inter-transient pairing is preferred unless there is some knowledge (for example plant monitoring) about the order in which the transients will be applied.

424. The usual approach is to use an inter-transient (between transient) pairing method, which is conservative and gives high LFCG. However, two of the RP's DTAs adopted an intra-transient (within transient) pairing method with the transient accounting more closely representing the plant operation. The two exceptions are the 'Main Feedwater Nozzle to Steam Generator Drum Can No.2 Weld' report (Ref. 127), and the 'Steam Generator Tubesheet to Primary Head Weld' (Ref. 128) summary report. I consider the significance the effects on the LFCG as part of the RP's reconciliation (Sub-section 4.3.6).

425. The use of inter-transient pairing is acknowledged as a significant conservatism in the assessment. This notwithstanding, the use of two methods for LFCG may lead to inconsistencies in the derivation of the QEDs. In my opinion there needs to be consistency in the LFCG method that underpins the DSM values. I therefore raise the following assessment finding to ensure that this matter is tracked during the site-specific stages.

AF-UKHPR1000-0192 – The licensee shall, as part of detailed design, justify that consistent approaches to lifetime fatigue crack growth are employed in its defect tolerance assessment methodologies, such that judgements can be made on the level of defect tolerance and the development of the avoidance of fracture demonstration cases.

4.3.7.5 DTA Generic Points

426. As noted above the initial review work provided the means to sample the DTAs, and to gain clarification of the RP's assumptions, approximations, and methods, in particular, their validity, scope, and compliance with the R6 procedure. A further aim was to identify generic points from the application of the RP's DTA methodology. My sampling of the RP's DTAs was purposely wide ranging and covered different types of HIC welds in a range of components (major vessels and piping), as provided in Table 7 below:

Table 7 – List of RQs raised to review component DTAs.

Sampled Component DTA	RQ Raised - (Ref. 28)
Reactor Pressure Vessel	RQ-UKHPR1000-1194 RQ-UKHPR1000-1195 RQ-UKHPR1000-1199 RQ-UKHPR1000-1200 RQ-UKHPR1000-1202 RQ-UKHPR1000-1307 RQ-UKHPR1000-1469 RQ-UKHPR1000-1549
Pressuriser	RQ-UKHPR1000-1198 RQ-UKHPR1000-1304
Steam Generator	RQ-UKHPR1000-1604 RQ-UKHPR1000-1605 RQ-UKHPR1000-1606 RQ-UKHPR1000-1738 RQ-UKHPR1000-1739
Main Coolant Lines	RQ-UKHPR1000-1305 RQ-UKHPR1000-1306 RQ-UKHPR1000-1552 RQ-UKHPR1000-1711
Reactor Cooling Pump	RQ-UKHPR1000-1463
Main Steam Line	RQ-UKHPR1000-1682
Main Steam Isolation Valves	RQ-UKHPR1000-1669

427. Following the completion of the initial review work, RQ-UKHPR1000-1546 (Ref. 28) was issued to capture the generic points and gain clarification of the RP's position. A query was raised relating to the RP's consideration of radial-axial defects, which had only generally been considered as a sensitivity study, with little discussion of the results or the implications when the safety margins were apparently poor. Responses from the RP provided more consideration of radial-axial defects and the implications, including further refinement at some of the assessment locations. The additional information provided indicated that, when considering the relevant defect aspect ratios (acknowledging that for axial defects only smaller defect length to depth ratios are more likely), it is possible to demonstrate that acceptable DSMs (>2) can be determined for all components. Whilst updated reports covering radial-axial defects in more detail have not been reviewed, the RP has shown that they are providing the consideration of radial-axial defects required, but ultimately it will remain important to demonstrate that the assessment results provide adequate defect tolerance margins for a wider range of defects. I therefore raise an assessment finding to ensure that this is tracked to completion during the site-specific stages:

> AF-UKHPR1000-0193 – The licensee shall, as part of detailed design, undertake defect tolerance assessment covering all relevant weld locations and defect orientation for high integrity components, including the vulnerable areas of the parent forging. This should expand the scope of assessments undertaken during GDA and should demonstrate that the limiting locations have been assessed.

428. In the responses to RQ-UKHPR1000-1546 the RP also acknowledged that a 55MPa WRS had been used in a small number of assessments (in particular the RPV core shell to transition ring weld rather than the higher recommended value of 0.3 times mean room temperature yield stress). The RP also committed to subsequently considering the impact of the WRS assumptions within its DTA work, as per the RP's response to RQ-UKHPR1000-1546 (Ref. 28). I consider the implication of these assumptions further under reconciliation (Sub-section 4.3.6). It will suffice to say that

the use of the most up to date guidance in R6 is fundamental to its application. I also note the RP's initial reluctance to progress this point, though recognise that this may in part be related to the imposition of a tight GDA timescale. Nevertheless, it is important that the most up to date guidance underpinning the R6 procedure, including that pertaining to weld residual stress values, is incorporated into the DTA.

429. An associated generic point was related to the WRS value used in the baseline assessments. I found that, in several cases, baseline assessments used WRS based on lower bound (LB) materials properties rather than mean material properties as prescribed within the R6 procedure, as discussed in 'Review of Defect Tolerance Assessments and Comparative Calculations' (Ref. 19). In several cases, the RP was able to demonstrate that increases in WRS to account for mean properties would not adversely impact the assessments performed (Ref. 19). The RP also acknowledged that the application of LB material properties when determining WRS would be rectified within more detailed assessments during the site-specific stages. I address the implications of these assumptions under reconciliation (Sub-section 4.3.6). I also recognise the commitment made by the RP but consider it important that appropriate mean yield stress based WRS values are included in the DTAs. Irrespective of my conclusions relating to reconciliation, in my opinion, the RP should have demonstrated that appropriate yield stress values are incorporated into its calculations of the proximity to plastic collapse and fracture in accordance with the extant R6 defect assessment procedure.

430. One area that was highlighted for review was the methodology applied for dissimilar metal welds. The results of my comparative work included a review of the RPV Inlet Nozzle to Safe End Weld, which was selected for review primarily to assess if the correct methodology was being applied for dissimilar metal welds. Following this work, no significant points relating specifically to the assessment of dissimilar metal welds were unresolved.

431. Late in Step 4 of the GDA, I was made aware of an unresolved query relating to the consideration of out of plane collapse in Sub-section 2.4.2 of the assessment of the PZR Man-Way Flange Upper Cylindrical Shell Weld (Ref. 19). Sections I.8 and II.4.2 of R6 state that assessments need to consider collapse of the structure as a whole, and that caution must be taken where collapse may be governed by the stresses in the plane of the defect rather than the stress normal to the defect. By using an Lr solution that does not consider out of plane stresses and therefore out of plane collapse, the RP had not considered all of the potential collapse modes, hence the collapse load is potentially underestimated and the limiting defect size and defect safety margin potentially overestimated.

432. I raised this point with the RP, in RQ-UKHPR1000-1304 and RQ-UKHPR1000-1546 (Ref. 28) and advised the RP that the solution in Section IV.1.8.1 of R6 does take account of in-plane and out of plane collapse for circumferential defects in a cylinder. Another option would be to use the solution in Section IV.1.9 of R6 to calculate the out of plane Lr. It is likely that the out of plane Lr will bound the in plane Lr in some cases. I asked the RP to consider the effect of out of plane Lr in its DTAs and in its DTA methodology document, as there was the potential for non-conservative ELLDS because Lr may be under-estimated.

433. On further review, it was my opinion that this point may be equally applicable to several of the DTAs performed, hence this was identified as a generic point. I regard this as a shortcoming, which along with the other generic points, highlights the need for further improvements in the RP's DTA guidance, in particular, the DTA Methodology document, to ensure more consistent approaches that accord with accepted practice are adopted in the UK HPR1000 R6 assessments.

434. I consider the effect on reconciliation in Sub-section 4.3.11, but irrespective of my view on the veracity of the RP's AOFD, I consider that the RP's DTA methodologies could be improved from further guidance, to ensure consistency and a conservative result.

435. In my opinion, it is important that this matter is tracked to conclusion, I therefore raise the following assessment finding to address this generic point.

AF-UKHPR1000-0194 – The licensee shall, as part of detailed design, demonstrate that consistent approaches are implemented in the defect tolerance assessment, in accordance with the relevant defect tolerance assessment procedure. This should include, but not be limited to, consistent use of weld residual stress values, appropriate yield stress in failure parameters and identification of the limiting conditions for plastic collapse.

4.3.7.6 Assessment of DTA Conservatism

436. To inform my view on whether there is an adequate level of conservatism in the DTAs, the assessment of a sample of the inputs, and the generic points relating to the loads, in particular, residual stress values, identified above, need to be balanced against the known conservatisms in the DTA methodology. The RP indicated in its Reconciliation Strategy documents (Ref. 112) (Ref. 110) that the aim of the refinement process for the DTAs is not to remove all conservatism, instead the aim is to maintain a robust avoidance of fracture demonstration (Ref. 112) (Ref. 110). I sampled the DTA's for radial-circumferential defects in the RPV because I needed to make a judgement as to whether to progress comparative calculations for the welds referenced in RO-UKHPR1000-0006 (Ref. 49). In combination with the DTA Methodology document my review highlighted several conservatisms:

- Option 1 FAD used in all DTA work – the most conservative R6 option.
- Level 1 residual stress levels and profiles.
- Upper bound FCG laws and transient combinations that are a conservative representation of expected plant operation.
- Conservative transient numbers and magnitudes, maximum warm-up and cooldown rates, large heat transfer coefficients.
- Conservative fixed aspect ratio when the final predicted aspect ratio is smaller.

437. In contrast, the RP has also invoked certain assumptions which inadvertently result in a reduction in the conservatism in DTA:

- Use of a potentially non-conservative crack tip temperature in the determination of material properties, discussed in RQ-UKHPR1000-1200 (Ref. 28) (Ref. 110).
- Use of lower bound yield stress in the determination of WRS in the Kr calculation.

438. As the minimum DSMs for radial-circumferential defects in the RPV for the core shell to transition ring weld and inlet nozzle to safe end weld are 6.1 and 3.65, respectively (i.e. significantly above the target DSM of 2), I consider that an adequate level of defect tolerance is evident, irrespective of the removal of some conservatism.

439. The limiting case for progressing RO-UKHPR1000-0006 (Ref. 49) appeared to be the RPV flange nozzle to core shell weld with a minimum DSM of 2.1. However, balancing the known conservatisms in the DTA with those that have been removed, I was satisfied that overall an adequate level of conservatism had been retained for the purposes of informing a judgement on the RP's reconciliation process, with the initial objective of the comparative work to progress RO-UKHPR1000-0006 (Ref. 129).

4.3.7.7 DTA Comparative Calculations

440. A number of different components and regions were selected for more detailed review and comparative calculations. These were informed by my GDA Step 2 and 3 assessment work and the results of my initial DTA review work in Step 4 of GDA.

441. For the detailed review work, additional data was requested and provided by the RP. This included transient pressure and temperature data, material properties and stresses extracted from the FEA and used within the assessments. This data was used to carry out independent R6 defect tolerance assessments, to understand the RP's assumptions and approaches and to allow comparisons of the results. Details of the comparisons and queries on any observed differences were raised in RQs for each of the deep-dive reviews performed. Note that the transient numbers in the following section refer to those detailed in the 'Design Transient Specification' (Ref. 41).

442. With the support of my TSC, I undertook comparative DTA calculations for two RPV locations referenced in RO-UKHPR1000-0006 (Ref. 49), specifically, the RPV flange nozzle to core shell weld and the RPV inlet nozzle to safe end weld. I also undertook comparative DTA calculations for the MCL hot leg to SG inlet nozzle safe end weld. A summary of these comparative calculations is provided below with further details available in 'Review of Defect Tolerance Assessments and Comparative Calculations' (Ref. 19). The results of this comparative DTA work along with my views on the materials and TJ inputs subsequently informed my view on the veracity of the RP's reconciliation process (Sub-section 4.3.6 of my report).

RPV Flange Nozzle to Core Shell Weld

443. This weld was selected for comparative work because the DTA for this component had recently been re-issued. It was also referenced as a reconciliation example within the resolution plan for Regulatory Observation RO-UKHPR1000-0006 (Ref. 49).

444. The RP submitted its DTA for the RPV flange nozzle to core shell weld (Ref. 122). This work was undertaken by CGN and was subject to the verification process. With TSC support I undertook a high-level review of the DTA and raised RQ-UKHPR1000-1200 (Ref. 28) to progress some points of clarification. These questions were answered satisfactorily, with the generic points relating to the DTA methodology captured above.

445. My TSC then undertook comparative DTA calculations on behalf of ONR, a follow-on RQ, RQ-UKHPR1000-1549 (Ref. 28), was subsequently raised to understand the differences between the RP's and my independent calculations.

446. The 'Review of Defect Tolerance Assessments and Comparative Calculations' (Ref. 19) presents the ELLDS depth for each transient for both the independent calculation, and the time step at which the ELLDS occurs. These differences in ELLDS values are discussed below. The following can be noted from the comparison:

- The same limiting transients for each condition category are identified by the independent calculations:
- The identified bounding transients are considered to be reasonable, i.e. from experience, they are the transients that would be expected to lead to the most onerous loading conditions.
- The majority of the independently calculated ELLDSs are within 2mm of the reported values.
- The majority of the limiting time points in the independent calculations are either the same, or an adjacent time point.

447. The RP provided LFCG calculations for a 10mm by 60mm start of life defect (Ref. 122). The following can be noted from comparison with the independent calculation (Ref. 19):

- The overall LFCG is similar, with an end-of-life depth of 13.2mm for the report and 12.8mm for the independent calculation. The end-of-life aspect ratios are also similar, 4.73 (Ref. 122) and 4.82 for the independent calculation (Ref. 19).
- There is a small difference in the SIF values, as seen from the first two transient pairings. It is not possible to determine whether this is due to differences in the stress profiles (for example, crack face pressure), SIF solution implementation or limiting time point selection due to a lack of this data in (Ref. 122).
- There are some differences in the transient pairings this is most likely due to the small differences in SIF ranges between the calculations.

448. The response to RQ-UKHPR1000-1549 (Ref. 28) provided further information to resolve and explain the differences between the report and the independent calculations. I judged that these differences were not significant in terms of undermining the confidence that the RP is correctly applying the R6 procedure. I conclude there is good agreement on the limiting ELLDS and LFCG calculations. A QEDS of 10 mm for the RPV flange nozzle to core shell weld was therefore supportable for reconciliation from DTA.

RPV Inlet Nozzle to Safe End Weld

449. This weld was selected for comparative work because it is a dissimilar metal weld. In addition, this weld is identified as an example for reconciliation in the RO-UKHPR1000-0006 resolution plan (Ref. 49), and so confidence in the output from the DTA was important to the development of the RP's avoidance of fracture demonstration.

450. The RP submitted its DTA for the RPV inlet nozzle to safe end weld (Ref. 124). This assessment was undertaken by EDF Energy and was subject to EDF Energy verification and RP oversight. With support of a TSC, I undertook comparative DTA calculations. I subsequently raised RQ-UKHPR1000-1469 (Ref. 28) to further my understanding of the differences between the RP's and my independent calculations.

451. Using information provided in the independent calculation (Ref. 19) and the 'Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld' (Ref. 124), I undertook a comparison of the ELLDS and QEDS + FCG results for the list of transients considered by the RP. From this comparison, I noted the following:

- The ELLDS for the Condition A/B/C/Test transients in the independent calculation are all slightly larger than in the 'Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld' (Ref. 124), this is expected given the omission of Mode II and III SIFs from the independent calculations. The limiting transients are the same in all but one case, with the limiting time points either identical or close to the reported values. Given the complexity of the calculations and the amount of data involved, one would not expect a 100% match to the limiting transient and time-points – the crucial comparisons are between the ELLDS's values which show good agreement.
- The ELLDS for the Condition D transients do not show agreement with the 'Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld' (Ref. 124), with the ELLDS values from the independent calculations significantly smaller circa 24% in some cases. This was raised in RQ-UKHPR1000-1469 (Ref. 28) the response to which clarified that the description of the calculation of end force for Kr was incorrect, and the RP provided further clarification of the calculation of the end force used in the assessment. The response reconciled the differences in the calculations and repeat calculations using the revised

inputs gave a good level of confidence in the ELLDS values for the Condition D transients.

- The identified bounding transients are considered to be reasonable i.e. from experience they relate to transients that would be expected to lead to the most onerous loading conditions.
- The QEDS + FCG values in the independent calculation are all smaller than in the 'Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld' (Ref. 124), which is expected given the omission of Mode II and III SIFs. Overall, there is a good level of confidence in the QEDS + FCG values.

452. The response to RQ-UKHPR1000-1469 (Ref. 28) closed out the comments and the comparative assessments performed give confidence that the RP's assessments are correctly applying the R6 procedure. The comparative calculations therefore provide confidence in the RP's approach (Ref. 19). A QEDS value of 5 mm for the RPV inlet nozzle to safe end weld was therefore supportable for reconciliation from DTA.

Main Coolant Line – Hot Leg to SG Inlet Nozzle Safe End Weld

453. This component was selected for comparative calculations because it was a piping weld subject to MCL pipework loading and also because it enabled sampling of the RP's approach to a different DTA challenge.

454. The ELLDS limiting time points from the 'Defect Tolerance Assessment of MCL Hot Leg to SG Inlet Nozzle Safe End Weld' (Ref. 130) and independent calculations (Ref. 19) were compared. The results of the comparison raised the following points:

- Several ELLDS calculations in the comparative assessment do not match (Ref. 130). The difference is most likely to be related to how pipework loads (thermal, pressure, crack face pressure, etc.) are combined, but limited details were available in the 'Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld' (Ref. 124).
- The limiting time points identified in additional information relating to the 'Defect Tolerance Assessment of MCL Hot Leg to SG Inlet Nozzle Safe End Weld' (Ref. 130) appear identical for all paths (Ref. 19). This is not the case in the independent calculations, with many of the limiting time points not matching the comparative assessment. However, it was subsequently confirmed that the difference between the ELLDS for the different paths for different time points was not significant, because there are several time points with similar conditions, and so the difference was not a cause for concern.
- The yield strength in the information provided (Ref. 130) is as low as 125.7 MPa, which is lower than the yield strength at the relevant temperature in RCC-M Table Z.I.2.2. This would reduce the value of Lr and therefore the ELLDS and DSMs, which is conservative, but the source of this low yield strength is unclear.

455. Note that if the smaller ELLDS from the independent calculation was used, the limiting DSM would reduce from 2.1 (Path 1-5, T2-11) to 1.6 (Path 6, T3-6).

456. A portion of the results is presented of the FCG calculation for Path 2 from the report and independent calculations (Ref. 19). A comparison between the results shows:

- The maximum and minimum time points in the independent calculation do not always match those provided in the report.
- The ΔK_{cp} values provided in the report are not consistent with either the K_a max and K_a min values, or the Δa and Δc values.
- The Δa and Δc values are reasonably consistent between the report and the independent calculation. It is expected that the independent calculation would give lower growth as it only considers ΔK_I.

457. I raised RQ-UKHPR1000-1711 (Ref. 28), to gain further clarification from the RP on the points raised above. The most significant responses are summarised below.

458. Regarding the lack of clarity in how the pipework loads have been combined the RP provided further information on the loads and methods in response to RQ-UKHPR1000-1711 (Ref. 28). Although not covered in the DTA Methodology Document, this was consistent with the approach adopted in the assessment of the RPV Inlet Nozzle to Safe End Weld (Ref. 124).

459. The RP also undertook a sensitivity study with the thermal pipework stress considered in the assessment in several of the transients and assuming this was a primary stress. The results indicated a 10%-15% reduction in the ELLDS, giving a new limiting ELLDS of 32.9 mm for transient T3-6, and a DSM of 1.98.

460. Noting the DSM was marginally below 2, the RP also carried out a sensitivity on the FCG calculation, using intra-transient (within transient) pairing rather than the inter-transient (between transient) pairing as reported in the RP's response to RQ-UKHPR1000-1711 (Ref. 28). This reduced the QEDS + LFCG size from 16.5mm to 10.8mm, which would give a DSM of 3.2 rather than 1.98, demonstrating the conservatism in the approach to transient pairing in the FCG calculation.

461. In addition, the RP clarified that the ΔK_{cp} values were not K_{max} - K_{min}, but instead were the maximum intra-transient range for the transient pairing, which were presented to demonstrate the conservatism in the LFCG calculation by using inter-transient pairing.

462. Thus, the comparative assessments performed give confidence that the assessments are correctly applying the R6 procedure. A QEDS value of 8mm for the MCL hot leg to SG inlet nozzle safe end was therefore supportable for reconciliation from the DTA.

463. Other than the generic points relating to the DTAs discussed above, all comments raised were satisfactorily closed following responses from the RP. Nevertheless, most of these points should ideally have been dealt with by the RP at an early stage in the development of its DTA Methodology Document (Ref. 110) to ensure a robust and consistent approach across its R6 assessments.

464. The comparative calculations along with the more detailed review work inform a view on the adequacy, assumptions, approximations, in the RP's methods, and in particular the validity, scope, and compliance with the R6 procedure. The result of this process has been generally positive, with the results showing good agreement between the RP's calculations and the independent ELLDS and LFCG calculations that were obtained for the RPV flange nozzle to core shell weld. The comparative work for the RPV inlet nozzle to safe end weld initially indicated some significant differences between the RP's results and my independent calculations, but these were subsequently resolved through the RP providing further explanation of its assumptions. Similarly, for the MCL hot leg to SG safe end weld, initially large differences between the RP's results and my comparative calculations were observed. The details of the differences in the capture of the limiting time points were not fully resolved in the time available, but overall, the limiting ELLDS values obtained by the RP and my TSC were very similar.

465. In its RQ responses the RP was able to close out the majority of the comments raised, and so the comparative assessments performed provide a basis for confidence that the RP has an understanding and, in the most situations, can correctly applying the R6 procedure.

466. That said, it will be important to keep track of the generic points identified above to ensure that they are suitably rectified, particularly given the RP's commitments to

resolve them during the site-specific stages. I have therefore raised several assessment findings to track these matters to a conclusion.

4.3.7.8 DTA Verification

467. ONR expects that where high reliability is required for components and structures or where otherwise appropriate, the sizes of crack like defects of structural concern should be calculated using verified and validated fracture mechanics methods with verified application (ONR SAP EMC. 34).

468. During Step 3 of GDA (Ref. 7), I sampled the RP's arrangements for managing verification, to gain confidence that the RP has robust verification processes in place to check the accuracy of the results presented, so that the meaning of the results can be considered properly. The RP explained that the quality control of DTA analysis and subsequent reporting will mainly be based on its internal processes, supplemented by the UK requirements for independent verification of DTA. Within the RP's procedures the verification scope covers input data, analytical models, detailed processes and results. The RP stated that it intends to conduct independent verification of all DTA work, and it was clarified that the Checker will be independent of the DTA production and carry out the calculations separately. According to the RP, 'independent checking' refers to a third party repeat verification commissioned by the RP and is separate from the DTA work carried out by the RP.

469. From my GDA Step 3 review, I concluded that the processes outlined by the RP did not fully meet ONR's expectations for verification of DTA, due to the significance of the DTA results. there appeared to be a degree of ambiguity as to the use of independent means. For those calculations that directly support a HIC claim, I expected the highest level of rigour be applied to the DTA and that these aspects are defined and documented, to ensure a consistent application of the analysis. I consider it necessary to continue my review of this topic as part of my GDA Step 4 assessment.

470. From my assessment of the DTAs in Step 4 of GDA, a number of errors were identified during the reviews in what the RP supplied as checked and approved documents. These included incorrect or inconsistent tabulated data and out-of-date references. Whilst the responses received from the RP were sufficient to clarify any issues identified, there was a reluctance to update documents to correct minor errors or to incorporate some of the useful information in responses to RQs that had been raised. Whilst this does not fundamentally impact the conclusions drawn relating to the RP's proficiency with using the R6 procedure, the RP's approach does not reflect accepted practice with regards to reporting and could lead to similar difficulties during the site-specific stages.

471. I am also mindful that I have raised several assessment findings relating to the RP's implementation of the R6 procedure. Several of these generic points I would have expected to be challenged through the implementation of a robust verification process. I recognise that the RP has developed its capability and proficiency in the use of the R6 defect assessment procedure over the course of the GDA, but noting the overall position, consider that further progress is needed in relation to the RP's informed customer capability for its DTA work. I therefore raise an assessment finding to ensure that this is tracked to a satisfactory conclusion:

AF-UKHPR1000-0195 – The licensee shall, as part of detailed design, develop and implement a robust verification process to underpin the defect tolerance assessment work in support of the avoidance of fracture demonstrations for high integrity components.

4.3.8 DTA Strengths

472. The RP's weld ranking process provided a reasonable and pragmatic means of selecting the HIC welds for detailed assessment during GDA.

473. The RP responded constructively to several RQs raised throughout the GDA. The majority of my comments were satisfactorily addressed by the RP, with the responses often providing useful and additional evidence that could improve the structural integrity case.

474. The RP has developed its understanding and proficiency in the use of the R6 procedure over the course of the GDA, this was evident from the results of my independent comparative calculations, which provides confidence that the RP has an understanding and, in most situations, can correctly apply the R6 procedure.

4.3.9 DTA Outcome

475. The RP has chosen a complex process to establish the limiting ELLDS values with the output ultimately dependent on the confidence in the transient definitions. The value and efficiency of the application of this process when balanced against the demanding timescales of licencing will warrant review. Although I am satisfied that overall an adequate level of conservatism is retained in the DTAs, it is important that any further reductions in conservatism are reasonable and justified.

476. I also have raised several assessment findings to ensure that the licensee addresses certain generic points in the application of the R6 procedure (AF-UKHPR1000-0192) and to improve the robustness of its verification arrangements including its intelligent customer capability with respect to its DTA work (AF-UKHPR1000-0195).

477. In conclusion, I judge that overall, there is an adequate level of conservatism in the DTAs to warrant their consideration in the reconciliation for the AOFD.

4.3.10 Review of Technical Justifications

478. During Step 2 of GDA, the RP described how within GDA it would substantiate its claim of high reliability Non-Destructive Examination (NDE) in support of the avoidance of fracture case (Ref. 111).

479. The RP's avoidance of fracture case for HICs aims to demonstrate that a defect that may be of structural concern would be readily detected and rejected during manufacture with high reliability. The RP has recognised that the manufacturing NDE prescribed by RCC-M and the supporting standards may not provide the level of reliability required for its avoidance of fracture case (Ref. 26). Consequently, the RP has included an inspection, that is additional to the code inspections, to be applied at the end of manufacture and that is targeted at detecting and rejecting the defects of structural concern. The RP described this as an 'objective-based' inspection (Ref. 131) (Ref. 132) to differentiate it from the code-based inspections; the emphasis is that objective-based NDE is specifically designed to meet a given set of objectives which here is to detect and reject defects of structural concern with high reliability.

480. The expectation for GDA is for the RP to demonstrate that high reliability non-destructive examination (NDE) can be performed during the manufacture of HICs and that this reliability can, in the future, be demonstrated through a formal process of inspection qualification. In practice, this entails applying NDE methods that are based upon sound physical principles and that are widely used in industry.

481. The RP stated in its 'Re-grouping Application of Weld Ranking Procedure' (Ref. 131) that the manufacturing inspections for HICs will be qualified in accordance with the ENIQ methodology (Ref. 16) comprising the following principal elements:

- The qualification of the NDE procedure and the NDE personnel are separated.
- NDE procedures are qualified through a combination of written technical justification and practical demonstration. The technical justification brings together the physical basis of the NDE techniques with experimental and theoretical evidence to support the ability of the NDE to meet the pre-defined inspection objectives.
- The qualification body (a body of experts acting independently of the inspection organisation and licensee) assesses the technical justification and the results from the practical trials. If the qualification body judges that the outcome is satisfactory, it will issue a qualification certificate or equivalent statement.
- NDE personnel are qualified to apply a specific inspection procedure; personnel qualification includes blind trials to demonstrate competence in applying the procedure.

482. The application of the ENIQ methodology is considered as relevant good practice within the UK for confirming that NDE systems (comprising procedures, equipment and personnel) have the required capability for delivering the specific inspection objectives.

483. In my opinion, the RP's strategy for qualifying the end of manufacture NDE of HICs (Ref. 131) is a well-developed document that describes the principal activities for inspection qualification along with the roles and responsibilities of the relevant organisations.

4.3.10.1 RP Approach for Qualified Inspections During GDA

484. Sub-section 4.3.1.1 describes how the RP applied a process of weld ranking to select a limiting set of HICs for its avoidance of fracture demonstration. This process identified limiting cases with regard to defect tolerance along with limiting cases for ultrasonic inspection (those deemed to be where ultrasonic inspection was most difficult). The number of limiting cases was further reduced for the purposes of demonstrating the capability of the NDE by, first grouping components according to similarities in geometry, materials and likely NDE techniques, and then determining a bounding case for each group (Ref. 131).

485. The RP presents its approach to providing evidence that manufacturing NDE is able to deliver the required capability in its ‘Inspection Qualification Strategy for High Integrity Component’ (Ref. 111). The main element of this approach was to present the evidence for the NDE capability in the form of ‘GDA technical justifications’. These documents were shortened versions of the technical justifications that would eventually be needed for a full qualification and included a summary of the inspection objectives, an overview of the proposed inspection techniques and available evidence to support the capability of the NDE.

486. The RP sought assistance from several UK organisations familiar with the UK expectations and in the approach required for demonstrating high reliability within the context of GDA. Where relevant, the impact of the specific arrangements is discussed in the later sections for the separate components.

487. I was satisfied that the RP’s grouping approach produced suitable limiting cases and that the GDA technical justifications were likely to provide sufficient evidence to support the avoidance fracture claims for all of the components within the group. The conclusions from my assessment of these bounding GDA technical justifications are presented below.

488. An essential input to objective inspections is a description of the defects to be considered and the overall aims of the NDE. The defect descriptions include the dimensions of the defects that are required to be detected along with their characteristics. The RP presented this information in the inspection specifications for

each of the components for which GDA technical justifications were produced (Ref. 133) (Ref. 134) (Ref. 135) (Ref. 136) (Ref. 137) (Ref. 138) (Ref. 139) (Ref. 140).

489. The RP recognises in each of the inspection specifications that the defect characteristics (shape, roughness, location, orientation) would eventually be determined by an expert elicitation panel. For the purposes of GDA, the RP has proposed defect descriptions based upon experience from similar components. The proposed defect descriptions are reasonably broad and are likely to encompass the potential manufacturing defects for the components in question.

490. The RP's approach to the defect descriptions includes postulated manufacturing planar defects that might be considered as arising from the more common defect mechanisms such as lack of fusion and weld cracking where the defects are aligned predominantly in the welding direction. The RP has also included what it considers to be more speculative defects that are aligned perpendicular to the welding direction.

491. While the assumed defect characteristics may be appropriate for the purposes of GDA, it is important that the licensee should commission expert elicitation from suitable experts.

AF-UKHPR1000-0196 – The licensee shall, as part of detailed design, justify the assumed manufacturing defect descriptions in inspection specifications for high integrity components. This should include, but not be limited to, the use of suitable expert elicitation.

492. The RP has defined the primary objectives of the inspection as being the detection and rejection of longitudinal defects at the QEDS or larger and for which high reliability must be eventually demonstrated through the full rigour of inspection qualification. It also has defined secondary objectives for the detection of transverse defects at the QEDS or larger for which a capability demonstration is provided but may not be subject to the full rigours of qualification.

493. Clearly, the validity of separating the primary and secondary objectives in this way will ultimately depend upon the more complete defect definitions produced by the expert elicitation post GDA. However, I am content that this distinction is a reasonable assumption for the purposes of GDA.

494. Ordinarily, the QEDS that is derived from the DTA should be available prior to the inspection design and justification. However, due to the time constraints of the GDA, it is recognised that the reliability demonstration for the NDE may need to be performed in parallel with the DTA. Consequently, the RP used assumed QEDS' that it judged would be reliably detected and rejected and would also support an adequate DSM once the DTA was complete. This eventual conclusion is presented in the reconciliation reports for each of the sampled components.

4.3.10.2 Assessment Approach

495. An important factor that influenced my approach to assessing the individual technical justifications was that the end-of manufacture inspection for each HIC will eventually be subject to the rigour of full qualification. Consequently, my assessment within the scope of GDA was to establish confidence that the:

- inspection objectives could be met using well established NDE techniques that were based upon sound physical principles. Here the principal objective is to detect and reject all planar defects at or greater than the QEDS;
- the component was, as far as reasonably practicable, designed to facilitate the end-of manufacture NDE; and

- there was a reasonable prospect that an inspection could be developed that would be successful in a full qualification exercise. In practice, this means that the ultrasonic techniques provide several means for providing good margins for detection and characterisation.

496. This confidence was established by basing my review of the GDA technical justification on the following:

- understand how the proposed techniques are based upon sound physical principles and are aligned with the principle of objective-based NDE;
- establish that the NDE techniques are well established and mature;
- judge whether the evidence and conclusions in support of high reliability is suitable and sufficient; and
- judge whether the limited evidence demonstrated sufficient margins against the inspection objectives to account for the uncertainties in the RP's analysis.

497. It is important to note that my assessment does not underwrite the use of any given set of NDE techniques nor restrict the development of additional techniques that may be required in full qualification.

4.3.10.3 GDA Technical Justifications: Generic Features

498. The RP chose ultrasonic inspection as a means of meeting the inspection objectives of high reliability detection and rejection of planar defects larger than the QEDS. Of the available volumetric NDE methods, ultrasonic inspection is considered the most reliable as it is not sensitive to the gape of planar defects. In contrast, the ability to detect planar defects using radiography (the main alternative volumetric technique) is critically dependent on defect gape and the alignment of the radiographic beam with the plane of the defect.

499. The RP produced the GDA technical justification in the format of an ENIQ style technical justification (Ref. 17). These technical justifications presented the inspection objectives and the proposed NDE methods along with evidence to support the claim that the NDE techniques are likely to be capable of meeting the inspection objectives. While it is not a GDA requirement to use the ENIQ format for such a purpose, I welcome this approach since it demonstrates the RP's understanding of the ENIQ methodology for inspection qualification and was helpful in seeing how the information could be extended to form part of a full qualification. As such the documents that have been produced during GDA are expected to be a basis for producing full technical justifications in the future.

500. The evidence presented to support the claimed capability of the inspection followed a similar pattern for each of the GDA technical justifications. Firstly, a set of 'worst-case defects' were identified; these are limiting defects that have a combination of parameters for which the inspection performance is judged to be the lowest. Theoretical evidence was then presented to support the inspection performance for these limiting cases.

[redacted]

[redacted]

503. While I accept the principle of presenting theoretical evidence within GDA, my assessment considers whether there was sufficient margin to compensate for any uncertainties in the results and lack of direct practical data.

504. The proposed method for defect characterisation (determining whether a defect is planar or volumetric) applies the method defined in EN ISO 23279 which compares the ultrasonic responses from a range of probes to determine the reflectivity as a function of beam angle. The physical basis of this method for characterisation seems sensible and is based on sound physical principles. The flow chart that is applied to establish the nature of the defect defaults to a planar defect at each stage and in this respect the method appears to be conservative. Noting that the RCC-M code requires that all planar defects are repaired, the apparent conservative nature of this evaluation method appears beneficial. However, weld repairs are often detrimental for structural integrity. Consequently, an overly conservative approach in defect characterisation that may result in unnecessary repairs is undesirable. I note that, while the application of industrial standards may provide good technical solutions, the high reliability that is demanded for an inspection of a HIC remains to be established; furthermore, the reliability of the approach defined in EN ISO 23279 will vary with the inspection situation and will need to be established during the full qualification exercise.

RPV Nozzle Flange Shell to Core Shell Weld

505. The RP chose the RPV nozzle flange shell to core shell weld as a relatively simple case for its first demonstration of high reliable NDE in support of its avoidance of fracture claim.

506. The inspection specifications (Ref. 138) (Ref. 141) included:

- a primary scope of detecting and rejecting longitudinal defects for which high reliability of detection and rejection must be demonstrated. The target defect size for the primary scope (the QEDS) was 10mm through-wall by 60mm long.
- a secondary scope for which the demonstration is less stringent. This secondary scope included transverse defects having a size of 10mm x 30mm and rough longitudinal defects having a size of 10mm x 60mm with a tilt range of up to 20º.

507. From my review of Revision A of the partial technical justification (Ref. 141), it was not clear to me that the ultrasonic techniques were based upon sound physical principles or that the principle of objective-based inspections had been appropriately considered. I therefore asked the RP to clarify in its future documents how these concepts had been appropriately considered. Furthermore, there were numerous quality related shortfalls in the document that reduced my confidence in the RP's review and verification process. I raised a number of points for clarification in RQ-UKHPR1000-0469 (Ref. 28) to address these shortfalls and to communicate my expectations for the subsequent technical justifications.

508. While Revision B of the partial technical justification (Ref. 138) represented a significant improvement, I was not convinced that the principle of objective-based inspections had been fully embraced. My reasons for this conclusion were:

- The technical justification includes scans that do not appear to add any value.

- There is a reference out to the RCC-M code, apparently to indicate that the number of beams described in the technical justification exceeds that of the code. While this may be the case, it is not deemed relevant.
- Compliance to industry standards appears to be important. While many standards may be relevant, attempting to comply with others may inhibit the development of optimum techniques.
- The technical justification describes scanning restrictions for the standard set of probes, there are instances where a different choice of angles might be beneficial.
- The partial technical justification described the detectability of defects against a fixed reporting threshold (relative to a calibration reflector). As there was no discussion of the noise level, it was not clear whether there was any scope for reducing the reporting threshold and thereby increasing the detection margins.

509. The RP submitted a further revision of the 'RPV Nozzle Flange Shell to Core Shell Weld' taking into account the above feedback, which addressed the above points and was a significant improvement on the previous versions. Importantly, this provided me with confidence that the RP had a sufficient understanding of the relationship between the NDE and the avoidance of fracture demonstration.

510. The proposed inspection comprised of conventional pulse-echo and tandem scans from all the available surfaces including a specific scan from the tapered surface optimised to detect defects at the outer surface of the weld. The tandem scans are important for providing detection of near vertical embedded defects (defects located away from the inner and outer surfaces). The use of skewed tandem scans was offered to improve the detection of skewed defects.

[redacted]

512. Overall, I was satisfied that the RP had demonstrated an understanding of the concept of high-reliability, objective-based NDE in support of its avoidance of fracture case. The proposed NDE techniques were mature and based upon sound physical principles. In my opinion, the GDA technical justification had presented an appropriate level of evidence, based largely on theoretical arguments, with adequate margin for detection.

513. No separate evidence was presented for the secondary scope as the GDA technical justification argued that the ultrasonic scans used for both longitudinal and transverse scans were similar and the effect of the vessel curvature is small. Consequently, the modelling evidence used for the longitudinal defect detection was also relevant for transverse defects. While the modelling considers defects with tilts up to 5º, it is reasonable to assume that secondary defects with tilts up to 20º are favourably oriented for detection by high angle probes.

514. I am satisfied that through the GDA technical justification, the RP has demonstrated that:

- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 10mm x 60mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques for the secondary objectives are likely to have good capability for detecting and rejecting transverse defects having a size of greater than 10mm x 30mm and for defects with tilts up to 20º tilt.

MCL Hot Leg to SG Inlet Nozzle Safe End Weld

515. The RP selected the weld joining the MCL hot leg to the SG inlet nozzle as a limiting case to be the subject of a GDA Technical Justification (TJ). The principal feature of this weld that requires particular attention is the potentially higher ultrasonic attenuation in the weld and parent material when compared to low alloy ferritic steels. Furthermore, the anisotropic grain structure in the weld distorts the ultrasound which can adversely affect defect detection and characterisation. Consequently, I focussed my assessment on reviewing how the RP had taken these factors into consideration in identifying suitable NDE techniques and in the evidence provided to support the capability for detecting the defects of concern.

516. The inspection techniques described in the GDA technical justification (Ref. 142) applied a range of angle compression and shear wave probes from the external surface of the MCL pipe and SG safe-end. The RP claimed that inspections from the inside surfaces were not possible due to access restrictions and concluded that the ultrasonic attenuation was sufficiently low that high reliability could be achieved from external scans alone.

517. The proposed inspection applied angle compression and angle shear wave probes in either direct pulse echo or in a tandem configuration; the latter to provide detection capability for near vertical defects.

518. I reviewed Revision B of the TJ (Ref. 143), and raised several questions and clarifications in RQ-UKHPR1000-1653 (Ref. 28).

519. Generally, I was not satisfied that the RP had adequately considered uncertainties associated with the ultrasonic attenuation of the pipe and safe-end forgings. By applying probes from only the external surface meant that the ultrasound for some probes would traverse a relatively large distance to reach the inner surface. The technical justification argued that the grain size for both sets of forgings would be small and that the corresponding ultrasonic attenuation was low and would therefore not impede inspections from the external surfaces. Unlike low-alloy ferritic forgings, the grain size of austenitic stainless steel and nickel alloy forgings can be difficult to control, and, in my opinion, there remained a risk that the attenuation values claimed in the technical justification may not be achievable in practice.

520. I recognised that this was a site weld and consequently physical access would be more difficult than that available if the weld was laid within a factory environment. Nonetheless, I judged that it might be reasonably practicable to scan ultrasonic probes from the inner surface and asked the RP to consider whether such scans were indeed feasible. I noted that the RP, in its consideration of ‘design for inspectability’ had extended the counterbore region that would enable probes to be scanned from the inner surface. The RP explored this possibility and concluded that, in the context of providing high reliability NDE for a HIC weld, it was practicable and appropriate to apply scans from the inner surface; the RP included such scans in Revision C of the technical justification (Ref. 144). I was satisfied that this approach, along with the choice of probes was sufficient to mitigate the uncertainty associated with ultrasonic attenuation.

521. In RQ-UKHPR1000-1653 (Ref. 28), I sought further clarification of the RP’s approach related specifically to the following points:

- The description of the self-tandem techniques was incorrect to the extent that they could not be applied in practice.
- I judged that some of the probes would not be effective at the required ranges.
- The parameters for some of the probes did not appear to be correctly described and consequently it was not clear that they would be effective.

522. I would have expected these points to have been noted before the technical justification was issued and I used RQ-UKHPR1000-1653 to understand the effectiveness of the RP's review process. The RP responded by undertaking a root cause analysis and implementing enhanced scrutiny of such documents.

523. From my review of Revision C of the GDA technical justification (Ref. 144), I conclude that:

- The proposed inspection techniques are judged to be appropriate for the MCL pipe to SG safe-end weld and consider the level of attenuation assumed and the anisotropy of the material.
- The inspection sensitivities defined for each of the scans are expected to be achievable in practice.
- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 8mm x 48mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques for the secondary objectives are likely to have good capability for detecting and rejective transverse defects having a size of greater than 8mm x 24mm.

Main Steam Isolation Valve Casting Repair Weld

524. Unacceptable defects found in the MSIV cast body, reported by any of the in-process NDE stages, are removed by grinding and then repaired by filling in the excavated region with weld metal. The resulting repair welds are separated into minor and major repairs. Minor repair welds are those that have dimensions less than the QEDS and will not be subjected to the qualified inspection performed at the end of manufacture (Ref. 145). The RP's argument here is that manufacturing defects arising from the repair process cannot be of a size of structural concern. Consequently, the qualified high reliability NDE performed at the end of manufacture is constrained to major repair welds; those that could contain a manufacturing defect at the QEDS or larger.

525. I note there is a potential difference in the way that allowable major repair weld excavations are described in the inspection specification (Ref. 145) and the GDA technical justification (Ref. 146). The former states that it may be necessary, due to local geometry, to excavate a defect from the surface furthest away from the defect position. In contrast, the technical justification states that any defect that requires an excavation of greater than 40mm will cause the cast valve body to be scrapped. For my assessment, I have worked on the basis that any weld repair greater than 40mm deep would lead to a rejection of the whole cast valve body.

526. The technical justification (Ref. 146) states that the shape of any major excavation is a balance between the removal of as little weld metal as possible and producing a profile that allows for effective ultrasonic inspection. The main features of allowable shapes are:

- uniform in plan view (circular, elliptical or extended elliptical);
- flat bottom to the excavation; and
- walls that are inclined by 30º to the surface normal.

527. Unacceptable manufacture defects can occur at any location in the casting body and consequently, it is not possible to know in advance the shape of the excavation and the inspection conditions for each repair weld. While general principles for the inspection design can be described, there will be aspects of the NDE techniques that will be bespoke to a specific repair weld.

528. I have inferred from the inspection specification and the GDA technical justification that the weld metal is laid down in a uniform linear sequence, and that where there is a major axis (for elliptical and extended elliptical repairs) the weld direction will be in the major axis direction.

529. For the purposes of GDA, the RP assumes that the defects of concern for the qualified end of manufacture inspection are planar smooth and rough defects on the weld fusion face or in the body of weld repair. The primary inspection objective is to detect and reject planar defects on the fusion boundary and aligned with the welding direction in the weld body having a size at the QEDS (through-wall 10mm x 60mm long) or greater.

530. A secondary objective has been defined that is for defects in the body of the weld repair aligned transverse to the welding direction. The latter includes the detection and rejection of defects that are:

- 10mm (through-wall) x 30mm (long) for defects in the weld body or at the fusion faces.
- 10mm x 60mm lying at the bottom flat surface of the repair.

531. As for the other cases presented in GDA, the primary inspection objectives are the subject of a qualified inspection to demonstrate high reliability. The capability of the secondary objectives is to be demonstrated through capability statements and will not be subject to the full rigour of inspection qualification.

532. The RP has sought to demonstrate its approach to inspecting any major repairs by selecting an appropriate 'difficult' repair location (Ref. 146). The GDA technical justification presents its approach to developing appropriate techniques and presents arguments and evidence based upon physical reasoning and theoretical modelling. The selected case is for a weld repair 40mm deep located in the body crotch region where the inlet/outlet nozzle meets with the body neck. I note that the selected example, is symmetric about the top-dead centre of the valve body and that, in practice, a real weld repair location is likely to be more complex. Nonetheless, I am satisfied that this is a reasonable selection of location as it has the smallest radius of curvature and has a complex saddle shape.

533. The proposed techniques comprise ultrasonic probes with a range of angles which are scanned from all the available surfaces. The technical justification demonstrates the general approach of tailoring the probe angles to meet the specific case being considered. Furthermore, it states that, where necessary, the probe shoe will be shaped to match the scanning surface to maintain coupling, for example when scanning in the crotch region where the radius of curvature is lowest.

534. The probes are scanned predominantly in a direction perpendicular to the welding direction for the primary scope. The technical justification also states that towards the end of the repair weld the weld fusion face may not be aligned with the welding direction; in such regions the inspection should also include scans perpendicular to the fusion face.

535. The technical justification presents coverage diagrams for each of the scans (probe and surface combinations) and shows the inspection volume is covered by an ultrasonic beam that is expected to provide an effective detection mechanism. [redacted]

536. I note that the coverage diagrams and the modelling have simplified the geometry by assuming that the plane containing the ultrasonic beam and the section through the repair weld are coplanar. In practice, the situation is more complex, particularly for scans applied from the opposite surface to the repair weld. Consequently, it will be

necessary to consider the full three-dimensional situation for each repair weld case and define scans that provide the appropriate coverage and angle of incidence to the defect.

537. Notwithstanding the above limitation, I am satisfied that the RP has presented sufficient information within GDA as to how ultrasonic techniques will be developed for specific cases of repair welds. The arguments and the evidence presented in the GDA technical justification enables me to judge that an ultrasonic inspection, when fully developed:

- is likely to meet the primary inspection objective of detecting defects at the QEDS or larger; and
- will have a capability that will be demonstrated through inspection qualification.

538. The GDA technical justification and my assessment focusses its attention on the primary inspection objectives that are the subject of the qualified inspection. If similar principles are adopted in developing techniques for the secondary objectives, I would expect an appropriate level of capability to be demonstrated.

MSL Connection to the Penetration Flange Weld

539. The RP selected this as the limiting case for a MSL weld (Ref. 147) due to the potential scanning restriction imposed by the penetration flange. Due to the potential impact of the scanning restriction (limited coverage with high angle probes from the flange side of the weld), the RP introduced some additional probes to be scanned from the inside surface.

540. While the inspection equipment will be specified during the site-specific stages, the RP proposed automated (mechanised) scanning with digital data collection and offline analysis. The technical justification implies that this option is preferred over manual scanning to enhance the reliability of the inspection. It does however, state that the manual techniques may be required for scans from the inner surface. I welcome the use of automated techniques which reduces the potential human factor errors that can occur with manual inspection.

541. Other than the scanning restriction on the flange side, there are no significant challenges to the ultrasonic inspection.

542. A revised GDA technical justification was submitted (Ref. 148), which presented modelling evidence (this has been summarised in Sub-section 4.3.5) to support the effectiveness of the ultrasonic techniques and concludes that good margins exist for the primary objectives.

543. I am satisfied that through the latest GDA technical justification (Ref. 148), the RP has demonstrated that:

- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 5mm x 30mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques for the secondary objectives are likely to have good capability for detecting and rejective transverse defects having a size of greater than 5mm x 30mm.

Reactor Coolant Pump Flywheel

544. The RP selected the reactor coolant pump flywheel as a limiting case for a non-welded HIC, with the access restrictions presenting some challenges to the deployment of ultrasonic scans (Ref. 111).

545. The defects included in the primary scope are identified as smooth or rough defects either parallel to the top and bottom surfaces or oriented in the axial circumferential direction. The secondary scope included defects in the radial circumferential planes.

546. The RP has presented evidence for its claim of a high reliability inspection at the end of manufacture performance in a 'capability statement' (Ref. 149). While the title of the document is different from the other HICs considered for its avoidance of fracture demonstration the format and general content was the same as the GDA technical justifications. Furthermore, the capability statement confirms that the inspection system (procedure and personnel) will be qualified in the same way as the other HICs. Consequently, my assessment has considered the RCP flywheel in the same way as for the other HICs in this section.

547. The inspection comprises 0º, 38º, 45º and 70º probes scanned from all of the accessible surfaces in the principal directions. The capability statement includes the use of Time of Flight Diffraction (TOFD) but does not explicitly clarify that the technique is not claimed as a primary detection method as is the case for other HICs (see the discussion for the RPV inlet nozzle to shell weld). Notwithstanding this, it appears that the required detection performance can be delivered by pulse-echo techniques alone with TOFD being used for defect evaluation.

548. The capability statement is inconsistent regarding the method of deployment. Some parts of the document discuss the manipulator and digital data acquisition systems whereas other parts define the requirements for manual scanning and manual flaw detector. However, since the flywheel forging has a relatively simple geometry, I do not believe that the deployment method is material to my assessment within GDA.

549. The capability statement presents the physical basis for the inspection design and presents evidence through coverage diagrams and theoretical modelling to demonstrate that there are adequate margins for the detection of the primary scope of defects.

550. Overall, I am satisfied the capability statement for the RCP flywheel is sufficient for demonstrating that:

- The proposed NDE techniques when fully developed are likely to detect and reject defects contained with the primary scope (planar defects the QEDS of 10mm x 60mm) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques are likely to have good capability for detecting and rejecting defects within the secondary scope.

RPV Inlet Nozzle to Safe End Weld

551. The RP selected the weld joining the RPV inlet nozzle to the austenitic stainless steel safe-end as a bounding case for the group of nozzle to safe-end dissimilar metal welds; the other dissimilar welds being between the steam generator primary nozzle and safe-end, and the RCP nozzle to safe-end. The RP concluded that the access and thicknesses were similar across all of these welds, but it was likely that the RPV case would having the lowest QEDS.

552. The characteristic features of the dissimilar metal welds are:

- the coarse grain anisotropic structure of the weld and buttering and the potential for high ultrasonic attenuation in the safe-end;
- the buttering layer between the low alloy ferritic steel nozzle and the main weld introduces an additional fusion face and increases the overall weld width; and
- the presence of coarse grain anisotropic cladding on the inner surface of the nozzle.

553. The RP identified a primary scope of objectives (Ref. 137) as being high reliable detection and rejection of smooth and rough longitudinal planar defects (aligned in the welding direction) at the QEDS of 5mm x 30mm, or larger. The range of defect tilt and skew was 0 ± 8^{o} and 0 ± 5^{o} respectively.

554. A secondary scope was defined as being the detection and rejection of smooth and rough transverse defects with tilts and skews of up to 20º and having a size of 5mm x 30mm and rough longitudinal defects with tilts of up to 20º and size of 5mm x 30mm.

555. The proposed ultrasonic techniques described in the GDA technical justification (Ref. 142) included scans from all of the inspection surfaces available at the end of manufacture and included scans from the end face of the safe-end ring. The scans include conventional pulse echo techniques, using angle compression wave probes to minimise attenuation and beam distortion and tandem arrangements for the detection of embedded near vertical defects. The tandem techniques included both two probe tandem and self-tandem arrangements.

556. The GDA technical justification stated that, in addition to the inspection of the completed weld it would be prudent to apply a 0º probe on the buttering surface prior to making the main weld. This would provide good detection of defects at the fusion face between the buttering and the nozzle.

557. Probes scanned in the circumferential direction were designed to detect transverse defects.

558. In general, I was satisfied that the proposed techniques had been targeted at the defects of interest and had considered to some extent the main features of the weld. I noted that while the GDA technical justification had considered the attenuation in the weld material, I did not believe it had included the effect of the weld anisotropy. It was also not clear how the potential ultrasonic attenuation in the stainless-steel forging had been considered and whether the noise level would allow the inspection sensitivity to be increased if necessary. I raised these points in RQ-UKHPR1000-0958 (Ref. 28) to which the RP responded positively. The RP's response provided more detail regarding the welding process and the likely anisotropic structure that would result. It then provided more information as to how the impact of the grain structure on inspection performance could be minimised. The RQ response also included further information to support the inspection sensitivity.

559. I noted that the twin probe tandem technique used angle compression wave probes in contrast to the more conventional angle shear wave probes. Similarly, angle compression wave probes were used to detect surface breaking defects via the corner trap mechanism whereas shear waves are known to give a much higher corner response. I sought clarification of my observations in RQ-UKHPR1000-0958 (Ref. 28). The RP's response recognised that an angle compression wave probe tandem arrangement was relatively unfamiliar and would require further experimental work to justify its capability. Angle compression wave probes had been selected to minimise the attenuation and distortion effects of the weld, buttering and cladding. The RP also recognised that the corner trap signal using angle compression wave probes is significantly weaker than when angle shear waves are used and explained that

compression waves were preferred to minimise the effect of the coarse grain anisotropic weld material. The RP supported this choice with evidence of the ability to detect surface breaking defects with angle compression wave probes.

560. The GDA technical justification presented evidence for the primary scope of defects only. [redacted] The technical justification recognised that the signal to noise ratio seen in the dissimilar metal welds is lower than that for low alloy ferritic steel welds and consequently the margin for detection is reduced. Nonetheless the results showed that the defects of concern would be reliably detected by more than one beam and that, if necessary, the probe parameters could be fine-tuned to improve the signal to noise ratio during the future development of the inspection procedure post GDA.

561. While no evidence was provided for the capability of the secondary scope, physical reasoning and coverage diagrams were presented for the detection of transverse scans. The technical justification explained that practical evidence gathered from realistic welds would be required to demonstrate the detection capability for transverse defects. I note that reasonable capability is expected for the detection of transverse defects in the inner and outer regions of the welds, but detection is more problematic for mid-wall defects due to the relatively long beam paths through attenuative weld metal.

562. The secondary scope of defects also includes longitudinal defects with a higher range of tilts (up to 20º). Here it is reasonable to accept that such defects will be readily detected as the higher tilts reduce the angle of incidence for some of the beams included in the proposed inspection.

563. I am satisfied that ultrasonic inspection techniques have been proposed that take account of the situation seen in dissimilar metal nozzle to safe-end welds and that an understanding of the inspection challenges in these welds has been demonstrated.

564. Overall, I am satisfied that through the GDA technical justification, the RP has demonstrated that:

- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 5mm x 30mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques for the secondary objectives are likely to be capable of detecting transverse defects having a size of greater than 5mm x 30mm in the inner and outer regions of the weld. Practical evidence is needed to establish this capability and further inspection development would be needed to detect transverse mid-wall defects.

SG Tubesheet to Primary Head Weld

565. The RP selected this component (Ref. 150) as the limiting case for a steam generator weld due to the scanning restriction imposed by the tubesheet on one side of the weld and the transition from the straight to curved section of the channel head on the other.

566. It is understood that the inspection will be performed following the post weld heat treatment at which point I expect the divider plate will have been welded to the inside of the primary head. There is, however, no mention in the inspection specification (Ref. 151) nor the technical justification (Ref. 150), of the divider plate presenting any impediment to the inspection of the tubesheet to primary head weld. In practice, this may not present a significant issue as the divider plate provides additional

reinforcement and the defect tolerance assessment presented in RO-UKHPR1000-0006 (Ref. 128) would not be applicable. Nonetheless, the divider plate weld (which is a fillet weld) would restrict the circumferential coverage for the region of the tubesheet to primary head weld that is adjacent to the divider plate. For axial scans (applied for the primary scope of the inspection) this would provide a scanning restriction of more than half of the footprint and may affect the detection and evaluation (particularly length sizing) of any defects in this region. Furthermore, the detection of transverse defects (secondary objectives) would be more severely impeded as the scanning restrictions would be more severe.

567. While the inspection equipment will be specified during the site-specific stages, the RP proposed automated (mechanised) scanning with digital data collection and offline analysis. I welcome the use of automated techniques which reduces the potential human factor errors that can occur with manual inspection.

568. The proposed inspection techniques combine scans from both the inner and outer surfaces of the weld in both directions. The inspection also includes scans from the outer curved surface of the channel head to good effect where the curvature improves the angle of incidence for defects in some regions.

569. The revised GDA technical justification (Ref. 152) presented modelling evidence (this has been summarised in Sub-section 4.3.5) to support the effectiveness of the ultrasonic techniques and concludes that good margins exist for the primary objectives.

570. I am satisfied that through the GDA technical justification (Ref. 152), the RP has demonstrated that:

- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 10mm x 60mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- The proposed NDE techniques for the secondary objectives are likely to have good capability for detecting and rejective transverse defects having a size of greater than 10mm x 60mm.

RPV Inlet Nozzle to Flange-nozzle Shell Weld

571. The RPV inlet nozzle to flange nozzle shell weld represented a case of a large diameter nozzle to shell weld where the geometry varies continuously around the weld circumference.

572. Revision A of the GDA technical justification (Ref. 153) fell short of my expectations in several key areas.

573. Firstly, the proposed inspection included the use of the TOFD technique as a primary detection method, and it was initially argued that TOFD was essential in providing the high reliability inspection capability. In the context of GDA, I judged that claiming TOFD as an essential primary detection method did not provide sufficient confidence that a high reliability inspection was achievable. I recognise that TOFD can play a role in evaluating defects that are detected by other means, but it is not sufficiently robust to be claimed as a primary detection mechanism. My reasons for this judgement are:

- A premise for reliable inspection is that inspection techniques are based upon sound physical principles, for example specular reflection of ultrasound from the face of a defect. In such cases, high signal responses are expected that provide good margins for detection.

- Detection with TOFD relies on diffracted signals from the top and bottom of planar defects. These signals are inherently weak and the signal to noise ratio is generally low.
- Defect detection using TOFD generally relies on the NDE operator identifying patterns of signals above the background noise from a 2-D image.
- It is difficult to reliably quantify signal responses with respect to calibration reflectors and consequently it can be difficult to demonstrate reliable detection.
- The diffracted signals are dependent on the characteristics of the defect tips which has not been defined in sufficient detail for the defects of concern.

574. In RQ-UKHPR1000-1279 (Ref. 28), I sought further clarification from the RP on its proposal to use TOFD as an essential primary detection method for high reliability inspection. The RP responded with a revised TJ (Ref. 154), where TOFD was retained as one of the inspection techniques but not claimed as a primary detection mechanism.

575. In my opinion it was not possible to draw reliable conclusions regarding the inspection performance from the evidence presented in Revision A of the technical justification. A significant piece of the evidence referred to a study of the ultrasonic inspection for a boiling water reactor nozzle to shell weld.

576. While this was a 'set-through' nozzle similar to that of the EPR™ RPV inlet nozzle to shell weld, the dimensions were sufficiently different that it was difficult to extrapolate the results reliably. Furthermore, I did not believe that the technical justification had presented an adequate demonstration for the reliable detection of rough defects.

577. I regard Revision B of the GDA TJ (Ref. 154) as a significant improvement in the following areas:

- [redacted]
- The variable geometry around the circumference of the weld was considered. Here, the four cardinal positions of the weld were assessed along with circumferential positions of 45º, 135º, 225º and 315º. The intermediate locations were selected as these are where the geometry has the greatest effect of deflecting the beam out of the plane containing the defect normal.
- The technical justification recognised the importance of having a scanning mechanism that could provide a variable skew for probes deployed from the vessel surface. This mechanism is required to compensate for the geometrical deflection of the ultrasonic beam out of the plane containing the defect normal.

578. The RP's proposed inspection techniques consisted of a range of ultrasonic probes scanned from all available surfaces:

- Scans from the nozzle bore were shown to be effective for defects located in much of the inspection volume with the precise coverage varying with the azimuthal position. Angle shear wave probes were also applied from the nozzle bore to detect defects near the inner surface via the corner trap mechanism.
- Scans from the external vessel surface make use of the vessel curvature to provide ultrasonic beams close to normal incidence on the defect at appropriate azimuthal positions. The modelling calculations show correspondingly high amplitudes with respect to the reference reflector. I note however, that relatively long ranges are involved in these cases and in practice, probes with larger crystals than those proposed (20mm x 22mm) may be required to achieve an adequate signal to noise ratio.
- Similarly, scans from the inner surface on the nozzle side of the weld made use of the curvature to reduce the angle of incidence to the defect.

- Tandem and self-tandem scans are proposed at an azimuthal position of 180º and up to 10º either side of this position. These are to provide reliable detection of vertical and near vertical defects. It is not possible to perform the tandem inspections at an equivalent azimuth of 0º due to the scanning restriction imposed by the flange.
- The GDA technical justification proposed skewing the ultrasonic beam by either rotating the probe mechanically for scans from the vessel surface or using wedges for scans from other surfaces such as from the nozzle bore.
- The ultrasonic techniques used to detect longitudinal defects (the primary scope) and applied from the vessel and nozzle surfaces (not the nozzle bore), are proposed for the detection of transverse defects (secondary scope) by scanning in the circumferential directions (with respect to the nozzle axis).

[redacted]

580. The GDA technical justification focussed its attention to providing evidence for the primary scope of the reliable detection longitudinal defects and did not present any detailed evidence to support the secondary scope of detecting transverse defects. While I accept that it is reasonable to accept that an effective inspection procedure can be developed for the eventual end of manufacture NDE, its capability will need to be supported by an appropriate level of evidence.

581. I am satisfied that through the GDA technical justification, the RP has demonstrated that:

- The proposed NDE techniques are likely to detect and reject longitudinal defects at the QEDS of 10mm x 60mm (the primary inspection objective) with high reliability.
- The NDE techniques when fully developed and implemented through full inspection procedures are likely to succeed through inspection qualification.
- It is likely that suitable NDE techniques can be developed for the secondary objectives of detecting and rejecting transverse defects having a size of greater than 10mm x 30mm.

4.3.11 Reconciliation

582. My assessment of the RP's approach and implementation of its reconciliation process to support its AOFD focussed on three aspects:

- The role of AOFD in the RP's safety case and its reconciliation strategy.
- The reconciliation examples provided for RO-UKHPR1000-0006 (Ref. 49).
- Review of the overall position for the AOFD (QEDS and DSMs) taking cognisance of the inputs from the materials, DTA and part TJs.

583. My views on the veracity of the RP's approach to reconciliation were informed with progress on RO-UKHPR1000-0006, which served as a means to clarify my expectations. Accordingly, a summary of the RP's response to RO-UKHPR1000-0006 forms an integral part of my assessment, with further details available (Ref. 129).

584. I raised RO-UKHPR1000-0006 (Ref. 49) to:

- Address the gaps identified during my GDA Step 2 Structural Integrity assessment (Ref. 4) and clearly articulate the regulatory expectations;
- Ensure that the avoidance of fracture demonstration considers the holistic aspects and does not impart unrealistic burdens on the individual factors for HIC (e.g. DTA, material properties and inspection activities);
- Gain confidence that the RP understands the conditions for use of avoidance of fracture demonstration and that the RP has satisfactory processes to strike the required balances on the contributing elements of the demonstration.

585. To capture the above items, several actions were raised under RO-UKHPR1000-0006 (Ref. 49), which included:

- ROA1 - Role and importance of the avoidance of fracture demonstration to the UK HPR1000 safety case.
- ROA2 - Identification of the contributing elements of the avoidance of fracture and their relationships.
- ROA3 - Justification of the inputs used in the defect tolerance assessments

4.3.11.1 Assessment Strategy

586. In response to RO-UKHPR1000-0006 (Ref. 49), the RP provided a series of technical submissions to present further evidence, reasoning and clarifications to underpin and demonstrate application of its reconciliation process.

587. I reviewed the prominence of the avoidance of fracture demonstration in the RP's safety documentation. This was important because an adequate avoidance of fracture demonstration is reliant on an integrated approach involving several technical specialists within the structural integrity discipline. I needed to gain assurance that the RP understood the significance of its avoidance of fracture demonstration in underwriting its HIC claims for the GDA.

588. I then focussed on the RP's strategy document for reconciling the DTA, with NDE and the material properties, notably, the RP's plan and expectations with respect to verifying fracture toughness values to underwrite the DTA. The RP should also show a basis for confidence in the achievement of inspection qualification, not only for the welds identified in RO-UKHPR1000-0006 (Ref. 49), but more widely for a set of challenging HIC welds selected from the RP's weld ranking.

589. Finally, I sampled the RP's application of the approach through two example RPV HIC welds provided in response to RO-UKHPR1000-0006 (Ref. 49). The aim here was establish whether the approach to reconciliation was applied appropriately, the inputs were adequately conservative and that overall, the claimed defect size margins (DSM) were reasonable i.e. an adequate level of defect tolerance was demonstrated with sensible demands being placed on NDE and the expected achievement of fracture toughness values.

4.3.11.2 Role of AOFD in the Safety Case

590. The RP responded to ROA 1 of RO-UKHPR1000-0006 (Ref. 49) by improving the clarity of how its avoidance of fracture demonstration supports its safety case. Notably, by expanding the description of the evidence in Chapter 17 of the PSCR (Ref. 80) and in Appendix D of the 'Safety Case Methodology for HIC and SIC Components', (Ref. 26) (Ref. 155). In addition, as part of the development of its avoidance of fracture demonstration for HIC components, the RP identified the need to provide a basis to reconcile the key inputs. To inform my assessment, I sampled these documents along with a number of other related submissions as appropriate.

591. I noted that although the importance of reconciliation was outlined in (Ref. 26), it appeared that a specific argument covering reconciliation for the Avoidance of Fracture demonstration was not proposed by the RP. I raised RQ-UKHPR1000-0807 (Ref. 28) to clarify whether the RP intended to provide a specific claim/argument in its avoidance of fracture demonstration to cover the importance of reconciliation, and if so where and when the evidence would be captured in the structural integrity safety case.

592. In its response to RQ-UKHPR1000-0807 (Ref. 28), the RP confirmed that a specific claim and argument featured in the 'Safety Case Methodology' document (Ref. 155). Nevertheless, the RP acknowledged that there was an absence of linkage to the evidence from the DTA, high reliability NDE and material property values to underwrite the reconciliation process for its AOFD. The RP proposed several changes to the 'Safety Case Methodology' (Ref. 155). The main changes to the safety documentation involved clarifying the role of the AOFD, the contributing factors and their relationship, through a route map to the claims, arguments and evidence. The RP also committed to developing its reconciliation strategy document and providing a reconciliation report for each of the HIC weld and limiting feature selected in the GDA. The specific evidence covering reconciliation would be presented in Revision E of the Safety Case Methodology for HIC and SIC Components (Ref. 26) and updated CSRs for HIC components.

593. I was content that the commitments made by the RP had met the intent of ROA 1. In addition, I subsequently confirmed that the 'Safety Case Methodology for HIC and SIC Components' (Ref. 26) had captured the need to link the evidence from DTA, NDE and material properties (ONR SAP EMC. 3, Ref. 2) via a reconciliation process to the claims/arguments proposed for the AOFD.

594. The generic topic of the traceability of the claims, arguments and evidence within the SI safety case was the subject of RQ-UKHPR1000-0661 (Ref. 28) and my sampling a selection of the CSRs, is addressed in Sub-section 4.1 on my report.

4.3.12 Reconciliation Strategy

595. The status of ROA 2 and ROA 3 of RO-UKHPR1000-0006 was informed by the RP's delivery and evidence to meet the intent of the actions, in particular, assessment of the reconciliation strategy and its application in accordance with the RO-UKHPR1000-0006 Resolution Plan (Ref. 49).

596. In response to ROA 2, the RP provided several documents (Ref. 122) (Ref. 156) (Ref. 157) (Ref. 124) (Ref. 158) (Ref. 159) to describe and demonstrate its strategy for avoidance of fracture reconciliation I sampled these documents along with several other related submissions. My expectation was that there was structured and robust approach, which clearly indicates how adequate DSMs are demonstrated along with the actions to be taken if the initial analyses do not provide adequate margins. I raised RQ-UKHPR1000-0452 to obtain information on the RP's reconciliation process and the format in which it would be documented (Ref. 28).

597. In its response to RQ-UKHPR1000-0452 (Ref. 28) the RP proposed three rounds of reconciliation:

- DTA and TJs undertaken in parallel with conservative inputs and methods in DTA and with the TJs informed by engineering practice and previous GDA experience (ONR SAP EMC. 2, Ref. 2);
- revision of the DTA inputs / assessment criteria; and
- revision of the TJ.

598. At a high level this approach appeared reasonable. The application of the reconciliation process was discussed at a technical exchange meeting (Ref. 160),

where it was apparent that several of the DSM values proposed by the RP were well above the target DSM value of 2 (ONR SAPs EMC.28, EMC.33, Ref. 2 and para. 5.92ff, Ref. 5). I was concerned that the RP had in some cases focused on achievement of large DSM values without recourse to ensuring that the underlying inputs were supportable. In consequence this could result in undue demands for the qualification of defect sizes in the TJs which were approaching the limits of the NDE capabilities, rather than placing the emphasis on the high reliability inspections.

599. I reiterated to the RP that within GDA, demonstration of avoidance of fracture reconciliation should be a balance between demonstrating a basis for confidence in achievable QEDS values from reliable NDE, a conservative DTA with support of a plan to verify materials properties, in particular, direct fracture toughness testing. Thus, pushing the limits of NDE capability to achieve a DSM significantly above 2 should not be the driver for going beyond the expectations for the AOFD. The RP's DTA and TJ teams therefore needed to work together to achieve an appropriate balance in the inputs and hence a meaningful reconciliation.

600. I subsequently assessed the RP's 'Avoidance of Fracture Reconciliation Strategy' (Ref. 161), and its initial application and raised RQ-UKHPR1000-1247 (Ref. 28) with my main queries covering the aim of the RP's reconciliation process, the selection of QEDs values, achieving balance in the reconciliation, and further clarification of the presentation of the claims, arguments and evidence for individual HIC components.

601. The response to RQ-UKHPR1000-1247 (Ref. 28) clarified that the main aim of the reconciliation process was to demonstrate an adequate DSM of about 2, through DTA, with a basis for confidence in the achievement of inspection qualification and with fracture toughness verified via direct measurement during site-specific stages.

602. The RP also confirmed that assumed QEDS values were based on both engineering practice and previous GDA experience, and that assumed QEDS values informed both the TJs and DTAs, which were undertaken in parallel for the GDA. The assumed QEDS values were appropriate for most cases. However, as a bounding approach had been used to minimise the number of TJs submitted in GDA (for example, for the RPV/PZR shell to shell welds, only the RPV TJ was provided), this could lead to smaller QEDS values and hence additional DSM margins for some HIC welds.

603. The RP also provided an example to illustrate how balance had been achieved in the reconciliation process. The example showed how internal discussions between its DTA and NDE teams had informed the selection of a QEDS value of 8mm for the MCL to SG inlet nozzle weld (Ref. 130). Notwithstanding these clarifications, the RP committed to reconsidering the assumed QEDS values to ensure that reasonable demands were being placed on NDE for the set of challenging HIC welds identified in GDA.

604. In RQ-UKHPR1000-1247, I also sought further clarification of how the claims and arguments in the RP's safety case were linked to the evidence in the specific DTAs and TJs for the HIC welds identified in GDA. The RP's response (Ref. 28) included the route from the claims, arguments and evidence in the safety case documentation using its hierarchy for the safety case i.e. from Chapter 17 of the PCSR (Ref. 79), via the Safety Case Methodology document (Ref. 26) and Reconciliation Strategy (Ref. 112) to the evidence in its DTAs and TJs along with specific reconciliation reports for HIC welds and features identified in GDA. The RP also committed to updating the "Introduction" or "Purpose" in its evidence documents post the response to RQ-UKHPR1000-1247 to clearly explain how the evidence supports its reconciliation claims including the links to the Safety Case Methodology and PCSR Chapter 17 submissions (Ref. 28).

605. The DTAs covered in RO-UKHPR1000-0006 (Ref. 49) pre-dated the response to RQ-UKHPR1000-1247 (Ref. 28), so I followed-up this point by undertaking a high-level review of a sample of the updated submissions for its AOFD. My sample included the three DTAs provided within RO-UKHPR1000-0006 (Ref. 122) (Ref. 124) (Ref. 162), and I found that none of the three documents reviewed had been amended to reflect the commitment made in RQ-UKHPR1000-1247 (Ref. 28). I performed a sample check of a more recently completed DTA (Ref. 163) and found also that the committed changes had not been applied. Finally, I sampled a reconciliation report (Ref. 164) and again found no reference to the commitment made in RQ-UKHPR1000-1247. I consider this to be another example of where the RP has failed to update its safety case to reflect commitments to improve traceability. I consider this shortfall to be captured within the scope of AF-UKHPR1000-0186 that I have already raised within Sub-section 4.1 of my report. For completion, I have also recorded this instance within my sample for Sub-section 4.8 below.

606. The Reconciliation Strategy (Ref. 112) was updated to reflect the clarifications provided in response to RQ-UKHPR1000-1247 (Ref. 28). In particular, to emphasise the need for joint discussions between the DTA, NDE and materials teams at the start of the reconciliation process. I was satisfied with the stronger emphasis placed on an integrated approach. However, in Step 3 of GDA, I had observed that there was no provision for further courses of action, if the third round of assessment fails to demonstrate reconciliation. I regard this as not as significant as ensuring that there is joint working at the start of the process and that there is provision for refinement of either the DTA or QEDS values. I therefore consider this matter to be a minor shortfall.

607. Overall, I was satisfied the RP's proposals clarified the aims of the reconciliation was to demonstrate an adequate DSM without recourse to unreasonable expectations on NDE and subject to an appropriate material testing strategy to verify the material property assumptions e.g. direct fracture toughness testing. I also gained confidence that the selection of the QEDS value in GDA had been informed by inter-disciplinary discussions with the RP providing evidence of these internal discussions. The RP also took steps to further clarify the 'golden thread' between its claim and arguments in the higher-level safety documentation to the evidence in the DTAs, TJs and reconciliation reports post the response to RQ-UKHPR1000-1247 (Ref. 28). The RP also committed to update its Reconciliation Strategy to reflect the clarifications provided in RQ-UKHPR1000-1247 (Ref. 28).

608. This notwithstanding, it is how the inputs are integrated in practice to the specific HIC welds identified in RO-UKHPR1000-0006 (Ref. 49), that would provide the basis for confidence in the adequacy of the RP's reconciliation process. This is considered next.

4.3.12.1 Reconciliation Example - RPV Flange Nozzle to Core Shell Weld (RO-UKHPR1000-0006)

609. The RP submitted its DTA for the RPV flange nozzle to core shell weld (Ref. 122). As noted in my comparative work through the RP's response to RQ-UKHPR1000-1549 (Ref. 28) along with discussion with my TSC, I was able to conclude that was good agreement on the limiting ELLDS and LFCG calculations for the DTA of the RPV inlet nozzle to core shell weld. A QEDS value of 10mm was therefore supportable for reconciliation from DTA.

610. In parallel, I reviewed the TJ provided for the RPV flange nozzle shell to core shell weld (Ref. 165). I raised RQ-UKHPR1000-0469 (Ref. 28) to progress my assessment. There were several points of clarification with the most significant, in terms of RO-UKHPR1000-0006 (Ref. 49), relating to the verification and RP's 'Informed Customer' capability, the apparent absence of an objective based approach to the inspections, and uncertainty in the basis of the defect descriptions and whether highly reliable

inspections could be achieved in practice. The RP's initial response did not fully address my questions. However, the points were subsequently addressed through successive revision of the TJ (Ref. 156). I was therefore able to conclude that there is a basis for confidence in the achievement of a QEDS value of 10mm for a 6:1 radial-circumferential defect during formal qualification at the site-specific stages (Sub-section 4.3.5).

611. The RP has used the lower bound fracture toughness values from the RCC-M and RSE-M codes. In addition, the RP committed to verifying the achievement of the fracture toughness values used in its AFOD by providing a strategy for the direct measurement of fracture toughness. This was provided in the RP's submission, 'Supplementary Toughness Test Requirements of Materials for HIC Components' (Ref. 115). I raised RQ-UKHPR1000-1186 and RQ-UKHPR1000-1415 (Ref. 28) to gain clarification of the scope and extent and verification of the fracture toughness values and to ensure that the RP's proposals would result in 'fully representative' testing (ONR SAPs paragraph 220, Ref. 2). Further details of my assessment are provided in Sub-section 4.3.7. In summary, I was content that the DTAs are founded on conservative lower bound fracture toughness values which will be subject to an adequate level of verification to ensure achievement in practice or to inform a view on the implication for the RP's avoidance of fracture demonstration should there be a shortfall in meeting the safety case assumptions.

612. I also reviewed the RP's reconciliation report which summarises the basis of the reconciliation (Ref. 157). I was satisfied that there was an auditable evidence trail from the reconciliation document to the evidence in the DTAs and TJs and that the RP had effectively applied its reconciliation process to the RPV flange nozzle to core shell weld. I also conclude that the key inputs from the DTA, the TJ and material properties provide a reasonable basis for the achievement of an adequate DSM and reconciliation for the purposes of the GDA.

4.3.12.2 Reconciliation Example - RPV Inlet Nozzle to Safe End Weld (RO-UKHPR1000-0006)

613. The RP submitted its DTA for the RPV inlet nozzle to safe end weld (Ref. 124). This assessment was undertaken by EDF™ Energy and was subject to EDF Energy verification and RP oversight. With TSC support, I undertook comparative DTA calculations. I subsequently raised RQ-UKHPR1000-1469 (Ref. 28) to further my understanding of the differences between the RP's and my independent calculations. The details of my comparative DTA work are presented in Sub-section 4.3.4. It will suffice to say that through the RP's response to RQ-UKHPR1000-1469 (Ref. 28) , the differences were explained satisfactorily. I therefore concluded that there was good agreement on the limiting ELLDS and LFCG calculations. A QEDS value of 5mm was therefore supportable for reconciliation from DTA.

614. In parallel I reviewed the TJ provided for the RPV inlet nozzle to safe end weld (Ref. 158). I raised RQ UK-HPR1000-0958 (Ref. 28) to progress my assessment. My key questions related to the capability of the NDE given the material structures at this dissimilar metal weld. The RP subsequently addressed these points via its response RQ-UKHPR1000-0958 (Ref. 28) and discussion at technical exchange meetings. I was therefore able to conclude that there is a basis for confidence in the achievement of a QEDS value of 5mm for a 6:1 radial-circumferential defect during formal qualification at the site-specific stages.

615. The position on the material properties, in particular the fracture toughness values and testing strategy, is as reported for the first reconciliation example. It will suffice to say that I was content that the DTAs are founded on conservative lower bound fracture toughness values which will be subject to an adequate level of verification to ensure

achievement in practice or to inform a view on the implications should there be a shortfall in meeting the safety case assumption.

616. I also reviewed the RP's reconciliation report which summarises the basis of the reconciliation. I was satisfied that the RP had effectively applied its reconciliation process to the RPV inlet nozzle to safe end weld and that there was an auditable evidence trail from the reconciliation document to the evidence in the DTAs and TJs. I also conclude that the key inputs from the DTA, the TJ and material properties provide a reasonable basis for the achievement of an adequate DSM and reconciliation for the purposes of the GDA for the RPV inlet nozzle to safe end weld.

4.3.13 Avoidance of Fracture Demonstrations & Reconciliation

617. In this section, I consider the RP's AOFD for a wider range of HIC components, and how for manufacturing defects considered in GDA, they have reconciled the inputs that underpin them. I present my key assessment considerations and judgements for the AOFD based on the evidence provided by the RP and informed by the discussions in the preceding sections relating to material properties, DTA, and the GDA TJs. These considerations inform an overall view on the veracity of the AOFD and are considered below for the purposes of GDA, with the emphasis placed on the achievement of the primary NDE objectives.

4.3.13.1 AOFD for the UK HPR1000

618. The RP's final position on reconciliation for the sample of HIC welds identified from its weld ranking is summarised in a series of reconciliation reports as summarised in Table 8 below:

Table 8: Results from AOFD for UK HPR1000 HICs

Component	Weld/AOFD	QEDS (mm)	QEDS + FCG (mm)	ELLDS (mm)	DSM	Ref.
RPV	Flange-nozzle shell to core shell weld[1]	10	13.2	28.2 (54.1)[1]	2.1 (4.1)[1]	(Ref. 157)
	Inlet Nozzle to Flange-Nozzle Shell Weld	10	14	31.4	2.2	(Ref. 164)
	Inlet Nozzle to Safe End Weld	5	5.4	20.3	3.7	(Ref. 159)
	Core Shell	10	17.6	59.8	3.4	(Ref. 162)
SG	Primary Nozzle Safe End to Primary Nozzle Buttering Weld	5	16.3	36.9	2.3	(Ref. 166)
	Tube-Sheet to Primary Head Weld[2]	10	14.7	54.2	3.7	(Ref. 167)
	Main Feeder Water Nozzle to Steam Drum Can No.2 Weld	10	10.9	25	2.3	(Ref. 168)
RCP	Casing Inlet Nozzle to Safe End Weld	5	5.6	23.3	4.2	(Ref. 169)
	Flywheel	10	10.1	89.6	8.9	(Ref. 170)
PZR	Man-Way Flange to Upper Cylindrical Shell Weld	10	11.5	36	3.1	(Ref. 171)

Component	Weld/AOFD	QEDS (mm)	QEDS + FCG (mm)	ELLDS (mm)	DSM	Ref.
	Upper Shell to Middle Shell Weld[3]	10	11.7	41	3.5	(Ref. 172)
MSL	MSIV Casing Crotch Repair Weld	10	13.5	28	2.1	(Ref. 173)
	Connection to the Penetration Flange Weld	5	5.	21.7	4.3	(Ref. 174)
MCL	Hot leg to SG inlet nozzle safe end Weld[4]	(8) 8	(16.5) 16.7	(34.7) 33.1	(2.1) 2.0	(Ref. 130)[5] (Ref. 175)

[1] With limited stable tearing applied, without DSM is 1.8 for transient T4-2-2
[2] Bounding case taken as Level D transient T4-7 (SSE+LOCA)
[3] Bounding case taken as external weld surface
[4] Bounding case taken as T3-6
[5] Bracketed values show results from initial DTA assessed by ONR. Final values are presented in the latest AOFD (Ref. 175)

619. I note that the RP claims a DSM of at least 2.0 for all HIC welds and features and with some locations having DSMs significantly in excess of 2.0. (Table 8). In principle, I welcome the availability of additional margins, but this is conditional on an adequate level of conservatism in the DTA and that the QEDS is underpinned by a basis for confidence in the achievement of qualified inspections with high reliability.

620. I also observe that several of the DSMs for Dissimilar Metal Welds (DMWs) are similar to those of the nozzle and shell type welds e.g. the SG inlet primary nozzle safe end to primary nozzle buttering weld and the MCL hot leg to SG inlet nozzle safe end weld. Whereas other DMWs appear to have DSMs significantly larger than non-DMWs, which is counter intuitive. The DMWs benefit from the smaller QEDS values but are potentially subject to higher fatigue crack growth as evidenced for the SG and MCL. Thus, the position for the individual DMWs is governed by the particular inputs e.g. the QEDS and the amount of cycling with locations such as the SG experiencing more temperature changes and hence load cycling that the RPV and RCP. I have therefore focused my considerations on those AOFDs with apparently relatively low DSM values circa ≤ 3.0 and where there is a need to take account of any concerns with the veracity of the inputs either from NDE or the DTA perspective. The RPV flange nozzle to core shell weld; the RPV inlet nozzle to shell weld; the SG Inlet Primary Nozzle Safe End to Primary Nozzle Buttering Weld; SG main feedwater nozzle; MSIV casing crotch (repair weld); and MCL hot leg to SG inlet nozzle safe end weld therefore warrant further consideration (Table 8).

621. In my assessment of the RP's response to RO-UKHPR1000-0006 (Ref. 49), the RP provided the AOFD for the RPV flange nozzle to core shell weld as an example of the application of the reconciliation process for a shell to shell HIC weld. The RP demonstrated a DSM of at least 2.1 for a QEDS of 10mm which was confirmed through independent calculations (Sub-section 4.3.4.2). In addition, my sensitivity studies indicated a marginal reduction in the DSM to about 1.85 with a 25% increase in yield stress in the Kr calculation. I therefore judge that the small reduction in DSM in this case is indicative of adequate defect tolerance, especially when the DSM is judged in the context of the additional conservative assumptions identified in the RP's DTA methodology (Ref. 112) (Ref. 110). I also note that there is a sound basis for confidence that a QEDS of 10mm (the primary inspection objective for GDA) is achievable through formal inspection qualification (Sub-section 4.3.5). I therefore consider that the RP has provided adequate evidence to underpin the AOFD claim for the RPV flange nozzle to core shell weld.

622. For the RPV inlet nozzle to shell weld (a nozzle weld) the RP claims a DSM of 2.2 with a QEDS of 10mm, (Table 8). The RP's DTA recognised the revised guidance in R6 to use a residual stress value of 0.3 Sy and incorporated this value appropriately in the Kr parameter. I also note that there is a sound basis for confidence that a QEDS of 10mm (the primary inspection objective for GDA) is achievable through formal inspection qualification (Sub-section 4.3.5). Thus, there remains a basis for confidence that an adequate AOFD case can be made.

623. A DSM of 2.3 and QEDS of 5 mm is claimed for SG inlet nozzle to safe end to primary nozzle buttering Weld (a DMW), (Table 8). As this weld is a DMW it is not subject to PWHT. However, as the RP's assessment pre-dated my query relating to the use of an appropriate residual stress value (mean) in the Kr parameter, the position for the SG Inlet Primary Nozzle Safe End is informed by my sensitivity study undertaken for the RPV flange nozzle to core shell weld (paragraph 443-448). Accordingly, I expect a marginal reduction in the DSM with a higher mean residual stress value incorporated into the Kr parameter. In addition, as the SG Inlet Primary Nozzle Safe End to Primary Nozzle Buttering Weld is grouped with the TJ presented for the RPV Inlet Nozzle to Safe End Weld, there is a sound basis for confidence that a QEDS of 5mm (the primary inspection objective for GDA) is achievable through formal inspection qualification (paragraphs 551-564). Thus, there remains a basis for confidence that an adequate AOFD case can be made.

624. The AOFD for the SG main feedwater nozzle weld is based on a claimed DSM of 2.3 for a QEDS of 10mm (Table 8). As the RP's assessment pre-dated my query relating to the use of an appropriate residual stress value (mean) in the Kr parameter, the position for the SG main feedwater nozzle weld is informed by my sensitivity study undertaken for the RPV flange nozzle to core shell weld (Paragraph 443-448). Accordingly, I expect a marginal reduction in the DSM with a higher mean residual stress value incorporated into the Kr parameter. This was also confirmed by the RP through a sensitivity study in its response to RQ-UKHPR1000-1604 (Ref. 28).

625. A further point from the DTA relates to the use of intra-transient (within transient) rather than the usual and conservative inter-transient (between transient) pairing in the FCG calculations. The RP responded constructively to this question and undertook a sensitivity study to establish the significance for the SG main feedwater nozzle weld. The results showed a small increase in the FCG from < 1mm to < 2.9 mm for a 6:1 internal longitudinal to the weld (radial-circumferential defect), giving a DSM of marginally below 2.0, for a conservative DTA (Ref. 127).

626. With regard to the NDE input, the QEDS claim of 10mm for the SG main feedwater nozzle is grouped with the RPV inlet nozzle to flange-nozzle shell weld, there is a sound basis for confidence that a QEDS of 10mm (the primary inspection objective for GDA) is achievable through formal inspection qualification (paragraphs 571-581). Thus, overall and taking account of the conservatism in the DTA there remains a basis for confidence that an adequate AOFD case can be made.

627. For the main steam isolation valve a DSM of 2.1 is claimed for a crotch corner weld repair with a QEDS of 10mm (Table 8). I had targeted this feature for sample comparative calculations, but the data to facilitate independent DTA calculations arrived late within the GDA and so I was unable to progress this work with my TSC. I draw some confidence from the fact that the RP's DTA work for the MSIV weld repair was sub-contracted to a UK contractor with experience in the application of the R6 procedure. In addition, I am aware that the assumed weld repair is positioned at a stress raising feature and with sensitivity studies invoked to establish the worst defect orientation. Nonetheless, I am aware that DTA for casting repairs may be challenging, the material properties of the casting and repair materials will need to be verified and the overall position needs to take into account my level of confidence in the achievement of inspection qualification. In view of these considerations and to ensure

that any residual risks associated with the AOFD for the MSIV are mitigated, I have raised AF-UKHPR1000-0197 below.

628. A DSM of 2.0 for a QEDS of 8mm is claimed for the MCL hot leg to SG inlet nozzle safe end weld (Table 8). However, the RP's assessment pre-dated my query relating to the use of an appropriate residual stress value (mean) in the Kr parameter, the position for the MCL hot leg to SG inlet nozzle safe end weld is informed by my sensitivity study undertaken for the RPV flange nozzle to core shell weld (paragraph 443-448). Accordingly, I expect a marginal reduction in the DSM with a higher mean residual stress value incorporated into the Kr parameter. In RQ-UKHPR1000-1552 (Ref. 28), the RP also committed to addressing the point during the site-specific stages (Ref. 28). From the NDE side, I note that there is a sound basis for confidence that a QEDS of 8mm (the primary inspection objective for GDA) is achievable through formal inspection qualification (paragraphs 515-523). Thus overall, and taking cognisance of the conservatism in the DTA, I judge that there remains a basis for confidence that an adequate AOFD case can be made.

629. The overall position on reconciliation is therefore that for the majority of welds and features addressed in the GDA there is a sound basis for confidence that adequate AOFD cases can be made. I have identified that there are some risks to address for the MSIV crotch corner weld repair, and to mitigate them, I have raised an assessment finding for the licensee to progress.

630. This notwithstanding, there are two further points to discuss, firstly, the implications of the observation that the RP may not have always invoked the limiting conditions for plastic collapse in its Lr calculations (Sub-section 4.3.4.2); and secondly, though a secondary objective for the GDA, the position on the tolerance and reconciliation for radial-axial defects.

631. As mentioned above late in the GDA, I was made aware of an outstanding point from the review of the PZR man-way flange upper cylindrical shell weld. Specifically, this relates to the RP not following the R6 procedure with respect to the identification of the limiting Lr conditions governing plastic collapse. Notably, collapse may be potentially limited by out of plane stresses and therefore out of plane collapse, Section II.4 and Section IV.1 of R6 (Ref. 12) cautions:

632. "Special care is needed when evaluating the limit load of circumferential defects in pressurised piping or vessels. In the absence of a defect, the hoop stresses in a closed-end pipe under pressure are typically twice the value of the axial stresses. For shallow defects the hoop stresses have a dominant effect in the limit load. Only, for deep defects, greater than about half the wall thickness, is it acceptable to estimate plastic collapse from the net section axial stress normal to the defect plane".

633. In this document, structural collapse is considered to be governed by failure of the section containing the flaw. The limit load calculated by the methods described below and in Section II.4 should then correspond to the spread of plasticity in the flawed section. The solutions detailed in Section IV.1 are of this type. The possibility of plastic collapse elsewhere in the structure should be separately investigated. The use of a limit load corresponding to such a remote failure mechanism in an assessment of a section containing the flaw may be overly conservative.'

634. In summary there is a need to ensure that the limiting conditions for plastic collapse are identified and to check that plastic collapse does not occur elsewhere from the flaw (which may or may not be influenced by the presence of the flaw).

635. An RP response was not received within the timescales of the GDA and so I have had to invoke a judgement on the significance of this point. For the PZR man-way flange upper cylindrical shell weld which, I judge that an adequate AOFD is likely to be

maintained because the DSM is relatively large at 3.1, and so there is contingency to address any uncertainty in the ELLDS.

636. Similarly, and based on the same rationale, I expect several of the AOFD in Table 8 to retain high DSMs irrespective of the potential for a higher Lr and hence smaller ELLDS. In addition, several of the ELLDS will remain bounded by plastic collapse normal to the plane of the defect. However, I recognise that there may be situations where the DSMs are relatively small and are also affected by a higher Lr if collapse is governed by out of plane stresses.

637. This potential shortfall does not affect the RPV welds in RO-UKHPR1000-0006 (Ref. 49), because my independent calculations confirmed the RP's results took account of out of plane collapse in the Lr parameters. This may also be the case for several of the other welds assessed in GDA. Additionally, if the component passed the design assessment for primary stress, then the uncracked Lr solution cannot exceed 0.67. An Lr value of circa 0.67 is in a relatively flat portion of the FAD, and so a significant reduction in margin would not be expected.

638. Overall and taking account of the points above it is my view that the RP has demonstrated adequate AOFDs for the majority of the HIC welds selected for the GDA. Nonetheless, it is important that any HIC welds with low DSMs which may warrant re-calculation of the ELLDS to confirm the adequacy of the DSM are identified and assessed as a matter of priority during the site-specific stages.

639. I have already raised AFs to further improve the methodologies that underpin the DTAs and to ensure robustness in the verification arrangements. I raise a further assessment finding as means for the licensee to mitigate any residual risks associated with the AOFDs provided in GDA and to ensure that any risks for the wider range of AOFD that will underpin the highest reliability claims for the UK HPR1000 are also reduced.

AF-UKHPR1000-0197 – The licensee shall, as part of detailed design, justify the position for any high integrity component welds with low defect size margins identified during GDA to confirm the end of life limiting defect size, and underpin the avoidance of fracture demonstrations. This should include, but not be limited to, the main steam isolation valve weld repair.

640. As highlighted in the NDE discussion, defects transverse to the welding direction (usually designated radial-axial in the DTAs) were a secondary objective for the development of confidence in the NDE capability for the purposes of GDA with the priority given to the most likely (radial-circumferential) manufacturing defects. In the DTAs, axial defects have tended to feature in the RP's sensitivity studies, but often there was little discussion of the results or the implications where safety margins are apparently low. Responses from the RP provided more consideration of axial defects and the implications, including further refinement at some of the assessment locations.

641. The additional information provided indicates that, when considering the relevant defect aspect ratios (acknowledging that for axial defects smaller defect length to depth ratios are more readily justified), it is possible to demonstrate that acceptable DSMs (>2) can be determined for all components. Whilst updated reports covering axial defects in more detail were not reviewed within the GDA timescale, the RP has shown that they are considering axial defects. Ultimately it will be for the licensee to demonstrate that its AOFDs are underpinned by adequate DSMs and reconciliations for axial defects. AF-UKHPR1000-0197 will assist with this activity and the licensee should also take cognisance of the risks identified from an NDE perspective (Sub-section 4.3.5).

4.3.14 AOFD Strengths

642. The RP's weld ranking process provided a reasonable and pragmatic means of selecting the HIC welds for detailed assessment during GDA.

643. The RP has developed an adequate material testing strategy that will support a robust avoidance of fracture demonstration. The RP's approach encompasses 'above code' demands on fracture toughness testing, built on conservative, lower bound values specified in relevant nuclear design and construction codes.

644. The RP responded proactively and constructively to the RQs raised on aspects of the AOFD throughout the GDA, with the responses often providing useful and additional evidence that could improve the structural integrity case.

645. The RP has developed its understanding and proficiency in the use of the R6 defect assessment procedure over the course of the GDA. This was evident from the results of my independent comparative calculations, which provides confidence that the RP has an understanding and, in most situations, can correctly apply the R6 procedure.

646. The RP's strategy for qualifying the end of manufacture NDE of HICs is a well-developed document that describes the principal activities for inspection qualification along with the roles and responsibilities of the relevant organisations

647. I sampled several of the GDA technical justifications, and through discussions with the RP, and responses to my RQs, in the majority of cases I was satisfied that the RP had demonstrated an understanding of the concept of high-reliability, objective-based NDE in support of its avoidance of fracture case. In my opinion, the GDA technical justifications had presented an appropriate level of evidence, based largely on theoretical arguments, with adequate margins for detection.

648. The RP clarified the aims of the reconciliation for its AOFD, and in principle, adequate DSMs could be developed without recourse to unreasonable demands being placed on either the NDE or material properties.

649. The RP provided suitable and sufficient evidence to close RO-UKHPR1000-0006 (Ref. 49) based on:

- confidence that the selection of the QEDS values in GDA had been informed by inter-disciplinary discussions within the RP's structural integrity discipline;
- there was an auditable evidence trail from the reconciliation document to the evidence in the DTAs and TJs; and
- the RP's demonstration of reconciliation for several HIC welds in the RPV.

4.3.15 AOFD Outcomes

650. The RP should ensure that the safety case is reviewed to demonstrate clear traceability to the evidence showing how the DTAs use conservative material properties, underpinned by fully representative sampling during the manufacture of HICs.

651. I was satisfied that there was an adequate level of conservatism in the DTAs to inform a judgement on the RP's reconciliation process, in particular, to progress RO-UKHPR1000-0006 (Ref. 49). However, any proposals to introduce further reductions in conservatism in the DTAs will need to be justified and controlled.

652. The RP chose a complex process to establish the limiting ELLDS values with the output ultimately dependent on the confidence in the transient definitions. The value and efficiency of the application of this process when balanced against the demanding timescales of licencing warrants review.

653. Application of the R6 defect assessment procedure is less prescriptive than design codes and requires interpretation and judgement to establish a valid result with the adequate conservatism. The RP improved its understanding and proficiency in using the R6 defect assessment procedure over the course of the GDA. However, I identified several generic points relating to the RP's application of the R6 defect assessment procedure. Several of these generic points could and should have been addressed in the development of the DTA methodology document. I have raised six assessment findings related to the development and implementation of the RP's DTA methodology (AF-UKHPR1000-0192 to AF-UKHPR1000-0197). These specifically relate to the consideration of conservatism and consistent use of input data for DTAs, the scope of welds considered during detailed design stages and the development of a robust verification process.

654. I judge that the assumed defect characteristics were appropriate for the purposes of GDA. Nonetheless, it is important that the licensee justifies the assumed manufacturing defect descriptions in inspection specifications for high integrity components. This should include, but not be limited to, the use of suitable expert elicitation. I have raised an assessment finding (AF-UKHPR1000-0196) to ensure this is addressed during the site-specific stages.

655. Whilst I conclude that for the majority of the HIC welds sampled, the GDA technical justifications provide a basis for confidence in the achievement of inspection qualification and hence objective-based high reliability NDE, there are a few areas that warrant either further development work or justification.

4.3.16 AOFD Conclusion

656. Overall, I am content that the RP has provided adequate AOFDs for the majority of welds and features for the purposes of the GDA and there is a sound basis for confidence that adequate AOFD cases can be made. I consider that the RP applied, in general, conservative methods in its DTAs and appropriate methods in the development of its GDA technical justifications to provide a basis for confidence in the future qualification of manufacturing inspections. The RP also developed and committed to implementing a materials testing strategy to underpin its AOFD.

657. I identified that there are some specific risks to address for the MSIV crotch corner weld repair, and to mitigate them, I have raised an assessment finding for the licensee to progress.

658. I have also raised assessment findings as a means for the licensee to mitigate any residual risks associated with the AOFDs provided in GDA and to ensure that any risks for the wider range of AOFD that will underpin the highest reliability claims for the UK HPR1000 are also reduced.

4.4 Structural Integrity Provisions, Design Codes and Standards

659. The design and construction of a nuclear power plant is a complex process and as such there exists many risks where inadequate components could be used. The main means of countering this is to use established nuclear design codes and standards which are internationally recognised (Ref. 2, ECS.2 and ECS.3).

660. Within the UK regulatory environment just following a recognised code or standard is not sufficient. ONR expects that the RP can apply the code intelligently, such that they are able to understand how the code or standard provisions protect against certain failure modes or risks.

661. I have therefore engaged with the RP on the subject of the use of applicable codes and standards and other proposals for underpinning structural integrity. The objectives

of my engagements were to gain confidence that the RP understands the code and how its use can control the risks inherent with the design of nuclear pressure equipment.

662. My sampling of the RP's provisions for SI included:

- Design Codes and Standards.
- Combinations of Codes and Standards.
- Additional SI Provisions.

4.4.1 Use of Design Codes and Standards

4.4.1.1 Suitability of Codes and Standards

663. The classification of SSC reflects the importance to nuclear safety and functional reliability, which then links the plant safety case to the engineering provisions, via the allocation of appropriate codes and standards (usually via an engineering schedule). ONR SAP ECS.3 (Ref. 2) states that SSCs that are important to safety should be designed, manufactured, constructed, installed, commissioned quality assured, maintained, tested and inspected to the appropriate codes and standards.

664. In Step 2 of GDA, I sought to establish that the RP is proposing adequate design and construction codes commensurate with the importance of the SSC to nuclear safety. Indeed, the selection and implementation of appropriate design, manufacturing standards and inspection provisions is central to a demonstration the risks of failure are reduced to ALARP. I raised RQ-UKHPR1000-0030 to establish the extent to which French, US and Chinese regulatory standards are intended to inform the selection of relevant codes and standards for the UK HPR1000. The RP's response to RQ-UKHPR1000-0030 (Ref. 28) explained that the selection of codes and standards for SSC is based on IAEA SSG -30 (Ref. 9) and is informed by both the safety function class and design provision (or barrier class). In addition, standard Class 1 and 2 SSCs are designed and constructed in accordance with nuclear specific codes and standards. For standard Class 3 SSCs nuclear or appropriate non-nuclear codes and standards may be used.

665. Nuclear pressure vessels and piping are designed to internationally accepted design codes and the RP has designed Fangchenggang-3 to the French nuclear design code, RCC-M (Ref. 14). The use of Chinese design codes and standards is limited to structures and components, which are non-safety classified i.e. they do not deliver nuclear safety functions. I noted the RP's position, but needed to establish whether for standard Class 3, the intent is to use nuclear codes or a combination of nuclear and non-nuclear codes with supplements. Notably, if non-nuclear codes with supplements are proposed then application of the nuclear exclusion under the Pressure Equipment Regulations will be an important consideration.

666. The PCSR, (Ref. 3) along with the SI classification document (Ref. 64), and the Suitability of Codes and Standards document (Ref. 176) expand on the selection of codes and standards for the UK HPR1000 SSC within the SI Discipline. The French RCC-M code is proposed for the majority of SSCs with the allocation of the RCC-M classes 1-3 governed by the safety class, M class and SI class. The design requirements set by the RCC-M code were reviewed by ONR as part of the UK EPR™ GDA. ONR concluded that the design provisions were broadly the same as those for ASME III on a class by class basis and are judged to be generally acceptable for nuclear pressure systems (Ref. 177). The design and construction provisions of RCC-M have since been implemented in the manufacture of the major vessels and piping for the UK EPR™ at Hinkley Point C (Ref. 178).

667. I was therefore broadly content with the proposed use of the RCC-M code for the majority of the pressure vessels and piping in the UK HPR1000 and with the use of a graded approach to design and construction, with the SI provisions proportionate to the importance to nuclear safety. I was also aware that the RP was proposing ta combination of codes for the SG and RCP; with design and manufacture to the US ASME III code and ISI to the French RSE-M code. These proposals are the subject of further review (see Sub-section 4.4.2). My assessment therefore focussed on the application and demonstration of design code compliance (including the RP's consideration of the significance of low margins against code limits); code versions; the design transient specification; and informed, by the experience from the UK EPR™ GDA, the consideration of specific areas where further work may be necessary to meet ONR's expectations.

Demonstration of Code Compliance

668. For GDA, ONR is seeking an adequate basis of confidence in design code compliance and that the fatigue usage calculation methodologies used by the RP are demonstrably conservative, with adequate margins, to underpin the planned 60-year design life for the UK HPR1000. Indeed, subject to an appropriate SI classification, code compliance is the basis for the demonstration that the SSCs of the UK HPR1000 are capable of meeting UK safety expectations, with the exception of components classified by the RP as HIC (see Sub-section 4.3).

669. In forming a view of code compliance, confidence is gained from the fact that the UK HPR1000 design has received approval in China. he country of origin. However, the regulatory approaches and expectations (including laws) differ between UK and the country of origin and hence the expectations for the demonstration of code compliance and the safety case may differ, which means that although confidence can be drawn from the achievement of design code compliance under other countries jurisdictions, there is a need to take an independent view.

670. With the support of my TSC, I sampled the RP's demonstration of design code compliance (Ref. 18). I agreed the following hierarchy of comment categories with my TSC:

- Category 1: Significant non-compliance, significant non-conservatism, significant safety issue, that would jeopardise the conclusions from any assessment.
- Category 2: Assessment routes or techniques not meeting accepted practice.
- Category 3: Major comments and points of clarification that could become Category 1 or 2 comments.
- Category 4: Minor comments and points of clarification.

671. I reviewed TSC comments before submitting them to the RP via the RQ process. For GDA, it was my expectation that all Category 1 and 2 comments need to be addressed. Category 3 comments and above require a response within GDA, with a potential to result in an Assessment Finding, if a basis for confidence to resolve within GDA is not established. Category 4 comments are unlikely to lead to assessment findings but should be noted and carried forward by the licensee as part of normal business.

Initial Review Work

672. The RP provided a series of dimensioning reports and general mechanical analysis reports for the primary circuit components to demonstrate design code compliance for design and operating conditions, usually in terms of minimum required thicknesses and stress or fatigue usage factors along with the margins against allowable limits. The dimensioning reports provide summaries of the design by rule calculations (sizing

calculations/scantling calculations), whereas the general reports of mechanical analysis cover the stress and fatigue calculations.

673. Initially, a high-level sampling review of the RP's design calculations for the RPV, PZR, SG, MCL and RCP (HIC components) along with the SL and RVI (non-HIC) was undertaken. A summary of the outcome of the initial code compliance review work is provided below with more details available in (Ref. 18).

674. The documents reviewed, and their associated RQs, are presented in Table 9 below:

Table 9: List of RQs Raised to Review Design Code Compliance.

Identified Component	RQ Raised (Ref. 28)
RPV Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1149 RQ-UKHPR1000-1467
Pressuriser Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1151
Steam Generator Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1152, RQ-UKHPR1000-1265
Main Coolant Line Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1148
Reactor Coolant Pump Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1248
Surge Line Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1303
Reactor Vessel Internals Dimensioning Report and General Report of Mechanical Analysis	RQ-UKHPR1000-1150

675. In addition, the sample covered a range of components including major vessels and piping. The section below mainly focuses on compliance with code stress/stress intensity limits with the fatigue aspects covered in Sub-section 4.4.1.2 of my report.

676. Several comments were raised against each of the RP's submissions. The majority of the RP's responses addressed the queries raised, initially through RQ responses and then by updating the relevant documentation. However, a number of responses did not fully address the comments. The most significant points not fully addressed in the initial review work are discussed below.

Pressuriser

677. In the response to RQ-UKHPR1000-1151 (Ref. 28) the following points were not fully addressed:

- The measure tap opening does not meet the design by rule requirements of RCC-M. The RP's response shows that the opening meets the requirements based on the design by analysis requirements.
- The elastic-plastic shakedown analysis based on ASME III NB-3228.4 has been performed in lieu of checks against the requirements of RCC-M B 3234.2 and RCC-M B 3234.7. The applicable design code for the pressuriser is RCC-M 2007, and an adequate justification was not provided for using the ASME BPVC instead of the applicable design code.
- Checks against the requirements of RCC-M B 3234.2 were not provided for the cladding-to-nozzle weld. The response states that these checks are presented

in the detailed reports but a summary was expected in the General Report of Mechanical Analysis.

Steam Generator

678. The absence of an explanation of assumptions and justification of factors affecting the stress analysis to show conservatism.

- Tri-axial stress checks as required by Clause NB-3227.4 of ASME III have not been included within the evaluation criteria.
- Incomplete summaries to inform a review of the analysis work done or how the code assessments were undertaken.

679. Several of the comments raised from the initial review of the SG were subsequently closed out via the receipt of additional information during the subsequent detailed review (see below). It should also be noted that a number of the points raised were generic and these are discussed in more detail below.

680. The most significant non-generic comment not fully resolved for the SG relates to the exclusion of the tri-axial stress checks. The RP's response states 'Tri-axial stress limit (NB-3227.4) is a special stress limit and only applicable to special cases. The tri-axial stress limit is not considered in this report since it is bounded by the basic stress limits in NB-3220 for the components assessed in this report.'

681. However, NB-3222 for Level A Service Limits states "Level A Service Limits must be satisfied for the Service Conditions [NCA-2142.4(b)(1)] for which these limits are designated in the Design Specifications and are the four limits of this paragraph and the Special Stress Limits of NB-3227". The check relates to local failure, as a state of high tri-axiality can cause the material to act in a brittle manner. Since the stress intensity limits used in the codes are based upon the maximum shear stress criterion, there is no limit on the hydrostatic component of the stress. Therefore, the special limit in NB-3227.4 on the algebraic sum of the three principal stresses is included.

682. The RP also indicated that the 'SG General Report of Mechanical Analysis' (Ref. 179) would cover the role of the dimensioning report. However, upon review it was found that the cited reference did not evaluate minimum wall thickness, reinforcement, closure sizing or flange design in accordance with ASME Sub-section NB or RCC-M Sub-section B. RQ-UKHPR1000-0916 (Ref. 28) was raised to make the RP aware of ONR's expectations in relation to demonstrating code compliance for the SG for the GDA.

683. The RP's response to RQ-UKHPR1000-0916 (Ref. 28) referenced a 'SG Base Design Report' (Ref. 180) which undertakes shell minimum thickness calculations, the results of which are summarised in the RP's response. Hence, the issue of minimum wall thickness and reinforcement were addressed. Regarding closures and flanges, the RP's response stated that the studs and covers were qualified using classical methods with references to a calculation document which was not available for the GDA. The topic of the RP's supply of design information relating to the SGs is discussed in below.

Main Coolant Loop

684. Several comments raised by RQ-UKHPR1000-1148 (Ref. 28) that were not resolved within the initial review were closed out via the receipt of additional information during the subsequent detailed review. It should also be noted that a number of the issues identified and raised as comments were also found to occur in other reviews. The most significant non-generic comments that were not resolved relate to whether RGP has been followed in the application of the design code:

- Interaction of nozzles, no formal checks are presented in (Ref. 181) (Ref. 182) to confirm that nozzles are sufficiently widely spaced to avoid interacting. It was therefore not clear whether any formal checks have been undertaken, or if the lack of interaction is based purely on judgement.
- The thermal stress ratchet checks from RCC-M B3653.7, which are required in addition to B3653.3 or B3653.5, have not been included.

685. This notwithstanding, it is judged that the risk to the design code compliance of the component is small based on previous experience of similar piping arrangements i.e. by inspection no nozzles appear to be positioned immediately adjacent to each other, and whilst checks for thermal stress ratcheting have not been reported, it is known that it is very unusual to fail these checks if the conventional shakedown checks are passed, which is the case.

Reactor Coolant Pump

686. RQ-UKHPR1000-1466 (Ref. 28) was issued, which raised several comments the majority of which were addressed by the RP. RQ-UKHPR1000-1601 (Ref. 28) was issued as a follow-up RQ. In summary, five comments were outstanding following the RP's response to the initial RQ. However, none of these comments are expected to represent a significant technical or design risk, as they do not relate to code compliance in itself but rather how the evidence is referred to in other documentation. For example, several calculations were included that did not appear to relate to the design by rule requirements of ASME BPVC or RCC-M. These calculations were more appropriate to showing compliance with the design by analysis requirements of ASME BPVC. Thus, it was unclear how these additional calculations fit into the overall safety case structure.

Surge Line

687. The documentation presented for the SL was similar to that for the MCL, so the comments raised in RQ-UKHPR1000-1303 (Ref. 28) effectively in addition to those raised against the MCL. Key points included:

- There was no evidence provided that the criteria for not undertaking fracture assessments were met.
- For each of the assessed regions, a summary table of what appeared be bounding results was presented, but there were limited details of the analysis work done or of how the code assessments were undertaken.

688. The RP's responses typically referred to the detailed stress and fatigue reports (as was found to be the case for other components). However, the more detailed reviews on other components found it difficult to trace the evidence and, in some cases, the lower-level documents did not always contain the claimed evidence, which is not to say that such evidence does not exist, it just could not be traced.

Reactor Vessel Internals

689. A large number of comments were raised against the dimensioning and mechanical analysis reports the majority of which were addressed in the response to RQ-UKHPR1000-1150 (Ref. 28). As a number of the points related to generic issues, these were captured in a consolidated RQ. The more significant non-generic comments are summarised below:

- Buckling of the core barrel has been assessed using published solutions as opposed to codified methods. Whilst the comment was addressed, as the buckling capacity has been evaluated in accordance with RCC-M in the RP's response, further justification is needed if non-codified routes are employed.

- The design stress for the Roto-lock insert was queried. The RP's response states that 'The basic allowable stress value S = min (1/5Rm, 1/4Re) = 163.75MPa according RCC-M ANNEX ZIII 320.' It is agreed that the basic allowable stress of 164MPa is appropriate, but the 'Design Specification of RVI' (Ref. 183) quotes a value of 267.5MPa. The actual stress applied to the component is 211MPa, and so it was not clear whether there was an error, or a further explanation of the methodology was needed. However, these are Class 2 components used for lifting of the upper and lower internals so that the risk to the design was judged small.
- ASME III Appendix A has been used as a method for determining stresses within the Lower Support Plate (LSP) since this was the only code-based method available for the RVI which is a code Class 1 SSC in accordance with RCC-M. ASME III A-8131(b) states that the actual values of the stress intensities in the perforated plate are determined by applying multiplying factors to the nominal stresses computed for the equivalent solid plate. The RP's response states "the ASME III A-8142.1 is suitable for hole-plate with tubes, but there is no tube in LSP. Therefore, it does not apply to LSP calculation". Therefore, in the application of ASME III Appendix A to the LSP, it does not appear that the design code is being followed. This is mitigated by the fact that the LSP is also assessed within the 'RVI Dimension Report' (Ref. 184) using Finite Element Analysis (FEA) and found to be compliant with the necessary code stress requirements.

Overview of Initial Review Work

690. Overall, the initial review work highlighted some potential difficulties relating to referencing, source data, lack of evidence, discussion on small margins or claimed conservatisms. These subsequently informed the more detailed review work and the overall conclusions drawn with respect to design code compliance (Sub-section 4.4.1.1).

Detailed Review Work

691. Informed by the initial review work, and taking cognisance of areas of interest identified by the ONR during Steps 2 and 3 of GDA, e.g. the assessment of dissimilar metal welds, FEA methods, and compliance with design by rule requirements, several regions and topics were identified for more detailed review. These detailed reviews focus on the RPV, but also consider the SG and MCL.

692. The documents reviewed, the rationale and their associated RQs, are listed below in Table10, with details available in (Ref. 18):

Table 10 – List of RQs for Detailed Design Code Review.

Identified Component	RQ Raised (Ref. 28)
RPV: Inlet Nozzle, Closure Head and Vessel Flange, CRD Adapter Sleeve, Sealing Analysis	RQ-UKHPR1000-1196 RQ-UKHPR1000-1545 RQ-UKHPR1000-1607 RQ-UKHPR1000-1608 RQ-UKHPR1000-1643 RQ-UKHPR1000-1735 RQ-UKHPR1000-1736
Steam Generator: Base Design Report and Transient Analysis Report	RQ-UKHPR1000-1302 RQ-UKHPR1000-1694
MCL: Surge Line Nozzle and Safety Injection Inclined Nozzle	RQ-UKHPR1000-1308 RQ-UKHPR1000-1547 RQ-UKHPR1000-1644 RQ-UKHPR1000-1732

RPV Inlet Nozzle

693. The RPV Inlet Nozzle was selected for a detailed review for the following reasons:

- It represents the major penetration in the RPV and is a significant stress-raising feature.
- It includes a dissimilar metal weld, which requires special consideration in the strength and fatigue assessments.
- The margins against allowable stresses and stress ranges are small (a minimum of 1.4% for the core shell at Path 117 under a Category III condition).
- It includes an RCC-M fast fracture analysis, the results of which were not presented in the Mechanical Analysis report.

694. RQ-UKHPR1000-1545 (Ref. 28) was raised which covered the detailed review of the stress analysis, fatigue analysis and fast fracture analyses (Ref. 185) (Ref. 186) (Ref. 187).

695. Similar comments were raised as in the initial reviews as the reports would benefit from more description and references to supporting data; however, the level of detail and references contained in the responses to the RQs provide confidence in the quality of the FEA modelling and post processing.

696. The most significant comment related to the assessment of the dissimilar metal weld, FEA model refinement and the extraction of stresses to ensure the limiting locations have been assessed. Detailed responses were provided by the RP which described how the dissimilar metal weld region was modelled to provide a conservative estimate of stresses, supported by sensitivity studies, and the subsequent design code assessments. Further detailed responses from the RP were provided to demonstrate the adequacy of the FEA model mesh refinement and supporting sensitivity studies, and the checks conducted to ensure that the limiting locations have been identified.

697. In summary, the RP provided adequate responses to the majority of the questions raised in RQ-UKHPR1000-1545 (Ref. 28). The outstanding comments were not expected to present a significant technical risk and any significant points were followed up as generic points (Ref. 18).

RPV Closure Head and Vessel Flange

698. The region of the closure head (CH), vessel flange, flange-nozzle shell and studs were selected for detailed review for the following reasons:

- The sealing of the closure head to the vessel body is one of the more complex regions of the RPV to model using finite element analysis due to the interaction and load transfer between several components.
- There is limited detail in the Mechanical Analysis report regarding the assessment of some of the components, such as the C-sealing ring and studs, and it is not clear whether the nuts and washers have been considered.
- For regions where there are low margins, such as the studs and head flange, it is important to understand the level of conservatism in the assessments.
- It includes an RCC-M fast fracture analysis, the results of which were not presented in the Mechanical Analysis report.

699. RQ-UKHPR1000-1608 (Ref. 28) was issued, with the RP providing satisfactory responses to the majority of comments. A further RQ, RQ-UKHPR1000-1736 (Ref. 28) was issued to capture the follow-up points. The more significant comments (category 3) that were not adequately closed were as follows:

- Details of the tri-axial stresses were not available in the 'Stress Analysis of Closure Head and Vessel Flange of Reactor Pressure Vessel report' (Ref. 188).
- The code checks covering progressive deformation is omitted from 'Fatigue Analysis of Closure Head & Vessel Flange of Reactor Pressure Vessel' (Ref. 189) on the basis that the risk is low, but no evidence was provided to support this assertion.
- The presence of the closure head penetrations has been approximated using "smeared" material properties. The method used to derive the equivalent material properties is appropriate, but the source only presents the method as applicable to structural material properties. Further evidence is required to demonstrate that this method is applicable for thermal material properties (Ref. 188) (Ref. 189).

700. In each of these cases it would be expected that these results would be readily available from the FEA models to provide clear evidence of code compliance and to strengthen the structural integrity case.

701. A detailed review of the sealing of the RPV CH to the vessel body was also conducted as:

- The sealing arrangement is one of the more complex regions of the RPV to model using FEA analysis due to the interaction/load transfer between several components.
- There is limited detail in the Mechanical Analysis report regarding the assessment of some of the components, such as the C-sealing ring and studs, and it was not clear whether the nuts and washers have been considered.

702. RQ-UKHPR1000-1607 (Ref. 28) was issued, with the RP's response addressing the majority of the comments. RQ-UKHPR1000-1735 (Ref. 28) was issued for follow-up points. The RP provided satisfactory responses to close the remaining comments. The residual comments were not expected to present a significant technical risk and any significant points were followed up as generic points (Ref. 18). One of the residual comments that was adequately closed had wider implications. This concerned the tolerance to more than one stud failure. The cited analysis shows that adequate sealing is maintained for the case where one stud has failed, but not for the case if two studs have failed (see below).

703. In summary, the RP provided adequate responses to the majority of the questions raised in RQ-UKHPR1000-1607 and RQ-UKHPR1000-1735 (Ref. 28). The outstanding comments were not expected to present a significant technical risk and any significant points were followed up as generic points (Ref. 18).

CRDM Adapter Sleeve

704. RQ-UKHPR1000-1643 (Ref. 28) was raised which covered the deep-dive review of the CRDM adapter sleeve and adjacent CH region in the RPV stress and fatigue analysis reports. For the 'Stress Analysis of CRDM Adapter Sleeve and Adjacent Closure Head of Reactor Pressure Vessel' (Ref. 190), 3 comments were raised, which related to the presentation of results and classifications of stress as global or local, for which satisfactory explanations were provided in the RQ response. I cover the fatigue analysis aspects in Section 4.4.1.2 below. It will suffice to say that several important comments were raised on these reports, to which the RP provided detailed and satisfactory responses to explain its approach and the work carried out. All comments raised have been closed.

Steam Generator Locations

705. RQ-UKHPR1000-1302 (Ref. 28) was raised to propose regions of the SG for detailed review and to request the associated references. The RP's response was that the requested references are BWXT proprietary documents, which were not within scope of the BWXT deliverables to the RP. Hence, they were not available. To permit a limited deep-dive review, the RP submitted its 'SG Base Design Report' (Ref. 180) and 'SG Transient Analysis Report' (Ref. 191), based on work undertaken by BWXT as the best available information. These were essentially high-level summary reports, albeit in some areas with slightly more detail compared to the SG General Report of Mechanical Analysis (Ref. 179). This notwithstanding, detailed design calculations were not supplied; hence, the scope of my review was limited compared to the level of scrutiny applied to other components.

706. RQ-UKHPR1000-1694 (Ref. 28) was issued which included a Category 2 comment relating to the FEA modelling of the feedwater nozzles. It was noted that the main feedwater nozzle is assessed using an axisymmetric model where the radius of the steam drum has been increased by a factor of 2.5 to ensure the correct hoop stress. However, the auxiliary feedwater nozzle is situated on the conical section of the steam drum. It was not clear what work has been done to ensure the use of an axisymmetric model remains appropriate given the actual plant geometry.

707. The RP stated that the use of a factor of 2.5 (as opposed to 2) to determine the equivalent cylindrical shell radius makes the assessment conservative, although the RP pointed out that the unfactored inside radius of the equivalent cylindrical shell was set to be equal to the normal distance (parallel to nozzle axis) from the nozzle centre (at the inside surface) to the conical shell centreline axis. Thus, the fact that the hoop stress in the shell will be higher at the top dead centre location on the nozzle has not been accounted for, which will erode some of the conservatism introduced through the use of a scaling factor of 2.5 rather than 2. In the 'Steam Generator Base Design Report' (Ref. 180), a minimum factor of 1.32 is reported for the auxiliary feedwater nozzle (local primary membrane stress at the nozzle to shell juncture under Design conditions), whilst the 'Steam Generator Transient Analysis Report' (Ref. 191) reports a maximum fatigue usage factor of less than 0.1. Given the (albeit not fully quantified) conservatism, plus the reported margins, the modelling assumptions are judged not to represent a significant technical or design risk

708. The most significant Category 3 comments included:

- Comparing the results listed in the 'Steam Generator Base Design Report' (Ref. 180) to those within the 'Steam Generator General Report of Mechanical Analysis' (Ref. 179), in many instances the values are identical, in other cases results are similar but different, and in some cases the results are very different. The larger differences were queried in RQ-UKHPR1000-1694 (Ref. 28) It is also noted that 'Steam Generator General Report of Mechanical Analysis' (Ref. 179) is dated prior to 'Steam Generator Base Design Report' (Ref. 180) and the 'Steam Generator Transient Analysis Report' (Ref. 191), so the RP was requested to confirm whether (Ref. 179) presents the latest information. The RP was also requested to explain what the relationship is between the design analyses reported in 'Steam Generator General Report of Mechanical Analysis' (Ref. 179) and those detailed in the BWXT reports (Ref. 180) (Ref. 179), i.e. whether the 'Steam Generator General Report of Mechanical Analysis' (Ref. 179) summarises the work reported in 'Steam Generator Base Design Report' (Ref. 180) and 'Steam Generator General Report of Mechanical Analysis' (Ref. 179), or if additional assessment work has been conducted by the RP which is then reported in 'Steam Generator General Report of Mechanical Analysis' (Ref. 179). The RP's response confirmed that the 'Steam Generator General Report of Mechanical Analysis' (Ref. 179) is intended to summarise the work reported in the ‘Steam Generator Base Design Report’ (Ref. 180) and ‘Steam Generator General Report of Mechanical Analysis’ (Ref. 179). The ‘Steam Generator General Report of Mechanical Analysis’ (Ref. 179) had since been up issued from the version reviewed and was now consistent with the ‘Steam Generator Base Design Report’ (Ref. 180) and the ‘Steam Generator General Report of Mechanical Analysis’ (Ref. 179).

- There is a very low margin between the required tensile area for the primary manway bolting and that actually available (0.5%) (Ref. 180). In addition, Table 1.1 of (Ref. 191) summary report gives a Fatigue Usage Factor (FUF) of 0.992 for the studs based on a design life of only 40 years i.e. less than the full SG design life of 60 years. Furthermore, (Ref. 191) specifies a limit of 80 pre-tension cycles for the studs. Given these very low margins it was queried whether any consideration had been given to changing the design of the primary manway, whether there is sufficient redundancy in the design to cope with a stud failure, and how the risk of failure of the current design is justified to be reduced to ALARP.
- The RP indicated that the studs can be replaced (i.e., they are a serviceable item) and that at every refuelling stage, visual inspection and ultrasonic examination are implemented to identify any degradation according to RSE-M. If any defect is detected, the stud would be replaced. The topics of stud fatigue and the redundancy in bolted seals is discussed further in Sub-sections 4.2.1.1 and 4.4.1.2 of my report.
- The RP stated that flow induced vibration loads were included in the assessment of the tubes. It was not clear how these loads were derived, as no details are provided within the ‘Steam Generator General Report of Mechanical Analysis’ (Ref. 179) or the ‘Steam Generator Transient Analysis Report’ (Ref. 191). In response the RP indicated that the flow induced vibration loads were derived from analysis of the critical regions of liquid and two-phase cross-flow in the tube bundle they also confirmed that the loads are included within the assessments reported in the ‘Steam Generator General Report of Mechanical Analysis’ (Ref. 179). As such the comment was considered closed, though the evidence was not clearly presented in the formal safety documentation.

Main Coolant Loop Locations

709. The detailed review of the MCL would focus on two regions; the Surge Line Nozzle and the Safety Injection Inclined Nozzle. The SL nozzle was selected because:

- It is a HIC weld.
- This is largest nozzle on the MCL and the margins for the SL are smaller than for the homogeneous MCL welds or the elbows.
- Limited details were provided in the Mechanical Analysis report regarding the modelling approach (Ref. 192).

710. RQ-UKHPR1000-1547 (Ref. 28) was issued which included several Category 2 comments. The Category 2 comments raised against the stress analysis (Ref. 193) are summarised below:

- The stress limit criteria for Test conditions are stated as being in accordance with RCC-M B3657, which in turn refers to RCC-M B3237. Requirement ‘a)’ of RCC-M B3237 states that the general primary membrane stress intensity shall not exceed 90% of the material yield stress at test temperature. The actual check undertaken by the RP was simply against the axial pressure stress and so did not represent the general primary membrane stress intensity. In consequence the stress measure evaluated by the RP was about half what it should be.
- There was also limited evidence provided in the MCL General Report of Mechanical Analysis (Ref. 192) to explain how the criteria for not undertaking

fracture assessments were met, in particular the limitations of moment loadings. The response provided by the RP stated that "the detailed information of limitations of external moments should be provided in each mechanical analysis report." However, this information was not available in (Ref. 193). The RP's response to RQ-UKHPR1000-1547 (Ref. 28) provided the required evidence, with a commitment to include it at the next issue of (Ref. 193).

- The level of detail provided in the 'MCL General Report of Mechanical Analysis' was also queried RQ-UKHPR1000-1148 (Ref. 28). The response provided by the RP stated that "the detailed information should be provided in each mechanical analysis report. In this general report, only the input data (for example, material properties, design loading and load combination) and the results are presented." On review there is very little additional information provided in the 'Stress Analysis of Surge Line Nozzle of Main Coolant Line' (Ref. 193). The expectation is that sufficient information should be provided to enable the assessment results to be recreated if necessary.

711. Reviewing the RP's responses to RQ-UKHPR1000-1547 (Ref. 28), all except one of the comments can be considered closed, although the responses reinforce some of the generic findings:

- Errors in checked and approved documents.
- Evidence to support claims in higher level documents not being provided in the lower-level documents, although the evidence was subsequently provided in the RQ responses.

712. The comment that has not been adequately closed out related checking whether the highest primary stress intensity had a 10% margin on yield under test conditions according to B3237 of RCC-M, instead the RP's checks were limited to simply checking the axial pressure stress. However, scoping calculations undertaken by my TSC indicated that the allowable test pressure for the pipe would be circa 31MPa and with a margin of 20%, which is below the margin of >40% quoted by the RP in the 'Stress Analysis of Surge Line Nozzle of Main Coolant Line' (Ref. 193).

713. RQ-UKHPR1000-1732 (Ref. 28) was raised to follow up on this topic. The RP's response acknowledges that both hoop and axial stress checks need to be considered and that the hoop stress check is contained within the dimensioning report (Ref. 181). The RP's response also acknowledges that the hoop stress check may be bounding. Hence, design code compliance has been demonstrated for the component under test conditions and the comment was closed.

714. The Safety Injection Inclined Nozzle on the MCL was also selected for detailed review for the following reasons:

- It is a HIC weld.
- Relatively low margins are predicted for Second Category stress conditions.
- Limited details are provided in the Mechanical Analysis report regarding the modelling approach.
- This nozzle is oblique and so the stress index method in RCC-M B 3685.3a does not apply.

715. RQ-UKHPR1000-1644 (Ref. 28) was issued with one Category 2 comment. Several comments raised in RQ-UKHPR1000-1547 against the SL were also applicable to the safety injection nozzle, though for clarity they were not repeated in the response to RQ-UKHPR1000-1644.

716. The Category 2 comment raised against the 'Stress Analysis of Safety Injection Inclined Nozzle of Main Coolant Line' (Ref. 194) related to the claim for mean slope at

the thickness transition for the auxiliary weld being less than 1/3. However, Figure 1 of the 'Stress Analysis of Safety Injection Inclined Nozzle of Main Coolant Line' (Ref. 194) shows a thickness transition of 45° (so a slope of 1/1) close to the weld. Whilst the RP's response did not contain any quantification of the spacing, independent calculations carried out as part of this review showed that the 45° transition was just outside the limit of influence of the auxiliary weld as defined by RCC-M and so was acceptable.

717. All RP comments raised in RQ-UKHPR1000-1644 (Ref. 28) were closed, though the RP's responses reinforce some of the generic findings:

- Errors in checked and approved documents.
- Lack of detail of claimed conservatisms.

Summary of Outstanding Comments – Design Code Compliance

718. As noted above broad and deep reviews were undertaken of a sample of the major vessels and piping in the UK HPR1000. Although several comments were raised against the RP's submissions the RP responded positively and addressed the vast majority of the comments and clarifications requested in the RQs.

719. The findings suggest that there is not a systemic problem with component designs meeting code requirements, on the basis that where evidence has been provided in response to RQs, the RP appears to understand the code requirements and has been able to demonstrate code compliance. However, several comments were raised where either insufficient evidence was provided to close out the comment, or the technical argument provided in response to the comment fails to justify the approach adopted. The comments not fully closed during the GDA include:

- Design by Analysis results being claimed as evidence in lieu of Design to Rule assessments from (Sections 3.2 and 3.7 (Ref. 18) and Sub-section 4.4.1 above), noting that this relates more to how the evidence is being referred to in other documentation as opposed to issues with the code compliance in itself.
- Tri-axial stress checks being omitted without adequate justification (Sections 3.3 and 4.2.2 (Ref. 18) and Sub-section 4.4.1 above).
- Thermal stress ratchet checks being omitted without adequate justification (Section 3.4 (Ref. 18) and Sub-section 4.4.1 above).
- The rules of ASME III Appendix A for tube sheets being applied incorrectly (Section 3.6 of (Ref. 18) and Sub-section 4.4.1 above).
- Lack of evidence that axisymmetric modelling of an asymmetric nozzle is appropriate (Section 4.3 (Ref. 18) and Sub-section 4.4.1 above).

720. It is my opinion that these represent only a low risk to the design of the components in question due to either the presence of reasonable margins, the fact that it is very rare for the omitted checks to be failed without also failing at least some of the checks that have been conducted and passed, or alternative means were used to justify the component. Nevertheless, it is important that these points, several of which are either generic or have the potential to become generic, and which inform the integrity demonstration for the most safety significant components in the UK HPR1000 are tracked to completion. I therefore raise the following assessment findings:

AF-UKHPR1000-0198 – The licensee shall, as part of detailed design, justify within the safety case where Design by Analysis is being claimed as evidence of code compliance in lieu of Design to Rule assessments.

AF-UKHPR1000-0199 – The licensee shall, as part of detailed design, justify that suitable tri-axial stress and thermal stress ratchet analyses are used to underpin the safety case. Where these analyses are not considered necessary, this should be justified.

AF-UKHPR1000-0200 – The licensee shall, as part of detailed design, demonstrate that the rules of ASME III Appendix A are applied for tube sheets.

AF-UKHPR1000-0201 – The licensee shall, as part of detailed design, document within the safety case the key assumptions and uncertainties applied in the finite element modelling methodologies, including where axisymmetric modelling of asymmetric nozzles is undertaken.

721. The RP's responses to questions raised on claims made for sealing of the RPV CH, based on tolerance to a single stud failure argument are also highlighted. Whilst the information presented identifies that the RP's approach meets the relevant code requirements, it is unclear whether the classification of this component assigned by the RP is reasonable, and whether a robust consequences argument has been presented to support it. Similar points were raised for other bolted connections, in that although stresses in the studs are within code limits and the studs may be a serviceable item, being tolerable to only a single stud failure may not guard sufficiently against common cause failure (Sub-section 4.2).

Generic Points and Conclusions – Design Code Compliance

722. The reviews of code compliance outlined above also identified a generic point concerned the traceability of evidence within the overall safety case to ensure a robust demonstration of code compliance. The issue of traceability of evidence is covered in Sub-section 4.1 of my report.

723. Further points relate to the RP's consideration of small margins and the level of conservatism in its design code assessments. In particular, for highest reliability components, where the expectation is that the safety case is especially robust and the corresponding assessment suitably demanding (ONR SAPs, Ref. 2). This means that the basis and context is provided when results are presented, particularly, if the results appear to suggest minimal margin against the acceptance criteria.

724. My high-level review of the 'RPV General Report of Mechanical Analysis' (Ref. 195) and the 'Reactor Pressure Vessel Dimensioning Report' (Ref. 196), identified that a number of results exhibit limited margin against the specified design acceptance criteria. I enquired as to how the RP analyses the results and determines its adequacy in RQ-UKHPR1000-0372 (Ref. 28).

725. The RP stated that a process is used to evaluate the margins available in the design calculation which considers factors such as dimensional tolerances and material properties. In addition to these analyses the RP indicated that, where limited margin is observed additional work may be performed to determine conservatism in the calculation through sensitivity studies or comparison to other calculations. I was encouraged by the RP's response, but with TSC support, I pursued its implementation in Step 4 of GDA (Ref. 18).

726. I raised comments regarding the small margins predicted over the minimum acceptable by the design code for several regions of the RPV including; CH flange, the closure studs, the sealing of the gasket, inlet and outlet nozzles and the core shell in RQ-UKHPR1000-1149 (Ref. 28).

727. Similarly, my review of the MCL design reports indicated a number of components with low margins (1% or less), on both primary stresses and shakedown. This notwithstanding, there were limited details presented in the 'General Report of Mechanical Analysis' (Ref. 192) to confirm that the assessments are conservative overall and that a small change in inputs would not lead to an unacceptable result. In addition, insufficient information was provided in the responses to RQ-UKHPR1000-1148 (Ref. 28) to establish that the level of conservatism and the result would not be sensitive to small changes in the inputs.

728. In some cases, the comments were answered through updates to the 'Mechanical Analysis' reports. This gives a basis of confidence that the RP understands the importance of evaluating small margins against the design code allowable limits taking cognisance of the level of conservatism in the analyses. However, my sampling of the RP's evidence did not support the view that the process was systematically applied. I draw confidence that the RP has a process available and has shown it can be implemented, however this has not been consistently. I have captured this under AF-UKHPR1000-0202 (below) to ensure it is rigorously applied for HIC components during site specific phases.

729. A final point relates to the lack of detail in the FE modelling methodologies. The RP cited 'The General FE Modelling document from the Introduction Report of Fangchenggang-3'. However, this was not available in GDA.

730. Overall, following my reviews of the RP's demonstration of design code compliance, the following generic points were identified:

- Referencing and Configuration Management.
- Source of System Loads in Pipework and External Loads on Nozzles.
- Lack of traceability of Evidence.
- Lack of Discussion of Small Margins.
- Lack of Detail of Claimed Conservatisms.
- Lack of Detail on FE Modelling Methodology.

731. The traceability aspects of the evidence and loads are captured in my assessment findings (Sub-section 4.1). I view referencing and configuration control as an integral part of traceability in the SI safety case, which has already been captured in AF-UKHPR1000-0186 raised in Sub-section 4.1 above. Aspects related to demonstration of assumptions and uncertainties used in modelling for design code compliance are covered under AF-UKHPR1000-0201 (above).

4.4.1.2 Fatigue Methods and High Fatigue Usage Factors

Fatigue Methods

732. To aid my initial assessment of the RP's fatigue analysis, it was important to understand the inputs to the analysis. From the review of the 'Reactor Pressure Vessel General Report of Mechanical Analysis' (Ref. 195), I noted that there was minimal information on the transients driving the presented results. In GDA Step 4, and with TSC support, I sampled several of the RP's fatigue analyses. An example was the SL, with my queries raised in RQ-UKHPR1000-1547 (Ref. 28) to gain clarification on a number of aspects of the RP's fatigue analyses:

- The key modelling assumptions were not discussed or justified as being appropriate and conservative.
- Limited details were provided to explain how the thermal analysis has been undertaken and how the thermal boundary conditions have been derived.
- The methodology for applying the RCC-M ZE200 mixed method analysis is not explained in any detail.

- Limited details are provided on how the fatigue assessment has been conducted.

733. Although the response to RQ-UKHPR1000-1547 (Ref. 28), contained adequate responses to my queries relating to the SL, I was mindful that these were also generic points (Ref. 18). In RQ-UKHPR1000-1619, I therefore also questioned whether the RP followed a generic methodology or guidance which explained its approaches, and how to consider conservatism, key assumptions, and uncertainties in the application of its fatigue analyses (Ref. 28). The RP supplied further information in response to RQ-UKHPR1000-1619 (Ref. 197). The RP also outlined its approaches, conservatisms, assumptions and uncertainties and sensitivity studies employed in the fatigue analyses report for the MCL Charging Line (Ref. 197).

734. I draw some confidence from these submissions that the RP is aware of the need to understand and discuss the significance of high fatigue usage factors, taking cognisance of the uncertainties and conservatisms. However, the response was very much focussed on the RQ-UKHPR1000-1619 (Ref. 28) questions with no commitment made on how to address the generic point. The licensee will need to ensure that these processes are applied systematically to ensure that the conclusions drawn, in situations where there are high fatigue usage factors, are robust. There is also a need to ensure that these processes are applied rigorously in situations where there are apparently low margins against code limits for HIC components (Paragraph 723 and 728). I therefore raise the following AF to track the production and implementation of these processes:

AF-UKHPR1000-0202 – The licensee shall, as part of detailed design, develop and implement a process for the evaluation of small margins against design code limits for high integrity components and in situations where fatigue usage factors are high. This should include, but not be limited to, the reactor pressure vessel components, such as the closure head flange, closure studs, core shell, inlet and outlet nozzles, and the main coolant loop charging line weld.

735. A further point identified from my initial high-level review of the ‘Reactor Pressure Vessel General Report of Mechanical Analysis’ (Ref. 195) relates to the specific transients included in the design fatigue calculations. It is stated in the ‘Reactor Pressure Vessel General Report of Mechanical Analysis’ (Ref. 195) that components are predicted to be subject to no more than 25 Level C transients within the station’s life. It is ONR’s expectation that throughout their operating life, SSCs should be operated and controlled within defined limits and conditions (operating rules) derived from the safety case (ONR SAP EMC.21, Ref. 2). For structural integrity, this includes the consideration of all reasonably foreseeable transients that could influence the degradation of structures and components, including fatigue. However, it was not clear whether these Level C transients were accounted for within the design fatigue calculations, or if there is a sound basis for their omission. To address this concern, I raised RQ-UKHPR1000-0372 (Ref. 28) to seek more information.

736. The RP did not fully answer the question raised in RQ-UKHPR1000-0372 (Ref. 28). In addition, if the Level C transients are to be excluded from the fatigue analysis, further information as to the actions that will be taken following the occurrence of such a transient is needed. I followed up this item in Step 4 of GDA by reviewing a sample of the ‘General Report of the Mechanical Analysis’ for the PZR and the SG (Ref. 198) (Ref. 199).The PZR report covered a component designed to RCC-M code, whilst the SG is designed to ASME III code. In both cases, my review highlighted that the Level C transients (such as SG tube rupture, small LOCA and small steam line breaks etc), which are expected to occur typically once in the design life, were excluded from the fatigue analyses. This is consistent with the design code rules where the fatigue calculations are limited to the more frequent transients (Category 1 and 2 in RCC-M or Level A and B in ASME III). Nevertheless, the more severe Level C and D transients

are included in the DTA work as potential limiting loads. I was satisfied with the RP's omission of Level C/D transients from the fatigue analysis accorded with accepted practice.

737. This notwithstanding, the RP offered no further information to clarify its approach to managing the integrity of components, should a Level C or Level D transient occur within the plant design life. I take confidence that the consequences of gross failure are reflected in the SI classification of the SSC. Moreover, the design codes include rules and guidance for plant revalidation.

738. I am also aware of international OPEX on the influence of environment on the design fatigue calculations (Ref. 27). I raised RQ-UKHPR1000-0278 to obtain more information on the RP's understanding and position on the influence of environment on design fatigue calculations (Ref. 28). I confirmed that the RP plans to use the probationary rules within RCC-M, which proposes a methodology for considering the environmental effects of PWR water on fatigue analyses for austenitic or austenitic-ferritic steel components (FEN factors). This is a developing technical area and hence the adequacy of this approach is outside the scope of GDA, but in its response to RQ-UKHPR1000-1368, the RP committed to addressing this change during the site-specific stages (Ref. 28). I welcome the RP's commitment to re-evaluate the fatigue life of components at the site-specific stage. However, I also recognise the residual risks relating to the potential for significant developments in the approaches for accounting for environmental effects and hence in fully justifying the SSC design life for the UK HPR1000. I consider this a matter that warrants tracking to completion during the site-specific stage. I therefore raise the following assessment finding:

AF-UKHPR1000-0203 The licensee shall, as part of detailed design, justify and implement approaches to account for the environmental effects of primary circuit coolant water, in its evaluation of the design life of structures, systems and components for the UK HPR1000. This should include, but not be limited to, the safety significant vessels and piping designed to the RCC-M code and, where combinations of codes are justified such as the steam generator and reactor coolant pump, to ASME III.

739. A further topic to emerge from my review of the RP's basis for confidence in the achievement of UK grid code compliance (see Sub-section 4.4.3.2 below), related to the RP's approach to the fatigue analyses for the control rod drive mechanism (CRDM) J-groove weld location. The CRDM adapter sleeve to closure head (CH) attachment weld includes a crack-like feature at the end of the crevice between these two components. The fatigue assessment of these types of features can be challenging and the RP's fatigue assessment for the CRDM J-groove welds differs from other parts of the UK HPR1000, which are essentially based in a Miner's law approach, using the relevant S-N curves from RCC-M Z. However, for locations with geometric discontinuities, such as CRDM J-groove weld, the RP has applied the criteria of Appendix ZD of RCC-M, in which a crack like analysis is performed as discussed in RQ-UKHPR1000-1461 (Ref. 28). Notably, despite the crack-like feature, a relatively low cumulative usage factor (CUF) of 0.6 was presented for the CRDM J-groove weld, which was unexpected (Ref. 200). These types of features are typically substantiated using DTAs, therefore with the TSC support (Ref. 18), I issued RQ-UKHPR1000-1643 to clarify the RP's position (Ref. 28).

740. My queries in RQ-UKHPR1000-1643 covered aspects of both the stress and fatigue analyses and included the presentation of results, origin of loads, FE modelling methods and the assessment of the crack-like feature. The RP confirmed that the CRDM J-groove weld (adapter sleeve crevice) has been assessed using Annex ZD of RCC-M 'Analysis of the Fatigue Behaviour of Zones with Geometric Discontinuities' (Ref. 14). This requires stresses to be extracted a specified distance from the

discontinuity, which has been modelled as crack, and uses specific fatigue initiation (S-N) curves to calculate the fatigue life of the feature.

741. My sampling of the other components confirmed that the RCC-M Annex ZD approach for crack-like discontinuities had wider application and for example was applied to small nozzle locations in the PZR upper and lower shells and at the weld between the heater sleeves and lower head in the PZR (Ref. 43).

742. Thus, although the RP's response to RQ-UKHPR1000-1643 (Ref. 28) clarified the approach to the fatigue analyses of the CDRM adapter sleeve crevice, there is a need to consider the efficacy of this approach. ONR's position on the use of RCC-M Annex ZD is that the approach is accepted in principle. However, its application needs to be underpinned by adequate evidence to validate the specific fatigue initiation (S-N) curves, in particular, for material interface regions in dissimilar metal welds (DMWs). I therefore raise the following AF to track the completion of the justification of the application of the RCC-M Annex ZD approach in the UK HPR1000:

AF-UKHPR1000-0204 – The licensee shall, as part of detailed design, justify the application of fatigue initiation analysis to RCC-M Annex ZD to components, and demonstrate the validity and veracity of the test data that underpin the specific fatigue initiation (S-N) curves.

High Fatigue Usage Factors

743. My review of the CUFs for the RP's grid code compliance work, highlighted high CUFs (≥ 0.75) in regions of the RPV closure head (head flange - 0.75, flange thread - 0.87 and stud thread - 0.98) and the charging line branch piping of the MCL -0.78. I needed to gain an understanding of what was driving these apparently high damage levels. In particular, were they solely due to conservative assessment assumptions or were the design features at these locations genuinely prone to fatigue?

744. With TSC support, RQ-UKHPR1000-1619 was raised to identify a complete listing of regions where CUFs of ≥ 0.75 were calculated (Ref. 28). In addition, I sought to gain further clarification to inform an independent view and understanding of the level of conservatism, uncertainties, assumptions and hence, the adequacy, of the fatigue design of a sample of components in the UK HPR1000.

745. The regions of highest predicted CUFs are the bolts on the SG primary manway and lower handhole bolts along with the RPV stud threads, where FUFs greater than 0.9 were predicted. The high FUFs in these regions are largely due to the use of a fatigue strength reduction factor of 4.0, as required by ASME III for high-strength alloy steel bolts (NB-3232.3(b)). A fatigue strength reduction factor of 4.0 is also required by RCC-M (B 3252.3) unless it can be shown by analysis or tests that a lower value is appropriate.

746. Nonetheless, I was concerned with the high fatigue usage factors presented in the closure components in the RPV and SG. The 'SG Base Safety Report' presents a very low margin between the required tensile area for the primary manway studs and that actually available (0.5%). In addition, the CUF of 0.992 for the studs is based on a design life of only 40 years (less than the full SG design life of 60 years claimed for the UK HPR1000). The 'SG Transient Analysis Report' also specifies a limit of 80 pre-tension cycles (Ref. 191).

747. A similar stress margin and high FUF for the primary manway studs under design conditions is reported in the 'Steam Generator General Report of Mechanical Analysis' (Ref. 179), though this appears to cover the full 60-year design life of the SG, and without the specification of a limit on pre-tension cycles. The RP subsequently confirmed that the 'Steam Generator General Report of Mechanical Analysis' (Ref.

179) has been up-issued and is now consistent with the 'Steam Generator Base Design Report' (Ref. 180) and the 'Steam Generator Transient Analysis Report' (Ref. 191). This would therefore appear to leave the SG primary manway design with limited stress margin, a fatigue life of 40 years and a limit of 80 pre-tension cycles. In RQ-UKHPR1000-1619 (Ref. 28), I therefore queried whether any consideration had been given to changing the design of the primary manway, whether there is sufficient redundancy in the design to cope with a stud failure, and how the risk of failure of the current design is justified as ALARP. The RP's response stated that the studs can be replaced (i.e. they are a serviceable item). At every refuelling stage, visual inspection and ultrasonic examination are implemented to identify any degradation according to RSE-M. If any defect (such as a crack) is detected, the stud would be replaced right away.

748. I draw confidence from the RQ-UKHPR1000-1619 response (Ref. 28) and from the fact that the UK HPR1000 SGs are a mature design, and that the RP's information are based on BWXT reports, which carry a Design Report Certification Statement and stamp (Ref. 180) (Ref. 191). It is my expectation that the licensee should establish the SI classifications and the CUFs for HIC closure components. If appropriate, further work should be undertaken to either justify the integrity of the component (e.g. DTA to design code requirements) or introduce mitigating measures (e.g. ISI or component replacement policies) to ensure the integrity of closure components are commensurate with their SI classifications and the 60-year design life of the UK HPR1000. I have previously raised AF-UKHPR1000-0191 for the licensee to confirm the SI classification of closures in HIC components (Sub-section 4.2 above), which should include consideration of the fatigue aspects.

749. The RP's response to RQ-UKHPR1000-1619 (Ref. 28) also explained why high FUFs were predicted in the identified regions, including a description of conservative assumptions used such as in the loads and transient combinations. In addition, for each region where high FUFs were predicted, the RP outlined its strategy and the measures in place to manage the risks, which included provisions for ISI, replacement or repair and the use of a fatigue monitoring system. These generally appeared reasonable with the majority of locations subject to ISI in accordance with RSE-M and/or repair or replacement measures available if needed. Nonetheless, I noted that for several regions, although repair or replacement was feasible, the RP was not proposing any means of monitoring for potential fatigue damage either by ISI or transient accounting e.g. the RPV Head flange, PZR heater sleeve and the MR bottom plate bolts in the RVI, as per the response to RQ-UKHPR1000-1619 (Ref. 28). For these components, it may be reasonably practicable to provide additional means to afford forewarning of failure either by introducing speculative ISI or monitoring the fatigue usage.

750. Furthermore, for some component locations the RP argued that the fatigue analyses were overly conservative and so in the absence apparently of a repair or replacement option, no additional measures to provide forewarning of failure were offered. These locations included the PZR lower head local to the heater sleeves and the SG blowdown nozzle and drain hole regions, as explained in the RP's response to RQ-UKHPR1000-1619 (Ref. 28). I am aware of OPEX from Sizewell B that highlights the potential for compromises to the structural integrity at these locations and which were subject to successful repairs, albeit the failure mechanisms were associated with Stress Corrosion Cracking (SCC) rather than fatigue. I also recognise that the consequences of failure for these locations are likely to be limited as reflected in the SI classifications. Nevertheless, I expect a demonstration that the consequences of failure are acceptable and that measures are taken to reduce risk were reasonably practicable e.g. speculative ISI and transient monitoring. I therefore raise the following assessment finding for the licensee to consider the need for such measures:

AF-UKHPR1000-0205 – The licensee shall, as part of detailed design, demonstrate that all reasonably practicable measures have been implemented to minimise risks to the integrity of locations with cumulative fatigue usage factors of ≥ 0.75. This should include, but not be limited to, operational experience, the uncertainties and conservatism in the analyses, where component repair or replacement option is not available and where additional measures to afford forewarning of failure could be implemented.

4.4.1.3 Code Version Control

751. Several of the code editions proposed for the UK HPR1000 major vessels and piping are over 10 years old RCC-M-2007 (design), RSE-M-2007, 2010 +2012 Addendum (ISI) and for the SG ASME III-2007 and 2008 Addendum (design). The content of individual design and manufacturing codes changes over time as OPEX is taken on board and errors in the codes are corrected. The earlier code editions therefore do not necessarily reflect current good practice. Hence, the version of the code used is important.

752. It was my opinion for the purposes of this GDA that the starting point should be the latest version of the code that is available within the GDA and prior to a design freeze. There may be valid reasons for selecting an earlier version of the code, but it is my expectation that these are documented. In addition, I expect that the user of an older code is aware of the changes that have been made between the chosen code (known as the code of construction) and the most recent issued version.

753. Within the GDA submission 'Suitability of Codes and Standards in Structural Integrity' (Ref. 176) the RP identified that certain selected codes for the UK HPR1000 are not the latest versions. In each instance the RP states that a review against a newer version of the specific code will be completed and the impact on the UK HPR1000 design will be identified. I viewed this as a positive approach, subject to the proviso that the benchmark is the latest version of the code.

754. To explore the RPs approach, I raised RQ-UKHPR1000-0277 and RQ-UKHPR1000-0417 (Ref. 28). The RP explained that it considered the latest version of the code to be the latest official English translation of the code. The RP also indicated that if a technical issue arises for which the nominated reviewer is not empowered to resolve (i.e. the change may have significant technical or project implications), the issue can be escalated using the RP's technical decision-making process (Ref. 201).

755. The RP has stated that currently there are no criteria with which to judge the significance of any potential changes. If a change is necessitated, a Technical Change Notice (TCN) will be developed and submitted to the change management process (Ref. 7). At a high level this approach appeared reasonable. I also noted that the code comparisons were to be documented and submitted to ONR as part of the GDA Step 4 submissions.

756. Whilst I was content that the RP intended to complete an appropriate activity to address this identified gap, I was of the view that the version of the code chosen was not appropriate. In my opinion it is the latest version of the code that should form the basis of RGP. The selection of the 2016 version (the latest English translation at the time of writing) compared to the 2018 version (the latest version) introduces a risk that important changes are not identified and will be uncovered later in the project when it will be more difficult to implement.

757. It was the RP's position that the time interval between release of the French version of the RCC-M 2018 and the associated English would not induce a significant risk. The RP also clarified that it is a member of the RCC-M China User Groups which provides an open platform in China to promote the exchange and application of the RCC-M

code. In addition, the RP stated that it currently held the chairman position of RCC-M China Specialised User Group (construction group). Through these groups the RP claims it has good communication channels with AFCEN and would have visibility of any significant changes in a timely manner.

758. In terms of the long-term awareness of changes to the codes of construction, I am content that this is not an issue for the GDA and can be resolved during the site-specific stages. However, I was aware of some significant changes that have been made within the 2018 version of RCC-M which should be considered by the RP. For example, RCC-M version 2018 incorporates accepted practice relating to forging manufacture based on the French ESPN order. The RP subsequently committed to using the 2018 version of RCC-M. The application of the code, in particular, the significant changes which include recent OPEX relating to forging manufacture were followed-up in Step 4 of GDA (Sub-section 4.5.1 of this report).

759. The position on code versions is more complex for the UK HPR1000 SG and RCP because the RP needs to consider developments in both the US and French design and construction codes. The RP provided code comparison documents covering developments in the RCC-M and ASME codes (Ref. 202). Nevertheless, in RQ-UKHPR100-1413, I asked the RP to explain under what circumstances a further review of the differences (gaps) between the US and French codes and standards would be initiated (Ref. 28). The RP indicated that a further comparison between ASME (extant version) used for SG and RCC-M/RSE-M (extant version) used for the SG and other major vessels of nuclear island will be performed at the site-specific stages. In addition, a gap analysis (or a review of) differences would be triggered by any key developments in either US or French codes. The details of these arrangements would be established with the licensee, but new gaps, if any, will be identified and considered in design and construction of the UK HPR1000 SG (and RCP) during the site-specific stages.

760. In summary, I am satisfied that the RP has recognised the need to keep abreast of developments in both the US and French nuclear design and construction codes and that adequate processes should be available to inform the design code provisions for the UK HPR1000 SG. The RP's approach gives a basis for confidence that significant changes in the design codes can be identified and actioned. However, during site-specific stages, it will be necessary for the licensee to develop and apply similar processes to establish the significance of changes in the design and construction code in the intervening years. Therefore, as compliance with an appropriate design code forms the basis for accepted practice in design, I raise the following assessment finding for the licensee to develop and demonstrate effective implementation of its processes for accounting for changes in the design codes during the site-specific stages.

AF-UKHPR1000-0206 – The licensee shall, as part of the detailed design and where the design approach of combining design codes is justified, demonstrate that all relevant design and construction code developments are implemented, taking cognisance of the need to reduce risks to as low as reasonably practicable.

4.4.1.4 Design Transient Specification

761. I expect the RP to provide a schedule of design loadings for structures and components with a conservative estimate of its frequency of occurrence for all design and accident conditions (ONR SAP EMC. 7, Ref. 2).

762. During my review of the methods for the stress analyses I reviewed the overall design transient specification. The RP has presented the design transient specification within, 'Design Transient Specification' (Ref. 41). From the information that was presented it was not clear why the number and occurrences of transients was appropriate or

conservative. In addition, there was limited detail upon which to base subsequent analyses. To address these observations, I raised RQ-UKHPR1000-0280 (Ref. 28).

763. The RP stated that the basis for transient identification includes consideration of:

- a) Normal operation and manoeuvring operation of the plant;
- b) Postulated initiating event identified for the plant;
- c) Design and manufacture requirements for the equipment, and;
- d) Operating experiences of similar plants.

764. The RP also indicated that conservative assumptions for the transient frequency and magnitudes are employed. As an illustration for the start-up transient, the anticipated number of occurrences is 40 for a design lifetime of 60 years, however, to account for re-fuelling and maintenance, a total number of 210 occurrences is assumed. In addition, for transient definition the maximum heat up rate (40°C/h) is considered.

765. The calculation of the transients is not usually an area for structural integrity to form a view on, so I engaged with the ONR Fault Studies Inspector to understand if these are adequate. I consulted the ONR Fault Studies Inspector who was content that the RP's methods should be appropriate for determining the magnitudes used within the transient specification. In addition, ONR's PSA Inspector was satisfied that the transient frequencies are conservative.

766. For SI it is important that the RP implements adequate processes to ensure that conservative estimates of loadings and combinations are employed. There is also an expectation that the CSRs and supporting documents should give a traceable narrative that provides the safety case user with confidence that all loading scenarios have been considered in a logical and rational manner, to minimise the potential for any unforeseen loading conditions, which may compromise the structural integrity of the plant (ONR SAP SC.2 and SC.4, Ref. 2).

767. With TSC support, I pursued this matter further in Step 4 of GDA. I sampled the pressuriser CSR (Ref. 31), in particular, Evidence 1-2-2*:* 'Design loads (including load combinations) have been determined using established and conservative procedures within design basis.

768. The detailed information held to underwrite Evidence 1-2-2 is claimed in the 'Design Transient Specification' (Ref. 41) and the 'Pressuriser Design Specification' (Ref. 203).

769. The 'Design Transient Specification' (Ref. 41) discusses the five condition categories and lists various design transients and their number of occurrences. The 1st Category transients are typically defined in terms of temperature, pressure and rates of change of the same quantities. Other category transients are typically described in terms of a series of high-level events. However, it is not clear how the individual transients were identified/selected, how the contributing events are determined and how the design parameters for the pressuriser are quantified. A list of design transients for Fangchenggang-3 is identified in the reference list, but is not referenced within the 'Design Transient Specification' document (Ref. 41). Thus, there is no traceable link back to the OPEX that informs the definition of the events and their numbers of occurrences.

770. Likewise, the Pressuriser Design Specification (Ref. 203) includes main design parameters for the PZR, and states that they are defined in the 'Main Design Technical Parameters' document (Ref. 204). The external conditions for the PZR i.e. the environment within the nuclear containment, are tabulated without reference to their origin. Many component mechanical loads (forces and moments at nozzles) are also tabulated in Appendix A of the 'Main Design Technical Parameters' document (Ref. 204), also without reference to their origin. The most recent revision (C) of the PZR

Design Specification may have some updated values of loads, but provides little further detail on their derivation (Ref. 205).

771. The most recent version of the Pressuriser CSR (Ref. 206) now states that: *"*external loads are mainly from GDA primary circuit mechanical analysis" and "loads and conservative load combinations are considered under different working conditions. Deadweight, pressure and thermal expansion analysis use the static method. Seismic and LOCA analysis use the time history method.*"* The CSR also states that the nozzle loads from the auxiliary pipework are derived using the 'PIPESTRESS' software. Whilst this may provide more information to an evolving picture of how the loads have been derived, the absence of references means that there is still an absence of traceability to the data defined in the 'Pressuriser Design Specification' (Ref. 203).

772. Similar lines of questioning, relating to the transparency and traceability of design loads, were raised in for example RQ-UKHPR1000-1148 and RQ-UKHPR1000-1150 (Ref. 28). The RP's responses were typically of a very high level and general nature, lacking any references to allow the safety case user to drill down into the detail.

773. A response to RQ-UKHPR1000-1465 (Ref. 28), provide some additional description of the process of deriving the design loads. In response to the RQ, further information relating to the PZR was received late in the GDA but could not be assessed (Ref. 207) (Ref. 208).

774. In conclusion, there is a lack of a consistent provision of succinct descriptions of the process by which the design loads, as listed in the design specification for each component, are derived. There is also a lack of full traceability i.e. references, to the sources of the loading magnitudes and their frequencies for those numbers.

775. To be clear, the shortfall is with traceability rather than the actual magnitudes and frequencies of occurrence of the transients. However, to accord with (ONR SAP SC.2 and SC.4, Ref. 2) from an SI perspective, there should be a traceable path from the assessment of OPEX, through the identification of postulated initiating events, definition and quantification of the design transients, the derivation of inputs to the mechanical analyses, the mechanical and thermo-fluid analyses, the collation and post-processing of the analysis outputs to the generation of the tabulated loads data in the relevant Design Specifications. I have previously raised an assessment finding (AF-UKHPR1000-0186) under Sub-section 4.1 above, which I consider captures this accordingly.

4.4.2 Assessment of Combining Codes and Standards

776. The UK HPR1000 nuclear island is designed to the French RCC-M code, but the RP is proposing a combination of both United States (US) and French codes for the SG and RCP components. Design and construction is to ASME III Class 1 (with RCC-M supplements), whilst the pre-service inspection (PSI) and in-service inspection (ISI) is to the French RSE-M inspection rules for mechanical components of PWR nuclear islands.

777. The use of a combination of established nuclear design codes to underpin the Structural Integrity provisions for a pressure boundary component that forms a principal means of fulfilling nuclear safety functions is novel to ONR. In addition, the RP's proposal appeared inconsistent with meeting ONR SAP ECS.3 (Ref. 2):

"The combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated."

778. The main intent of avoiding a combination of codes and standards for a single aspect of an SSC is to avoid selective and more lenient aspects of codes and standards being chosen. This may with the result in collective provisions potentially providing an inadequate basis to justify the integrity of the SSC, compared to the holistic provisions offered in an established design, construction and inspection code e.g. ASME III/XI or RCC-M/RSE-M.

779. I sought an explanation and justification of the selection of codes and standards proposed by the RP. These are important to underpin the structural integrity of key SSCs important for safety, and should be commensurate with the ONR expectation of reducing risks ALARP through a balanced consideration of the benefits and detriments of potential component design options. Within Step 3 of GDA, my enquiries were only focused the SG because it only became apparent during Step 4 of GDA that the RCP will also be designed and built to ASME III Class 1. My Step 4 GDA assessment considers this aspect, which will be discussed later on.

780. In completing my Step 3 GDA assessment, I was broadly satisfied that there is no evidence to suggest the RP's intent is to adopt selected aspects of each code. On the contrary, the collective SG code provisions for structural integrity exceed those established in the individual US and French design and construction codes.

781. Throughout my assessment, I raised a number of reservations related to the RP's consideration of relevant risks relating to the SG code provisions and their mitigation. I viewed this matter as a potential shortfall in the safety case for the UK HPR1000 SG, and as such, raised RO-UKHPR1000-0033 (Ref. 49). The purpose of this RO was to gain further explanation and assurance in the RP's process and consideration of SG code relevant risks and that that such risks are reduced to ALARP.

4.4.2.1 Overview of RP Proposals

782. Late in the GDA, the RP stated that its intent to also adopt a combination of US and French design and inspection codes for the UK HPR1000 RCP by using an approach similar to that used for the UK HPR1000 SG.

783. I have sought an explanation and justification of the selection of codes and standards to underpin the structural integrity of the SG and RCP are commensurate with reducing risks ALARP through a balanced consideration of the benefits and detriments of potential design options. In addition, as the RP classifies the RCP and primary/secondary SG pressure boundaries as HIC, the collective provisions for structural integrity need to provide an adequate basis for a highest reliability claim.

4.4.2.2 UK HPR1000 Steam Generator (SG) Code Proposals and Risks (RO-UKHPR1000-0033)

ALARP Optioneering

784. At Step 2 of GDA, I concluded that in Fangchenggang-3 the structural integrity provisions for the SG were founded on supplementing design to an established nuclear design code, namely ASME III, with additional measures to also achieve broad compliance with the French RCC-M code. Thus, collectively the overall provisions went beyond basic compliance with an established nuclear design code.

785. I sought to understand why these additional measures were implemented for Fangchenggang-3 in RQ-UKHPR1000-0109 (Ref. 28). The response indicated that this was to meet the requirements of the Chinese Nuclear Safety Regulator, the National Nuclear Safety Administration (NNSA). I also understood the rationale for the RP's selection of codes and standards for Fangchenggang-3 was primarily driven by a world-wide shortage in SG design and manufacturing capability. I concluded that that

there was no evidence to suggest the RP's approach is selective, adopting more lenient aspects of codes and standards potentially resulting in an inadequate basis to justification of component integrity, when compared to the holistic provisions afforded by compliance with an established nuclear design code (ONR SAP ECS.3, paragraph 173, Ref. 2).

786. Nonetheless, I emphasised that the use of a combination of codes and standards incurs risks that need to be managed throughout the complete life cycle of the component. Similarly, alternative code options, to improve coherency and consistency may also incur risks. In response, the RP initiated a high-level SG ALARP code assessment covering several UK HPR1000 SG design code options with consideration of the benefits, detriments and risks (and their mitigation) with the aim of assuring that its preferred option i.e. Option 1 as in Fangchenggang-3 using both US and French codes and standards was commensurate with reducing risks to ALARP.

787. The RP presented its high level ALARP assessment for SG code in 'High Level ALARP Assessment for SG Code' (Ref. 209). The SG design code options included that used in Fangchenggang-3 (Option 1) along with several other options including design, construction, and inspection to either ASME III/XI (Option 2) or RCC-M/RSE-M (Option 3) or a re-design of the UK HPR1000 SG to RCC-M/RSE-M (Option 4). The RP concluded that Option 1 (Fangchenggang-3) is the ALARP SG design and construction option for the UK HPR1000.

788. In accordance with the UK regulatory regime, ONR does not prescribe codes and standards for SSC, but recognises that design codes such as ASME III and RCC-M are sources of RGP. If full compliance with the provisions of these codes is achieved this would provide a high level of structural integrity demonstration. This is irrespective of the additional measures beyond normal practice i.e., recognised codes and standards that are expected to meet a highest reliability claim.

789. I identified a number of strengths in the RP's high-level SG ALARP Assessment:

- The RP had identified and considered a wide range of options.
- The RP had adopted a systematic approach with inter-discipline input informing a view on the SI provisions.
- There was independent input.
- There was some consideration of the potential wider plant risks.

790. In contrast, there was significant evidence provided to justify Option 1, which raised uncertainty as to whether the overall approach involved a balanced consideration of the options. The scoring was also heavily influenced by the potential cost and schedule impacts. This was countered by the observation that the scoring appeared not to take credit for the additional measures proposed for Option 1 compared to the alternative options.

791. My views on the options were primarily based on safety considerations. I recognised that Option 1 includes several additional measures compared to the alternative options, and although full compliance with either ASME or RCC-M/RSE-M is not achievable the collective SI provisions exceed those of an SG designed, manufactured and inspected to established codes and standards. For example, Option 1 achieves full compliance with the design and manufacture and inspection pre-service to ASME III, compliance with the design aspects of RCC-M along with compliance with significant aspects of the manufacturing and pre-service inspection requirements of RCC-M/RSE-M.

792. A corollary of implementing Option1 is that to meet code compliance the RP is committed to undertaking two PSIs; one to ASME Section XI during design and manufacture, and one to RSE-M as a fingerprint prior to operation. The access and

inspectability aspects are discussed in Sub-section 4.6.3 of this AR, it will suffice to say that these additional measures, in principle, afford significant reductions in risk, irrespective of the UK expectation for further measures to underpin a highest reliability claim. I am therefore satisfied that the RP has provided suitable and sufficient evidence to underwrite the view that, in principle, the RP's proposal to use Option 1 is viable.

793. I also conclude that based on safety considerations alone any of Options 1 to 4 are in principle a viable in terms of providing a sound foundation for the structural integrity case to underpin a highest reliability claim for the UK HPR1000. However, on a balance of the risks from its ALARP assessment, the RP judged that Option 1 was its preferred option for the design of the UK HPR1000 SG. I was satisfied Option 1 was tenable.

794. Nonetheless, to justify the use of Option 1 and comply with ONR SAP ECS.3 (Ref. 2), I sought further information and justification from the RP in several areas: completion of due process; the management of the SG code risks; use of OPEX and contingency planning.

Due Process

795. The RP's due process for undertaking the high level ALARP assessment for the SG included the input of independent UK expertise together with inter-discipline representation at two workshops (Ref. 210) (Ref. 211). I reviewed the minutes from these workshops and raised RQ-UKHPR1000-0219 to seek clarification of the completion of the RP's due process (Ref. 28). My question covered confirmation of the closure of actions, agreement of the scoring criteria, the level of support for the conclusions, and the views of the GNSL technical committee.

796. The RP confirmed that before the response to RQ-UKHPR1000-0219 was submitted that a draft was discussed without undue reservations by the members, which included the views of independents experts. The revision of the scoring criteria was to make the scoring more explicit and that there was no change to the overall conclusions in terms of the RP's ranking of the SG code options. The SG code strategy and options were also discussed by the GNSL technical committee who did not challenge the conclusions. I was therefore satisfied that the RP's SG code ALARP report had completed the RP's due process.

797. I raised RQ-UKHPR1000-0219 to seek clarification of the completion of the RP's due process, and as a pre-requisite, to considering the RP's management of the risks. I concluded that the SG code ALARP report (Ref. 209) has completed the RP's due process. This notwithstanding, the focus of the RP's response to RQ-UKHPR1000-0219, (Ref. 28) was a description of the quality assurance (QA) arrangements for the SG design and manufacture implemented in Fangechenggang-3, which are also proposed for the UK HPR1000. The underlying philosophy in developing the QA arrangements and how they are intended to control and/or mitigate risks at the physical and organisational interfaces was not explained, nor was it clear how these arrangements would be adapted for the licensee.

798. In addition, there needs to be a basis for confidence that in the future, adequate control arrangements can be developed to secure achievement of the design intent and also that there are no significant 'gaps' in the organisational responsibilities during the SG design, manufacture, installation and operation. I was therefore unclear how the design intent is sustained through-life and what information the RP will provide in GDA to ensure the licensee is able to maintain the UK HPR1000 SGs, and whether these are commensurate with reducing risks to ALARP.

Steam Generator Code Risks

799. The response to RQ-UKHPR1000-0219 (Ref. 28) confirmed that the RP was proposing similar QA agreements for the UK HPR1000 SG as implemented at Fangchenggang-3, and that surveillance of activities by, or for, the owner and the contractor or suppliers would be covered in the contractual documentation. However, veracity of the OPEX underwriting the RP's approach along with the underlying philosophy in developing the QA arrangements and how they are intended to control and/or mitigate risks at the physical and organisational interfaces was not explained, nor was it clear how these arrangements would be adapted for the licensee.

800. Combining codes and standards incurs potential risks that need to be managed through-out the plant life. To illustrate, for Option 1, potential risks included:

- The use of different QA systems during SG design, manufacture and plant construction, and operation with attendant differences in the responsibilities of the organisations involved leading to potential inadequate control arrangements or significant 'gaps' in the organisational responsibilities.
- Whether the interrelationships between RCC-M and RSE-M have been considered.
- The provisions for the EIMT and how the design intent is maintained through-life.
- The need to ensure adequate arrangements to apply RGP and how the design intent is maintained through life (e.g. repair, OPEX and demonstrating ALARP).
- Whether the RP has fully captured the scope of the additional measures expected to underpin its HIC claims for SG components, how these will be implemented, and whether there are adequate control arrangements to support the licensee.

801. I considered that the uncertainties relating to the adequacy of the RP's consideration of relevant risks relating to the SG code provisions and their mitigation represented a potential shortfall in the safety case for the UK HPR1000 SG. As this is setting a precedent in the UK for a component subject to a highest reliability claim, I raised Regulatory Observation RO-UKHRP1000-0033 to assist the RP in providing adequate evidence of its process, and to ensure that the risks associated with the RP's preferred SG code option for the UK HPR1000 are reduced to ALARP (Ref. 49).

802. RO-UKHPR1000-0033 included the following actions:

- ROA1 – Process for the consideration of SG code relevant risks.
- ROA2 – SG code provisions and mitigation of relevant risks, such that the RP should provide:
    - ROA2.1 – A demonstration that SG code relevant risks with the potential to affect the achievement of the design intent have been considered and that an adequate highest reliability claim for the UK HPR1000 SGs can be provided with relevant risks avoided, or reduced to ALARP.
    - ROA2.2 – An explanation of the process for taking account of the impact of future changes in the design, manufacturing and inspection provisions of the proposed codes e.g. ASME III/XI and RCC-M/RSE-M, on the UK HPR1000 SGs, and how these will be captured in the safety case.

803. In response to RO-UKHPR1000-0033, the developed a resolution plan (Ref. 49) which detailed several key steps that the RP would undertake:

- Documenting the approach to identifying the SG Code relevant risks and the key steps in the process.
- Identifying the codes and standards jurisdiction of ASME/RCCM/RSE-M.
- Identifying the key code differences between ASME and RCC-M/RSE-M relevant to the SG, including key difference between the codes related to design, manufacture, plant construction, pre-service inspection and quality assurance, including OPEX where applicable.
- Assessing the risks associated with any difference identified.
- Provisions of options to mitigate and reduce any identified risks to ALARP.
- Details of the RP's forward strategy and approach for maintaining oversight of applicable and concurrent code changes going forward.

804. The main documentation that formed the basis for my assessment is presented in the RO-UKHPR1000-0033 closure assessment note (Ref. 212). The key ONR SAPs applied in my assessment were ECS.2, ECS.3, EMCs 1-4 (Ref. 2), and the associated TAG (ONR NS-TAST-GD-016 - Ref. 5). I also consider relevant national and international standards (Ref. 213) (Ref. 11). The details of my assessment are recorded in the RO-UKHPR1000-0033 assessment note (Ref. 212) a summary of the key points and conclusion is provided below.

Review of ROA1-3 of RO-UKHRP1000-0033

805. I raised ROA1 to seek a further explanation of; the RP's process for identifying the SG code relevant risks, the topic areas considered, and their significance in terms of ensuring the adequacy of the SG code provisions for the UK HPR1000. It was my expectation that the RP's response should include, but not be limited to physical interfaces/code jurisdictions, QA and organisational responsibilities, EIMT provisions, and the organisational arrangements to ensure that a highest reliability case for the high integrity components in the SG can be developed.

806. In response to RQ-UKHPR100-1406 (Ref. 28) the RP provided a further explanation of its process for identifying and mitigating the potential risks when using a combination of design and construction codes for the UK HPR1000 SG. This was based on the five step process outlined in the resolution plan, but also included the links to the supporting evidence provided by the RP (Ref. 214) (Ref. 215) (Ref. 216) (Ref. 217) (Ref. 101). The process was developed from the successful application of similar approaches in China.

807. I was satisfied that the role of the supporting documentation in the RP's safety case was clear. I noted that the RP had considered key differences (gaps) between the US and French codes. The scope covered design, material procurement, fabrication, welding, non-destructive testing, and PSI. However, I was unclear how risks were identified, and if necessary, supplementary measures determined. The RP explained in the response to RQ-UKHPR1000-1413 (Ref. 28) that having identified key differences between the codes i.e. measures in RCC-M that do not feature in ASME, additional measures from RCC-M are adopted to supplement ASME and reduce risks. The approach was a requirement of the NNSA, with the identification of the key differences facilitated by a design review meeting to identify the measures to supplement ASME compliance to reduce risks.

808. In my opinion the RP has developed an adequate process for identifying the relevant SG code risks. Notably, the development of the SG code provisions is not based on combining or mixing the provisions of established US and French nuclear design and construction codes rather it is founded on supplementing full compliance with ASME design and construction rules with additional measures from RCC-M to achieve in broad terms a common baseline with other major vessels for the UK HPR1000 and in doing so reducing relevant risks.

809. A further step in the RP's SG code risk assessment process included the consideration of risks in SG design, manufacture, examination, installation, QA, EMIT, and operation along with additional measures if appropriate to reduce the risk to ALARP. The RP's evidence was provided in two submissions covering 'The QA Requirements for SG Design, Manufacture and Plant Construction' (Ref. 214) and the 'The Risk Analysis and Mitigation Measures of SG Codes' (Ref. 215). For ease of presentation, I cover the potential QA risks and component interface issues under ROA 1 as these topics featured in RQ-UKHPR100-1406 (Ref. 28). The wider topic of the identification, analyses, and mitigation of relevant risks in establishing the SG code provisions is addressed under ROA2 below.

810. In my opinion there are potential risks if there are 'gaps' in the QA provisions or organisational responsibilities, particularly at physical/ organisational interfaces. I therefore sampled the evidence submitted by the RP in its QA requirements document (Ref. 214). I noted that for Fangchenggang-3 there is a key role for the NNSA in overseeing the implementation of the QA arrangements and reviewing or approving non-conformances. I made the RP aware that in accordance with UK law, this responsibility lies with the licensee. Indeed, the QA documentation would need to be more explicit in terms of organisation and responsibilities if it was submitted post GDA, for instance, to include the role of licensee, Design Authority, and with consideration of the intelligent customer capability etc as per RQ-UKHPR1000-1406 (Ref. 28).

811. I also made the potential manufacturers of the major vessels and components for the UK HPR1000 aware of these expectations during a visit to China in January 2020 (Ref. 218). For the GDA the RP confirmed its understanding of the principles for design and manufacturing control to meet UK expectations in its response to RQ-UKHPR1000-0578 (Ref. 28). This principles for manufacturing control are discussed further in Sub-section 4.5.1 of my report.

812. In consultation with ONR's MSQA discipline, I issued RQ-UKHPR100-1406. My key questions covered the identification, if any, of potential risks relating to the QA arrangements and their mitigation, and any specific requirements relating to the QA arrangements imposed by the NNSA (Ref. 28). The RP explained the role of the QA requirements document in the safety case for the UK HPR1000 SG and how the evidence provided features within the process for addressing the SG code relevant risks. In addition, for the UK HPR1000 SG, similar QA provisions to that provided for Fangchenggang-3 SG are proposed including, ASME QA requirements, HAF003 aspects of Chinese practice without NNSA oversight (Ref. 219), ESPN Guideline No.8 (Ref. 220) (and other aspects of the RCC-M code mandated by the NNSA), along with the UK expectations for HIC. Thus, provided these QA provisions are implemented in practice no significant risks were identified for the UK HPR1000 SG.

813. According to Fig. F-2.5-1 of the 'Design Specification of Steam Generator' (Ref. 217), there are four types of nozzles on the SG, where transitions between the ASME code designed SG connects to RCC-M code designed pipework. Whilst the query regarding design code jurisdiction is relevant for all of these nozzle locations, I consider the interface between the SG primary inlet/outlet nozzles to be the most challenging transition to manufacture, on account of the dissimilar metal welds required between the two components. I therefore selected this particular weld as the focus of my query, to determine whether the RP has adequate arrangements in place to manage design code jurisdictions and organisation responsibilities at the interface locations.

814. I noted that the nozzle to safe-end buttering and welds along with the safe-ends are included as part of the SG supply. The materials were from RCC-M specifications, but I was unclear whether these features were subject to either ASME or the RCC-M provisions for QA, design, manufacture, and inspection. A related point was whether the welding was to ASME with materials from RCC-M, and if so whether the potential risks associated with the different set of dissimilar metal welds had been considered,

and if necessary, mitigated. I also wanted to establish how in meeting NNSA requirements if certain provisions from the RCC-M code are used to supplement ASME, how the use of non-ASME codes for design, manufacture and inspection for this part of the SG supply, is delineated and controlled with the SG supplier.

815. The RP clarified the physical and organisational jurisdictions; the parent material (including SG nozzle) and welding material in the dissimilar metal weld are specified to the ASME code, whereas the safe end material is specified to the RCC-M code. The welding process for the dissimilar metal weld was to the ASME code though the safe-end material is an RCC-M specification. Inspection of the dissimilar metal weld is also to the ASME code, but these are supplemented with additional NDE requirements during manufacture including a PSI in accordance with the RSE-M code.

816. For both the nozzles and safe-ends the code provisions would be supplemented with those from Chinese Regulatory Practice, HAF003, (Ref. 219) and the UK expectations (e.g. HIC). The choice of the code provisions had also sought to mitigate relevant risks by reflecting the suppliers experience and proficiency with the relevant design and construction code.

817. I was satisfied that the RP had provided adequate information to clarify the location of the physical and organisational interfaces arising at the connections between the nozzles and the safe-ends. I was also content that the RP has explained, how through the application of its process, they propose to manage any relevant risks arising from these interfaces.

818. Nonetheless, there are risks associated with the introduction of a different set of DMWs introduced in the SG design provisions because the nozzle and weld materials, along with the welding procedures, differ to those of the comparable interfaces in the other major vessels e.g. the RPV. For this aspect, I do not consider that sufficient information has been provided within GDA to fully demonstrate how this meets ONR expectations in ONR SAP EMC.14 (Ref. 2) that “manufacture and installation should use proven techniques and approved procedures to minimise the occurrence of defects that might affect the integrity of components or structures” (Ref. 2). The implicit assumption by the RP is that appropriate ISI data will be available to underpin the SG safety case i.e. there are no gaps in the inspection coverage due to the different code jurisdiction interfaces between the SG vessel and piping.

819. The licensee should consider the points above to inform the PSI/ISI proposals for DMWs at code boundary locations. This should ensure that the inspections and testing proposed are suitable and commensurate with reducing risks ALARP, given the different welding procedures, material differences and taking cognisance of the SI classification e.g. HIC claim. I have raised an AF below for the licensee to address during the site-specific stage. In addition, it is important that the licensee considers developments in ASME XI to ensure safety claims remain relevant and up to date.

AF-UKHPR1000-0207 – The licensee shall, as part of detailed design, demonstrate the inspections and testing of interface regions for dissimilar metals welds reduce risks to as low as reasonably practicable, where the approach of combining codes and standards is justified.

820. The RP also confirmed that they intend to follow Chinese practice with respect to the SG code provisions by supplementing full compliance with ASME design and construction rules with additional measures from RCC-M and RSE-M. This is an NNSA requirement with the aim of assuring that the major vessels in the HPR1000 are designed, manufactured, and procured to similar standards and there is a common baseline for ISI in the UK HPR1000 which is to RSE-M. I was content with this position on the basis that the SG code provisions for the UK HPR1000 should be no less than the country of origin of those provisions.

821. The RP committed to updating its QA requirements document to reflect the clarifications and conclusion provided in the response to RQ-UKHPR1000-1406. I am content that in principle that the RP responses have met the intent of ROA 1. I consider the completeness of the RP's capture of its process in the safety case documentation further under ROA2.

822. I considered that the response to ROA2 should include information on the SG code provisions along with measures taken to eliminate or reduce SG code relevant risks to ALARP. In particular, the SG code provisions needed to underpin the HIC classification i.e. how further measures beyond compliance with established nuclear design and construction codes will be implemented and controlled during design, manufacture, and installation to support the licensee.

823. There was also a need to explain the process for considering revisions to the design and construction codes. It was also my expectation that any such response would also demonstrate how the claims, arguments and evidence underpinning the RP's position and mitigation measures would be documented in the safety case for the UK HPR1000.

824. In response to ROA1 and ROA2.1, the RP submitted its 'Risk Analysis and Mitigation Measures of SG Codes' (Ref. 215). A companion submission, 'The SG Codes Relevant Risks Analysis and Assessment Report (Ref. 216), provided the response to ROA1 and ROA2.2.

825. I issued RQ-UKHPR1000-1413 (Ref. 28) with my key questions targeted at the completeness of the RP's identification/mitigation of relevant risks and clarifying my understanding of the RP's SG code provisions. My questions covered:

- the provisions for overpressure protection and operation
- the design provision for EIMT activities.
- confirming whether the RP's proposals to comply with the French ESPN order included the associated ESPN regulatory guides.

826. I was also unclear which code rules applied to SG design features and activities that are common to the UK HPR1000 plant e.g. overpressure protection, plant operation and EIMT activities. I wanted to confirm that the RP's proposals were appropriate and commensurate with reducing risks to ALARP. The RP explained that though the UK HPR1000 SG is designed to ASME, there is an additional requirement to ensure the overpressure protection provisions are not less than those of RCC-M/RSE-M. This is because the SG is part of the Main Steam System which is protected to RCC-M requirements. Similarly, the SG in common with the UK HPR1000 primary and secondary systems are operated to the rules of the RCC-M code as specified in the 'Definition of Plant Normal Operating Domain' (Ref. 221).

827. The RP clarified that there was adequate access to facilitate EIMT activities and the specific arrangements for the SG would be captured in the SG Equipment Operation and Maintenance Manual (EOMM) at the license stage. For GDA, the RP's intentions were held to be captured in several submissions provided in response to RO-UKHPR1000-0021 (Ref. 49) (Ref. 222) (Ref. 223) (Ref. 224). I have reviewed the referenced documents from a structural integrity perspective and confirmed that inspection of the SGs is accounted for accordingly within them. I have assessed the RP's generic PSI/ISI strategy (Ref. 223), in more detail under Sub-section 4.6.2 of my assessment report.

828. A further point relating to EIMT is associated with the design life of the SG and RCP (see Sub-sections4.4.1.2. and 4.4.2.2), which will depend on a fatigue analyses to ASME III rather than RCC-M. Notably, I identified the potential residual risks that will need to address the influence of the PWR water on the RP's fatigue analyses and

hence SSC design life (as discussed in paragraph 738 above). Thus, if the use of a combination of design codes is invoked for the SG (and RCP), it will be necessary to demonstrate that the design life has accounted for the detrimental effects of the PWR environment at the site-specific stage. I have raised AF-UKHPR1000-0203 to track the closure of this matter.

829. The RP confirmed that compliance with the ESPN order also included compliance with the French Autorité de Sûreté Nucléaire (ASN) Guideline No.8, which explains the principles and conditions under which the notified bodies and operator-specific inspection organisations are approved by ASN for the conformity assessment of nuclear pressure equipment and assemblies in line with the ESPN order (Ref. 225). The RP also referenced its response to RQ-UKHPR1000-0886 to explain that oversight of compliance would be undertaken by the Independent Third-Party Inspection Agency (ITPIA) for HIC and SIC-1 structures and components (Ref. 28).

830. I noted that under Chinese practice, the regulator (NNSA) may supervise and become involved in the decision making relating to design and manufacturing control. In contrast, for the UK, and in accordance with UK law, the licensee is responsible for nuclear safety, and so the emphasis would be on the licensee to develop and implementing adequate arrangements for design and manufacturing control including compliance with the ESPN order. ONR would oversee the process to ensure that the licensee developed and implemented adequate arrangements, as described in RQ-UKHPR1000-0886 (Ref. 28).

831. The final step in the RP's SG code risk assessment process is intended to draw conclusions on the SG code risks and to capture any relevant expectations relating to the UK context. In RQ-UKHPR1000-1420 (Ref. 28), I sought further clarification of the evidence provided in the 'The SG Code Relevant Risks Analysis and Assessment Report' (Ref. 216). My main questions covered:

- the proposals for PSI.
- the totality of the SG SI code provisions to achieve the RP's requirements for its safety case including meeting UK expectations.
- the role and documenting of the RP's SG code risk assessment process in the safety case for the UK HPR1000 SG.

832. In China the NNSA imposes additional measures during SG manufacture. A key additional measure includes undertaking a PSI to both ASME and RSE-M requirements. It was my understanding that a PSI fingerprint to ASME XI would be undertaken on completion of manufacture and post the hydrostatic test prior to shipment of the SG. In addition, a further PSI fingerprint to RSE-M is completed after installation in the UK HPR1000 plant. However, the RP also claimed "in addition, PSI according to ASME code Section III will also be performed after hydrostatic test of SG at the shop."

833. I therefore sought clarification of the scope of the inspections to ASME and, that in addition to manufacturing inspections, the intent is to provide a PSI fingerprint prior to installation and service with the scope of the PSI to ASME XI rather than ASME III.

834. The RP subsequently confirmed that all required volumetric PSI after the hydrostatic test of SG would be performed according to ASME Section III NB5410, and the requirements of ASME Section III NB5280, which refers to ASME Section XI. Furthermore, a separate PSI to RSE-M is implemented on site following SG installation. This site inspection supplements the ASME XI PSI to achieve a baseline (PSI) inspection of the UK HPR1000 SG to RSE-M as described in the RP's response to RQ-UKHPR1000-1420 (Ref. 28).

835. Thus, the RP's, proposals for the PSI were diverse and comprehensive, such that provided they were implemented rigorously, they would undoubtedly afford additional confidence in the integrity of the UK HPR1000 SGs, irrespective of the additional measures expected to underpin a highest reliability claim.

836. Regarding the other aspects of ROA2 (Ref. 216), is also intended to provide an overall demonstration that:

- the SI provisions for the UK HPR1000 SGs are adequate i.e. that all relevant risks resulting from the use of US (design, manufacture and inspection) and French (in-service inspection) codes have been considered, and, where appropriate, mitigated; and
- the totality of the SI provisions for the SG are adequate to achieve the RP's requirements for its safety case i.e. the level of integrity demonstration is commensurate with the SI classification and implied reliability.

837. For the first demonstration, the RP committed to updating its safety case documentation for the UK HPR1000 SG within the GDA to reflect the clarifications provided in response to RQ-UKHPR1000-1406, RQ-UKHPR1000-1413 and RQ-UKHPR1000-1420 (Ref. 28). In accordance with my sampling strategy for checking safety case consolidation under Sub-section 4.8 of my assessment, I reviewed two of the documents produced to support RO-UKHPR1000-0033 (Ref. 49). I was able to confirm that the changes had been completed in accordance with the commitments made in the relevant RQs. This is recorded in Table 13 below.

838. For the second demonstration, the RP explained its process for establishing the totality of the SI provisions for the UK HPR1000 SG. This included supplementing the approach detailed in the 'SG Codes Relevant Risks Analysis and Assessment Report' (Ref. 216) with further steps to meet UK expectations.

839. Informed by the response to RQ-UKHPR1000-0677 on Nuclear Pressure Equipment and a commitment made therein (Ref. 28), the RP outlined the approach to establishing the totality of the SI provisions for the UK HPR1000 SG. The approach was based on understanding the key differences (deltas) and then adding additional measures to meet either requirements or expectations. In brief, the totality of the SI provisions proposed for the UK HPR1000 SG comprised:

- ASME + supplementary measures from RCC-M /RSE-M (Chinese Practice)
- HIC measures (DTA, TJs, Fracture Toughness testing, ITPIA) (UK expectations)
- Conformity assessment (PE(S)R 2016 equivalence + ASN Guide 8, ITPIA) (UK expectations)

840. I have discussed supplementing compliance with ASME with additional measures from RCC-M/RSE-M and the measures to underpin a highest reliability claim above. For conformity assessment, the RP provided a compliance analysis which was detailed in Conformity Assessment of ASME III (for SG) and ASME VIII with PE(S)R 2016, described in the RP's response to RQ-UKHPR1000-0463 (Ref. 28). In this submission the differences between ASME III and PE(S)R were identified, and the assessment concluded that the equivalence of PE(S)R 2016 requirements can be met.

841. In summary, the RP provided an adequate response to RQ-UKHPR1000-1413 (Ref. 28) to provide a basis for confidence that their identification, and were appropriate mitigation, of relevant risks relating to the SG code provisions was suitable and sufficient. I was also content that the RP had developed an adequate process that would allow the totality of the SI provisions to be determined for the UK HPR1000 SG. Notably, the outputs took cognisance of both Chinese practice and UK expectations.

842. A common question raised in my review of the RP's key submissions in response to RO-UKHPR1000-0033 (Ref. 49), related to the role of the document (evidence) and the traceability of the conclusions drawn in the safety case for the UK HPR1000 SG. The RP acknowledged the lack of coherency in the SG safety case and responded with further explanations of the purpose of the evidence documents, the linkage between the submissions and the document hierarchy. The RP also committed to update the safety case documentation for the UK HPR1000 SG in response to the clarifications provided in response to RQ-UKHPR1000-1406, RQ-UKHPR1000-1413 and RQ-UKHPR1000-1420 (Ref. 28) within the GDA.

843. In ROA2.2, I also sought an explanation of the process for taking account of the impact of future changes in the design, manufacturing and PSI/ISI provisions of the proposed codes e.g. ASME III/XI and RCC-M/RSE-M, on the UK HPR1000 SGs, and how these will be captured in the safety case. As this is a generic topic, the position for the UK HPR1000 SG is addressed under Sub-section 4.4 (Code Versions).

Conclusions on the SG Code Proposals and Risks (RO-UKHPR1000-0033)

844. The RP provided an adequate explanation of its process for identifying and mitigating relevant risks arising from supplementing the SG design and construction code (ASME) with additional measures from the RCC-M and RSE-M codes. These additional measures are derived from a design review and meeting the requirements of the NNSA. I am also content that the RP has explained the linkage between its process and the evidence provided in its submissions in response to RO-UKHPR1000-0033 (Ref. 49).

845. The RP provided adequate information to clarify the location of the physical and organisational interfaces arising at the connections between the nozzles and the safe-ends.

846. The RP provided an adequate basis for confidence that their identification, and where appropriate mitigation, of relevant risks relating to the SG code provisions was suitable and sufficient.

847. I am also content that the RP had developed an adequate process that would allow the totality of the SI provisions to be determined for the UK HPR1000 SG. Notably, the SI provision developed for the UK HPR1000 SG take cognisance of both Chinese practice and UK expectations (UK legislative requirements and the highest reliability).

Use of OPEX

848. The RP's SG design and manufacturing code proposals for the UK HPR1000 SG draw on the collective experience for the SGs in the Chinese Pressurised Water Reactor CPR-1000 PWR fleet of civil reactors, where a combination of US and French codes for the SG provisions is extensively used (see RQ-UKHPR1000-0030 (Ref. 28) (Ref. 101). The CPR1000 and Fangchenggang-3 PWR designs differ, but the OPEX relating to the control of risks e.g. different management responsibilities, QA systems and the physical component interfaces is deemed relevant to the RP's proposal for the UK HPR1000 SGs.

849. I am aware that beyond the Chinese operating experience (OPEX) there are some specific precedents for using combined codes and standards. The most relevant, taking cognisance of the need to demonstrate high levels of structural integrity was the replacement of the North Anna, Unit 2, RPV Head in January 2003, the first time a major reactor component fabricated to the French RCC-M Code was used in a US PWR (Ref. 216).

850. The scope of the RP's review of the 'gaps' between the US and French design and manufacturing codes had clearly been informed by the North Anna, Unit 2 OPEX (Ref. 216). Nevertheless, the licensee will also need to be cognisant of developments in both US and French codes and be aware of relevant operational experience (OPEX).

Contingency Planning for UK HPR1000 SG Design

851. In reviewing the RP's optioneering for the selection of its proposed design codes for the SG (Ref. 209), I observed that several of the possible options including the RP's preferred option were predicated on the availability of the RP's chosen SG supplier. Thus, if there were difficulties with the availability of the preferred SG supplier there is the potential for a significant future risk to affect the design provisions for a highest reliability component in the UK HPR1000. This is primarily an issue for addressing in the site-specific stages, however in accordance with the intent of GDA (i.e. to avoid surprises and minimise key risks), I raised RQ-UKHPR1000-0722 (Ref. 28). My main points of clarification related to gaining further assurances that RP had adequate contingency arrangements in place along with a basis for confidence that ONR's expectations could still be met.

852. The RP confirmed that they have the necessary agreements to work with its preferred SG supplier, namely BWXT of Canada and given the good working relationship between the organisations, risk of difficulties is low. Nonetheless, as a contingency, CGN indicated that they have the competence and experience to design, develop and manufacture a new SG, if needed. The RP cited a new model SG which is undergoing approval and manufacture in China along with the extensive manufacture experience accumulated in China, were they had manufacture many SGs to meet a wide range of designer specifications.

853. The RP also identified a number of high-level requirements that would need to be met for a re-designed SG, these included the code requirements as well as assessing the implications for the layout and the plant performance. The RP also highlighted the need to meet UK expectations relating to ALARP, materials selection, the expectations for highest reliability, UK legislative requirements, conformity assessment and the role of the ITPIA.

854. The procurement decisions are primarily a decision for the licensee, but I am satisfied for the purposes of the GDA, that the RP has provided adequate evidence that other SG design options are available, there is a basis for confidence in their achievement, and that there is an adequate understanding of UK expectations to inform the design, development and manufacture of an alternative UK HPR1000 SG design, in the event that its preferred SG design and/or supplier is unavailable.

4.4.2.3 Reactor Coolant Pump Code Provisions and Mitigation of Relevant Risks

855. As previously discussed, through the resolution of RO-UKHPR1000-0033 (Ref. 212) ONR sampled the RP's arrangements for managing component qualification and justification of the component design against two different nuclear design codes. During engagements within Step 3 of GDA, it became apparent that the RP was also proposing to qualify the design and manufacture of the RCP against two different nuclear design codes. It is the RP's intent to follow the same approach of demonstrating safety for the RCP as has been developed and provided for the SGs.

856. In accordance with the RP's SI classification process, the RCP is designated as HIC, the same as the SG. As a HIC, the scope and depth of sampling conducted for the SGs in order to resolve the actions raised under RO-UKHPR1000-0033 (Ref. 49) has been considerable. The conclusions drawn from resolution and assessment of RO-UKHPR1000-0033 confirm that the RP's approach to managing risks associated with a mixed code approach is adequate, with a number of assessment findings raised.

Whilst my judgement on the RP's approach for managing risks associated with combining codes for the SGs provides confidence, I am cognisant that there are significant design and manufacturing differences between the two components. I therefore decided to sample the application of the RP's approach taken for the RCPs, rather than adequacy of the approach itself.

857. Given the importance of the RCPs for nuclear safety and commensurate HIC classification, I consider it reasonable within GDA that a commensurate level of safety demonstration should be presented for the RCPs, as has been provided for the SGs. ONR guidance in ONR SAP ECS.3 and Paragraph 173 (Ref. 2) is relevant in this instance: 'The combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated.'

858. I sought assurance from the RP that the same level of scrutiny and consideration for reducing risk ALARP for the RCP had been applied, as per the SGs under RO-UKHPR1000-0033 (Ref. 49), as detailed in RQ-UKHPR1000-1097 (Ref. 28).

859. The RP's response to my RQ confirmed that the evidence presented to justify for the RCP nuclear design and construction code selection would take the same format as has been assessed for the SGs. These documents record the identification, assessment and select process for RCP codes and standards.

860. I selected a number of these documents to sample, to confirm that the same level of detail and scrutiny had been applied to support the combining of codes as had been presented satisfactorily for the SGs. The RP has followed a stepwise approach to first presenting the safety case, starting with a high level ALARP review of RCP codes and standards (Ref. 226).

861. The RP clearly states in the ALARP review that the RP's 'partner' company (Shanghai Electric-KSB Nuclear Pump & Valve Co., Ltd - SEC-KSB) has been selected to undertake manufacture of the RCPs for the UK HPR1000. For this reason, the ALARP review has been undertaken by the RCP supplier, rather than RP itself.

862. The ALARP approach taken seeks to identify, compare and contrast options for reducing risk. This starts by consideration of what the RP's claims is relevant good practice (e.g. nuclear design, construction and inspection codes) and any available/relevant OPEX. Three options are identified, which include:

- Option 1 - design and construct the RCP according to ASME with additional requirements from RCC-M and RSE-M codes.
- Option 2 - design and construct the RCP according the ASME code.
- Option 3 - design and construct the RCP according the RCC-M and RSE-M codes.

863. The options are then subject to a number of assessment aspects related to safety, cost and timescale for implementation, for which the former category is most heavily weighted. Overall, Option 1 scores the highest, therefore risk is reduced ALARP.

864. In scrutinising the reasoning provided to support this decision, I am not satisfied that the decision-making process has been fully informed and is unbiased. Firstly, the OPEX presented is limited to a few plants in China. In my opinion, an OPEX review should be more comprehensive, in order to fully account for all known benefits and disbenefits of the options presented.

865. Secondly, one of the main safety aspects considered (which carries a 20% weight for the overall scoring criteria) is 'Engineering Risk'. Option 3 yields a low score for this

aspect, on the basis that "SEC-KSB have no experience and OPEX about RCP design and construction according the RCC-M and RSE-M. This will cause significant engineer risk, implementation cost and implementation time*"*. Whilst this may be the case, it is an evaluation criterion that is supplier specific and as such, I do not consider it to be fully 'generic'. This approach appears to be focused on the supplier's experience of manufacturing the RCP in accordance with ASME, however it is not clear whether the risk that will be taken on by the licensee to demonstrate compliance of a safety significant component to multiple codes is warranted by the benefit gained by a single supplier's familiarity of a code. This will also bound the safety case and the licensee to using pumps supplied by SEC-KSB. Should any change in supplier occur post GDA or during the site-specific stages, the licensee may incur a significant time and cost burden to re-write the safety case and repeat the design selection process for a different supplier, should that supplier choose a design code that matches the majority of the RCS. In my opinion, this isn't a significant risk to safety of the plant, however it has the potential for a high administrational and financial risk later in the construction phase.

866. Overall, I am satisfied that the RP has selected an appropriate nuclear design and construction code for the RCP. I am also satisfied that the RP has appropriate arrangements in place to identify, review and manage the risk associated with combining multiple codes for the RCP. I consider the lack of a broad OPEX review and unbiased optioneering process to be a minor shortfall, which should be developed and revisited during the site-specific stages, to show that the RP's approach to demonstrating compliance with multiple codes and standards for the RCP is robust and justified.

867. I am cognisant that a number of assessment findings and minor shortfalls have been raised for the SG associated with the use of combined codes. Where these are generic to the RP's approach of managing risks for combining codes (i.e. rather than those which are SG specific), I consider these apply equally for the RCPs, and should be addressed accordingly.

4.4.3 Additional SI Provisions

4.4.3.1 Hydrostatic Testing Policy

868. During Step 2 of GDA, the RP indicated that the 10-year requalification hydrotest of the primary circuit as specified by the RSE-M code will be applied for the UK HPR1000. It was ONR's view that the hydrostatic proof test performed at 10-year intervals may be of limited value to the structural integrity case when balanced against the increased radiological risk of performing such a test and the potential for introducing damage, for example, tearing or increased fatigue crack growth (Ref. 6).

869. It is ONR's view that periodic hydrostatic tests are of limited benefit and there exists a risk that failure during the test could lead to a release of contamination within containment and damage to surrounding equipment (Ref 2). On this basis ONR is of the view that there is not a strong ALARP basis to perform the periodic hydro tests.

870. ONR also noted within 'Technical Specification of Reactor Pressure Vessel Workshop Hydrotest' (Ref. 227)*,* two monitoring requirements (strain measurement and non-destructive testing post hydrotest) are to be removed for the UK HPR1000 RPV post the workshop hydrotest. It is ONR's expectation that where requirements guard against risks in manufacture these tests should be performed, where reasonably practicable.

871. I therefore needed a further basis for confidence that removing these monitoring requirements for the RPV was commensurate with reducing risks to ALARP, which included the consideration of what risks are being controlled by their application, what burden (time and effort) is incurred in performing these activities, and what access the

licensee would have to influence the information collected and records generated during the operating life of the UK HPR1000. I raised RQ-UKHPR1000-0651 to progress the position (Ref. 28).

872. The RP explained that for the reference plant (Fangchenggang-3), periodic hydro-tests during service are required according to the requirements of RSE-M and Chinese practice (Ref. 15). However, for the UK, the RP recognised that the periodic hydraulic test during service is not common practice because they are held to be of limited benefit and there is a risk that failure during the test could lead to a release of contamination within containment and damage to surrounding equipment (NS-TAST-GD-016, Revision 5, paragraphs 5.47 and 5.60, Ref. 5). In view of the ONR expectations, the RP agreed to cancel the periodic hydrotest and to amend the CSRs accordingly, as discussed in the RP's response to RQ-UKHPR1000-0651 (Ref. 28). I sampled the updated CSRs for the RPV, SG, RCP and SL and confirmed that these submissions reflected the commitments made in responses to RQ-UKHPR1000-0651 and RQ-UKHPR1000-1413 (Ref. 28).

873. In the absence of a periodic hydrotest the RP confirmed in its responses to RQ-UKHPR1000-0651 and RQ-UKHPR1000-1413 that the mechanical analysis, PSI post manufacture, along with ISI during service would be used to underpin confidence in the integrity of the component (Ref. 28). In addition, the 'Leak Monitoring System' is designed to monitor the leakage in the MCL, SL and the MSL to underpin the sealing performance.

874. Regarding the proposed removal of certain monitoring requirements during the workshop hydrotest, the RP clarified that the purpose of the strain measurement during the workshop hydrotest is to obtain the strain data of the typical parts of components for comparison with the mechanical analyses. The code requirement for strain measurements is applicable for the first of a kind for a series of units, and so strain measurement data for subsequent units are no longer required. Furthermore, the reason for cancelling the NDE post the workshop hydrotest was because the RCC-M code has no requirement for NDE after the RPV workshop hydrotest, but as the RPV is a non-replaceable component, and to further verify the component quality and the absence of manufacturing defects, the designer added the repeat inspection requirements, mainly additional manual UT, after the RPV workshop hydrotest.

875. However, for the UK HPR1000 RPV and other HIC components such as the PZR, SG, RCP and MCL, objective based high reliability NDE, will be undertaken beyond the design code-based inspections to underpin the avoidance of fracture demonstration. The high reliability NDE includes inspection qualification along redundant and diverse inspections at the end of manufacture to show that RPV is free of defects that could threaten their structural integrity through the lifetime of the plant. These high reliability NDE inspections for HIC components supersede the requirements imposed by the designer post the workshop hydrotest to assure the absence of manufacture defects for the major components of the reference plant.

876. I was satisfied with the RP's reasoning that for HIC components, the objective-based high reliability inspections would supersede the additional inspections proposed by the designer for Fanchenggang-3. This notwithstanding, the position for SIC-1 to SIC-3 structures and components was less clear from the response to RQ-UKHPR1000-0651 (Ref. 28). Notably, if additional manufacturing inspections are implemented in Fanchenggang-3 for other safety important SSC (non-HIC), particularly if they are non-replaceable e.g. typically these may include the RVI, there is a question as to whether it is ALARP undertake these additional manufacturing inspections post the workshop hydrotest. As ONR has a policy of in general not accepting a lower standard of demonstration than the country of origin, I will raise an assessment finding for the licensee to progress this matter during the site-specific stage.

AF-UKHPR1000-0208 – The licensee shall, as part of detailed design, identify any additional non-destructive examination requirements beyond the design code, imposed by the designer for structural integrity class 1 to 3 components, post the workshop hydrotest in the UK HPR1000 reference design. The licensee shall also establish whether it is commensurate with reducing risks to as low as reasonably practicable to implement any additional non-destructive examination for the UK HPR1000.

4.4.3.2 Structural Integrity Provisions for UK Grid Code Compliance

877. Throughout the course of my GDA Step 4 assessment, I was informed by the ONR Electrical Engineering specialist of a claim made by the RP on the UK HPR1000's resilience to sudden power demand changes in response to UK grid code requirements.

878. At a multi-discipline technical meeting, the RP presented a strategy to demonstrate the design reference can meet the UK grid code requirements in respect of extended low power operation (ELPO), and in part, the frequency response (FR) requirements (Ref. 228). The RP also set out how it intends to demonstrate during GDA that a viable option exists to fully meet the FR requirements of the grid code. This is referred to by the RP as "Step 1 – ELPO and 3% FR" and "Step 2 – 10% FR" (Ref. 229). The RP intends to underpin "Step 1" through the extant design reference safety case. However, for "Step 2" the RP has provided a high-level submission that presents qualitative judgements covering the potential implications for several technical disciplines, including SI (Ref. 230). In summary from an SI perspective, the changes to meet UK grid compliance result in an increase in the low power cycling frequencies.

879. ONR raised RQ-UKHPR1000-1451 to seek further clarification of the RP's strategy, deliverables, and timescales for the GDA (Ref. 28). The qualitative judgements underpinning the implications for structural integrity, are outlined in Paragraph 5.2.2.4 of 'Analysis of the Potential Gaps due to Grid Code Requirement' (Ref. 230). These judgements included consideration of the potential implications for demonstrating achievement of design code stress limits, fracture analyses, fatigue life and DTA. The RP' overall claim was that the potential changes associated with the transient profile and frequencies are unlikely to be significant for structural integrity.

880. I recognised that at this stage the claims for structural integrity are based upon qualitative arguments. However, much of the supporting evidence was detailed in Tier 3b documents, which had not been fully translated and so insufficient evidence was presented in the 'Analysis of the Potential Gaps due to the UK Grid Code Requirement' docuemnt (Ref. 230) to underwrite the RP's SI claims. I raised RQ-UKHPR1000-1461 (Ref. 28) to gain further clarification of the underlying evidence with my key questions covering:

- The basis for the view that the changes are bounded by the Step Load increase/decrease of 10% FP along with estimate of the increase in the CUF for the limiting locations in the RPV and MCL.
- The rationale for selecting the RPV inlet nozzle for the assessment of the fatigue considerations (CUF of 0.23), and clarification of whether other potentially challenging locations in the RPV with high CUFs at or above 0.75 would be assessed in GDA (Ref. 195).
- An explanation why the selection to the Charging nozzle location on the MCL, (CUF of 0.78), is held to bound other locations relating to the MCL, where high fatigue usage might be expected e.g. the surge line connections to the pressuriser and MCL.
- Further substantiation to support the view that the changes in the transient loadings and frequencies to achieve UK grid code requirements will not introduce adverse effects for the fast fracture analyses (part of design code

compliance) and the defect tolerance assessment (DTA) which underpin the RP's avoidance of fracture demonstrations for HIC components.

881. The RP's response did not fully address my queries, for example, the basis for the use of the step load increase/decrease of 10% as the bounding transient nor did the RP fully explain the basis for confidence that adequate fatigue usage factors would be maintained for other locations in the RPV and branch piping. I also observed that inconsistent conclusions were drawn in 'Impact Analysis of Grid Code Compliance on Existed DTA Reports and Stress Analyses Reports' (Ref. 231) and 'Analysis of the Potential Gaps due to the UK Grid Code Requirement' (Ref. 230). There was also a significant difference in the CUF factor for the charging line of the MCL a high value of 2.079 in 'Impact Analysis of Grid Code Compliance on Existed DTA Reports and Stress Analyses Reports' (Ref. 231) to an insignificant value based on 0.0000014/10%FP cycle claimed in 'Analysis of the Potential Gaps due to the UK Grid Code Requirement' (Ref. 230). I therefore issued follow-on RQ-UKHPR1000-1500 to progress matters (Ref. 28).

882. From the response to RQ-UKHPR1000-1500, I was able to confirm that the assumed step load increase/decrease of 10% was indeed bounding of the expected P-T transient.

883. The RP also explained the that the RPV core shell and MCL weld regions are directly affected by the reactor coolant system (RCS) transients, so they are selected as 'typical parts of typical equipment'. Whereas the RPV head flange and CRDM J groove weld, are more removed from the effects of the RCS transient, and their high CUFs are mainly attributable to the local geometries. Nevertheless, to underpin its judgements, the RP undertook additional fatigue analyses, which confirmed using conservative assumptions that the increase in CUFs was not significant over the 60-year plant life. This was provided in the RP's response to RQ-UKHPR1000-1461 (Ref. 28).

884. Similarly, the RP identified the charging nozzle connection to the loop 1 cold line as a 'typical part of typical equipment' for consideration of fatigue with a high CUF at the auxiliary weld of 0.78. However, I sought an explanation why the charging nozzle location on the MCL was held to bound other locations relating to the MCL where high fatigue usage might be expected, and were changes in the power cycling may be more acute e.g. the surge line connection to the pressuriser and MCL. The RP provided further information in its response to RQ-UKHPR1000-1461 (Ref. 28). In particular, the RP clarified that the effects of the step load increase/decrease of 10% were below the fatigue endurance limit in the RCC-M code for the MCL and SL. The charging nozzle location on the MCL was therefore an exception with an initial CUF estimate of 0.78.

885. I noted that a CUF of 2.079 for the auxiliary weld on the charging line off the MCL was estimated in further work (Ref. 230), as this differed significant from the initial CUF estimate, I questioned which result was correct and the consistency of the conclusions drawn in 'Impact Analysis of Grid Code Compliance on Existed DTA Reports and Stress Analyses Reports' (Ref. 231) and 'Analysis of the Potential Gaps due to the UK Grid Code Requirement' (Ref. 230). Indeed, the additional work suggested that either further justification or a replacement policy for the charging line within the 60-year design life was needed.

886. The RP clarified that the CUF value of 2.079 (Ref. 230) was a preliminary result and stated that additional analysis of the sensitive region will be carried out. After further analyses, the CUF for the MCL charging nozzle increased by about 0.03. The corrected CUF for 1 million extra cycles (note an increase of about 3000 cycles was expected) was therefore 0.81.

887. This notwithstanding, my review of the response to RQ-UKHPR1000-1461 (Ref. 28), indicated that, irrespective any increase in the FUFs from the grid code compliance

work, there were high CUFs (≥ 0.75) in the RPV closure head regions. I was also aware that the fatigue methodology used for the assessment of the RPV J groove welds differed from the usual miner's law type approach. I cover these topics as part of the design code compliance assessment work (Sub-section 4.4 of my report).

888. The RP also undertook additional work to estimate the effects on the fast fracture analyses (design code compliance) and the DTA, in particular, the fatigue crack growth estimates, which underpin the avoidance of fracture demonstration for the RPV.

889. The RP clarified that the low power 10% step change was not the limiting transient for the code fracture analyses. However, noting the significant changes in the CUFs claimed by the RP, I asked a TSC to undertake a sensitivity study to gauge the effect, if any, on changing the number of 10% power change transients in the LFCG calculations for the RPV inlet nozzle to safe end weld. The DTA for this location is more fully discussed under the avoidance of fracture demonstration (Sub-section 4.3.4 of my report), it will suffice to say that in terms of the UK grid compliance topic, my independent calculations, established that an increase of an additional 3,000 power 10% step changes did not have a significant effect on the FCG calculations for the RPV inlet nozzle to safe end weld (Ref. 232). Thus, in both cases the RP demonstrated that the effects were not significant giving confidence that an adequate safety case covering design code compliance (fast fracture) along with an avoidance of fracture demonstration for the RPV could be developed RQ-UKHPR1000-1461 and RQ-UKHPR1000-1500 (Ref. 28).

890. Overall, and based on the responses to RQ-UKHPR1000-1461 and RQ-UKHPR1000-1500, I was satisfied that the RP had provided adequate evidence for the purposes of the GDA to show that there was a basis for confidence in the achievement of UK grid code compliance from a SI perspective.

4.4.3.3 UK Legislative Requirements

891. Within the UK there are two main pieces of legislation which apply to pressure equipment. These are the Pressure Equipment (Safety) Regulations (PE(S)R) 2016 (Ref. 233) and the Pressure Systems Safety Regulations (PSSR) 2000 (Ref. 234)

892. In general terms, for new pressure equipment, it is the requirements of the PE(S)R 2016 that are of most interest. Within the PE(S)R 2016 there are requirements known as Essential Safety Requirements (ESRs) which are aimed at producing a minimum level of quality in the pressure equipment.

893. In addition to the ESRs the PE(S)R 2016 introduces the concept of conformity assessment which places a requirement to confirm that the ESRs have been complied with. ONR would expect that for the relevant pressure equipment the RP would ensure that these requirements are complied with and consistency is demonstrated through an appropriate conformity assessment.

894. As part of the PE(S)R 2016 there is an exclusion relating to pressure equipment designed for nuclear use (termed nuclear pressure equipment). The purpose of this aspect of the legislation is to allow more stringent requirements to be applied to those components important to nuclear safety. An example relates to ONR expectations for highest reliability structures and components which include provisions for third-party surveillance activities during design and construction (Sub-section 4.5 below).

895. The UK does not have any specific legislation to stipulate specific requirements for nuclear pressure equipment and as such is aware of international practice to apply additional requirements. The RP's strategy for identifying pressure equipment that are excluded from the PE(S)R and the requirements placed on such components were therefore explored at a principal level within the GDA.

896. For nuclear power plant that originate external to the UK it is ONR's expectation that pressure equipment be built to at least commensurate standards as that applied in the country of origin. This is to ensure that the pressure equipment to be used in the UK are at least as safe as if it were to be installed in the country of origin. The design and construction of the main pressure equipment for the UK HPR1000 is complicated by the fact that the design is of Chinese origin but built to a French and in the case of the SG and RCP American design codes. However, from the Chapter 17 of the PCSR, (Ref. 24) it was not possible to discern what legislative requirements are placed on pressure equipment to be installed in the reference plant design (Fangchenggang-3).

897. It is important that the requirements placed on the reference plant are understood to discern if the requirements placed on the UK HPR1000 will ensure a commensurate level of confidence in the pressure equipment. An example of additional requirements applied to nuclear pressure equipment is the French legislation ESPN 2014 (Ref. 220). This piece of legislation is as important as the RCC-M (Ref. 14) because the design code refers to aspects of the ESPN which are satisfied and those that it does not. The implication is that the use of RCC-M is not sufficient to meet the requirements of the ESPN. I am mindful that this implies that for nuclear pressure equipment designed to RCC-M there may exist a shortfall between what is proposed for the UK HPR1000 and that which may be accepted within France. It is important to understand the additional requirements that are applied by the ESPN and to determine if these are applicable to the UK HPR1000.

898. I raised RQ-UKHPR1000-0463 (Ref. 2) to request information on the RP's understanding of the UK legal requirements of pressure equipment and how these relate to pressure equipment specifically designed for nuclear purposes. In addition, I sought clarification on what requirements would be placed on nuclear pressure equipment within the Chinese legislative regime. The response to RQ-UKHPR1000-0463 (Ref. 28) gave confidence that the RP understood the UK legislative requirements and provided a useful summary of the position within the Chinese regulatory framework. However, it did not provide an indication of what would be applied for the UK HPR1000.

899. The legislative requirements for the supply of new pressure equipment within the UK are primarily documented within the Pressure Equipment (Safety) Regulations 2016 (PE(S)R) (Ref. 233). These requirements cover aspects such as design, manufacture and quality control and are applied in a graded approach depending on the pressure risk. Within the PE(S)R the nuclear exclusion exists to allow additional, and where appropriate more stringent, requirements to be specified, colloquially known as nuclear pressure equipment requirements.

900. These nuclear pressure equipment requirements are important because they inform the structural integrity provisions for metallic structures, systems and components with a pressure retaining function or that meet other conditions within the regulations. Thus, ONR expects the totality of the nuclear pressure equipment requirements for nuclear pressure equipment to be informed by PE(S)R supplemented with relevant aspects of international practice (e.g. China, France and US) and if appropriate, the UK expectations for highest reliability components (ONR SAP EMC.1-3, Ref. 2).

901. This topic was discussed at a SI technical exchange meeting (Ref. 218). ONR subsequently issued follow-up RQ-UKHPR1000-0677, (Ref. 28) which for nuclear pressure equipment which are excluded from the PE(S)R sought further explanation of the basis of the nuclear pressure equipment requirements for the UK HPR1000. I also requested clarification of the RP's sources of those nuclear pressure equipment requirements that are to be applied for the UK HPR1000 and its process for capturing the SI provisions for SSC.

902. The RP provided an informative response to RQ-UKHPR1000-0677 in which they explained that for Fangchenggang-3, the requirements for nuclear pressure equipment include the requirements in RCCM/RSE-M and some additional technical requirements according to Chinese practice. The additional requirements not covered in the RCC-M/RSE-M code included both technical requirements and additional quality assurance measures.

903. The additional technical requirements were illustrated for the RPV and included stricter controls on material composition limits (P, S, Cu and H) and on the non-metallic inclusion content; stricter limits on the Nil Ductility Transition Temperature (RTNDT) of filler metals used for main retaining welds < -30°C along with additional impact testing for production weld coupons; additional RT examinations for DMWs and more stringent UT examination of the core shell forgings. These are detailed in Table 1 of RQ-UKHPR1000-0677 response, (Ref. 28) and discussed further in Sub-section 4.5 below. These appear to reasonable measures to reduce risk and further measures may be developed in China based on OPEX and ONR would expect the licensee to keep these under review.

904. The additional quality assurance requirements accord with Chinese legislation. The quality assurance requirements in RCC-M A5000 are based on ISO 9001 are enhanced by the oversight of the Chinese Nuclear Safety Regulator, the National Nuclear Safety Administration (NNSA), and the regulation of quality assurance for nuclear power plants (HAF 003). An overview of the arrangements for design and manufacturing activities for nuclear pressure equipment was also provided in the response of RQ-UKHPR1000-0578 (Ref. 28). It will suffice to say that under UK law the responsibility for developing suitable and sufficient arrangements for QA rests with the licensee and ONR expects the license to develop an intelligent customer capability (Sub-section 4.2.1.1 above).

905. For the UK HPR1000, the RP proposed to implement the additional technical requirements as in Fangchenggang-3. In addition, UK legislation (PE(S)R) requirements and international practice for nuclear pressure equipment (such as ESPN) will be considered. In fact, most of the RCC-M ensures compliance with the ESR of PE(S) and ESPN. The RP also provided a comparison between ESR in the Directive 97/23/EC and the chapters of the RCC-M 2016 along with a comparison between the ESR from annex 1,2,3 of the Order of 12 December 2005 and the chapters of the RCC-M 2016 as provided in Appendix 1 and 2 of the RQ-UKHPR1000-0677 response (Ref. 28). The RP also committed to undertake a comparison between RCC-M 2018, PE(S)R 2016 and ESPN 2015, following which the any additional requirements not included in RCC-M 2018, will be assessed to decide whether these requirements need to be implemented for the UK HPR1000.

906. For the SG and RCP, which are designed and constructed according to US standards, the process for developing and capturing the provisions is similar to other nuclear pressure equipment, though the comparison is with the provisions of the ASME III design and construction rules. The details would feature in the response to RO-UKHPR1000-0033 Steam Generator Code Provisions and Mitigation of Relevant Risks (Ref. 49), as assessed in Sub-section 4.4.2.2 above.

907. The RP also explained its process for capturing the SI provisions for nuclear pressure equipment based on RCC-M and to meet UK expectations. This was a 3-step process comprising the review and identification of requirements not covered by RCC-M/RSE-M; a compliance analysis against UK legislations, additional requirements for HIC and gaps with RGP with any gaps between RCC-M and PE(S)R identified and measures implemented to ensure compliance; the totality of SI provisions will then be reflected in the component design or technical specifications.

908. I was content that the RP had developed adequate processes for establishing the totality of the SI provisions for the nuclear pressure equipment for the UK HPR1000. I was also satisfied with the commitment made to undertake code comparisons for RCC-M 2018 for pressure equipment designed to French codes. The RP subsequently provided a comparative analysis report between RCC-M 2016 and RCC-M 2018 Editions (Ref. 235).

909. Similarly, I was satisfied that the RP's process for establishing the totality of the SI provisions for nuclear pressure equipment could be adapted for components designed and constructed to the ASME III code. The RP subsequently provided a comparative analysis report between the ASME B&PVC 2007+2008 Addendum and 2019 Editions for the SG, and RCP (Ref. 236) (Ref. 237), respectively.

4.4.3.4 Assessment of Structural Integrity Provisions for In-Vessel Retention

910. The UK HPR1000 design includes an in-vessel retention (IVR) capability which floods the reactor pit and provides external cooling of the RPV during a severe accident scenario. This is to ensure the corium is retained within the RPV and the pressure boundary is not breached.

911. For those components that have been designated as HIC there is an expectation that these components will not fail when exposed to any credible loading conditions (i.e. within the design basis). Therefore, it is essential to have an accurate list of all load cases, as a result of a fault or hazard, for HIC structures and components.

912. During Step 3 of GDA, I identified that there was a risk that the reactor pit could be flooded during normal operation. This could be through a number of scenarios, including accidental operator action, spurious activation or mechanical / pipework failure.

913. As the injection of water could induce a significant stress on the RPV and has the potential for increased heat removal and reactivity insertion, I sought confidence that the RPV is either tolerant of such a scenario or suitable protection systems are in place such that the frequency of occurrence precludes consideration within the design basis.

914. In addition, I noted that within PCSR chapter 13 (Ref. 238), it is stated:

915. "After the relocation of corium, the wall of RPV becomes thinner due to melting by corium in lower head. The minimum thickness of the wall must have enough mechanical strength to maintain the integrity of the RPV."

916. I consider this claim as part of my assessment.

Inadvertent Flooding of the Reactor Pit

917. I raised the following RQ-UKHPR1000-0168 (Ref. 28) to obtain clarification on the risk of inadvertent actuation during normal operation.

918. In its response to RQ-UKHPR1000-0168, the RP presented a description of the IVR system and noted that it comprises a passive and active system. The passive system sources water from a designated water tank within containment to provide the initial cooling of the RPV. The active system sources water from the in-containment water storage tank (IRWST) and is used to maintain water levels during long term operation of the IVR system.

919. In terms of the inadvertent actuation of the system the response from the RP detailed information relevant to Probabilistic Safety Assessment (PSA) and Fault Studies. I engaged with the relevant inspectors to assess the information provided. The ONR PSA inspector identified that the PSA modelling was incorrect. In response to this

observation, the inspector raised RO-UKHPR1000-0032 (Ref. 28). This RO sought to gain a better understanding of the inadvertent flooding of the reactor pit fault sequence.

920. Of specific interest to the structural integrity assessment was:

"…the fault frequency should be determined, and a justification for the frequency should be provided. The justification should not solely rely on the current PSA models".

921. Through the resolution of this RO, the PSA inspector gained confidence that an initiating event frequency for spurious activation or leakage of the IVR system of 10^{-8} and 10^{-6} respectively was appropriate.

922. From a structural integrity perspective, we look for HICs to be able to withstand all design basis loadings. It is my understanding that design basis loadings are those that arise from normal operation and fault or hazards with an initiating event frequency of 10^{-5} or higher. Given that the RP states that the initiating event frequency of spurious activation is of the order 10^{-8}, I am of this opinion that it is acceptable for HIC structural integrity safety case to ignore this potential fault sequence.

923. In terms of leakage, I noted that the supported initiating event frequency is 10^{-6}, which is only just above the design basis threshold. Whilst this fault sequence could be censored from the design basis loading list, if a strict application of the 10^{-5} threshold was applied, I have given it further consideration. I have gained some confidence that the initiating event frequency estimation for leakage is potentially conservative as it does not include the leak detection systems in the reactor pit nor the potential operator actions.

924. If these initiating event frequencies were to be wrong, some confidence can be taken from the fracture analyses conducted by the RP. Using the RCC-M fast fracture analyses (Ref. 239) the RP claims that the RPV would survive both a spurious initiating and leakage events. This analysis approach is not consistent with the HIC AOFD methodology but given the initiating event frequencies I am content it is a reasonable approach.

925. I note that the lowest margin presented is 2% against a 3mm ductile tearing limit for an axial crack. This provides minimal margin, so I have considered the analysis in more detail. The limiting defect orientation is axial which is not deemed credible by the RP's review of the manufacturing process (Ref. 239) . For a circumferential defect a greater margin was predicted (~20%). In addition, weld residual stress is not included which it is my understanding would be compressive on the outer wall and hence could improve the resistance to crack propagation. Finally, the analyses consider cold water being applied to the entire outside surface, which I view as conservative. So, in summary whilst the analysis shows minimal margin against the identified limit there are areas of conservatism which could increase this margin.

926. For the position of GDA I am content that, from a SI perspective, the inadvertent flooding of the reactor pit does not pose a threat to the RPV integrity. This is based on the low likelihood of the spurious initiation and the confidence that can be taken from the RCC-M fast fracture analyses.

IVR Operation

927. As part of RQ-UKHPR1000-0168 (Ref. 28), I also queried on what basis adequate mechanical strength is determined during a severe accident.

928. For the deliberate activation of the IVR system, referred to as the 'IVR Condition', the RP has documented the justification within 'The Structural Integrity Assessment of

RPV on In-Vessel Retention Condition' (Ref. 240) and 'The Thermal Shock Analysis of RPV While Triggering IVR Condition' (Ref. 241). The calculations consider the static strength, thermal shock transient and creep.

929. In my assessment of the evidence provided to support the claim that the RPV has enough "mechanical strength", I have assumed that this means the RPV boundary will not fail. As this sequence is a beyond the design basis event (i.e. initiating event frequency of 10^{-7} or greater) I am of the opinion that it is reasonable that the HIC AOFD expectations should not apply during a sever accident scenario.

930. Within Step 3 of GDA I raised a number of points regarding whether the structural integrity justification considered all relevant aspects. The specific points I noted were:

- The codes used to do not support the material properties in the temperature range expected during IVR operation.
- The claimed operation time the IVR system is not stated.
- There is no identification of failure and degradation mechanisms.
- It is not clear if the credible loads have been combined to produce an operational load profile.

931. During Step 4 of GDA I have made the following progress:

- During discussions with the RP they clarified which sections of the code provided such information and I am now content that the code does provide the appropriate information. The only exception is tensile strength values where test data has been used instead which appears reasonable.
- During Step 3 of GDA, I had a concern that the analysed time didn't cover the full period of interest. It is now my understanding that the temperatures will have dropped significantly after 48hrs such that there is no risk of ongoing degradation from creep or boiling affects.
- The RP's submission (Ref. 240) was revised to include more information.
- I am now content that the dominant loads have been identified and are captured within 'The Structural Integrity Assessment of RPV on In-Vessel Retention Condition' (Ref. 240).

Conclusions of IVR

932. Based on the assessment conducted within Step 3 and Step 4 of GDA, I am content that the RP has addressed my concerns. This has resulted in additional information being made available to provide confidence that the IVR will maintain its integrity during deliberate IVR operation.

933. Based on the likelihood of activation both in a spurious or sever accident scenario, I am content sufficient justification, from a SI point of view, of the RPV integrity has been provided for the purposes of GDA.

4.4.4 Strengths

Code Compliance

934. I am content with the use of the RCC-M code for the design and construction of the majority of the vessels and piping in the UK HPR1000, and with the use of a graded approach to establish the SI provisions which reflects the importance to nuclear safety. My assessment focussed on the application and demonstration of design code compliance.

935. I undertook initial and detailed reviews of a sample of the major vessels and piping in the UK HPR1000. Although several comments were raised against the RP's

submissions the RP acted constructively to address my points with vast majority of the comments and clarifications requested in my RQs closed.

936. My review work indicates that there is not a systemic problem with component designs meeting code requirements, because in responses to my RQs, the RP demonstrated an understanding of the code requirements and has provided further information to demonstrate code compliance.

937. The RP outlined its approaches, conservatisms, assumptions and uncertainties and sensitivity studies employed in its fatigue analysis for the MCL Charging Line. This gives some confidence that the RP is aware of the need to understand and discuss the significance of high fatigue usage factors taking cognisance of the uncertainties and conservatisms.

938. For each region where high FUFs were predicted the RP outlined its strategy and the measures in place to manage the risks, which included provisions for ISI, replacement or repair and the use of a fatigue monitoring system. These appeared reasonable with the majority of locations subject to ISI in accordance with RSE-M and/or repair or replacement measures available.

939. The RP recognised the need to keep abreast of developments in both the US and French nuclear design and construction codes and demonstrated that adequate processes should be available to inform the design code provisions for the UK HPR1000 SG. The RP's approach gives a basis for confidence that significant changes in the design codes can be identified and actioned.

940. The RP demonstrated an adequate understanding of UK expectations to, if necessary, inform the design, development and manufacture of an alternative UK HPR1000 SG design.

Additional SI Provisions

941. I was satisfied with the RP's proposal to delete the periodic hydrotest. I was also satisfied with the RP's proposal to undertake objective-based high reliability inspections in lieu of any additional designer inspections imposed by the designer in Fanchenggang-3 for HIC components.

942. I am content that the RP had developed an adequate process that would allow the totality of the SI provisions to be determined for the UK HPR1000 SG. Notably, the SI provision developed for the UK HPR1000 SG take cognisance of both Chinese practice and UK expectations (UK legislative requirements and the highest reliability).

943. I was also satisfied that the RP's process for establishing the totality of the SI provisions for nuclear pressure equipment could be adapted for components designed and constructed to the RCC-M and ASME III codes.

944. The RP provided adequate evidence for the purposes of the GDA to show that there was a basis for confidence in the achievement of UK grid code compliance from a SI perspective.

In Vessel Retention

945. During my GDA Step 4 assessment of the IVR system I have identified the following areas of strength:

- The RP has been quick to understand my concerns and whilst not immediately evident the RP has had supplementary information available.
- The RP has identified shortfalls in its evidence and acted appropriately to address these concerns.

4.4.5 Outcomes

Design Codes and Standards

946. My initial review work highlighted some potential difficulties relating to referencing, source data, lack of evidence, discussion on small margins or claimed conservatisms. These subsequently informed the more detailed review work and the overall conclusions drawn with respect to design code compliance.

947. Several points of clarification were raised in my review work where either insufficient evidence was provided, or the technical argument provided in response to the comment did not fully justify the approach adopted. These include:

- Design by Analysis results being claimed as evidence in lieu of Design by Rule assessments noting that this relates to referencing the evidence as opposed to difficulties with the code compliance.
- Tri-axial stress and thermal stress ratchet checks being omitted without adequate justification.
- The rules of ASME III Appendix A for tube sheets being applied incorrectly.
- Lack of evidence that axisymmetric modelling of an asymmetric nozzle is appropriate.

948. I judge these represent only a low risk to the design of the components in question due to either the presence of reasonable margins, and that it is very rare for the omitted checks to be failed without also failing at least some of the checks that have been conducted and passed, or alternative means were used to justify the component. Nevertheless, it is important that these points, which inform the integrity demonstration for the most safety significant components in the UK HPR1000 are tracked to completion via the assessment findings raised.

949. My review of code compliance also identified in some instances difficulties with the traceability of evidence within the safety case, for example, the sources of the loads.

950. Further points relate to the RP’s consideration of small margins and the level of conservatism in its design code assessments for HIC components. I draw confidence that the RP has a process available and has shown it can be implemented, though my sampling suggests this was not systematically applied.

951. On occasion insufficient details were provided in relation to the FE modelling methodologies.

952. The application of the RCC-M Annex ZD approach to fatigue life estimation should be underpinned by adequate evidence to validate the specific fatigue initiation (S-N) curves, in particular, for material interface regions in dissimilar metal welds (DMWs).

953. My review of the compliance of the UK HPR1000 SG with the ASME III code was limited by the restraints of propriety information and so the scope of my review was limited compared to the level of scrutiny applied to other components.

954. I draw confidence from the fact that the UK HPR1000 SGs are a mature design, and that the UK HPR1000 SG design is underpinned by a design report certification statement. However, additional information will be needed for the site-specific stages, and further work informed by the SI classification, is warranted to justify the integrity of closures in HIC components.

955. For some locations with high fatigue usage factors, although repair or replacement was feasible, the RP was not proposing any means of monitoring for potential fatigue damage. In addition, for certain locations with high FUF, the RP considered the fatigue analyses were overly conservative and despite the apparent absence of a repair or

replacement option, no additional measures to provide forewarning of failure were offered.

956. During the site-specific stages, it will be necessary for the licensee to develop and demonstrate effective implementation of its processes for accounting for changes in the design codes.

957. There is a lack of a consistent provision of succinct descriptions of the process by which the design loads, as listed in the design specification for each component, are derived. The shortfall is with traceability rather than the actual magnitudes and frequencies of occurrence of the transients.

958. I have raised eight assessment findings (AF-UKHPR1000-0198 – AF-UKHPR1000-0205) to address these points.

Combination of Codes & Standards

959. ONR guidance in ECS.3 and Para. 173 of the ONR SAPs (Ref. 2) states that "the combining of different codes and standards for a single aspect of a structure, system or component should be avoided. Where this cannot be avoided, the combining of the codes and standards should be justified and their mutual compatibility demonstrated." From the relevant sections of the UK HPR1000 SG and RCP safety case that I have sampled, and taking cognisance of ONR guidance, I am satisfied that the RP has demonstrated that the combining of codes and standards for these components is justified.

960. In assessing the RP's approach to combining codes and standards for these components, I have found no evidence to suggest that the RP's intent in combining codes and standards was an attempt to minimise standards. I am satisfied that the RP has selected an appropriate nuclear design and construction code for the SGs and RCP.

961. The RP has explained and documented its process for identifying relevant risks arising from supplementing the UK HPR1000 SG and RCP design and construction code (ASME) with additional measures from the RCC-M and RSE-M codes. The RP also provided adequate information to clarify the location of the physical and organisational interfaces and how relevant risks arising at the connections between the nozzles and the safe-ends for the UK HPR1000 SG will be managed.

962. The RP clarified that there was adequate access to facilitate EIMT activities and the specific arrangements for the UK HPR1000 SG would be captured in the SG EOMM at the site-specific stages.

963. The RP's proposals for the PSI when using a combination of codes and standards were diverse and comprehensive, such that provided they were implemented rigorously, they would undoubtedly afford additional confidence in the integrity of the UK HPR1000 SGs and RCPs, irrespective of the additional measures expected to underpin a highest reliability claim.

964. Whilst the information assessed has provided a basis for confidence for managing the majority of relevant risks, there are still a number of aspects that I consider need to be addressed to fully meet ONR expectations that compatibility of the codes are mutually demonstrated. One example identified is the risk associated with introducing a different set of DMWs at the SG and RCP piping interface locations that employ combinations of codes. These warrant further demonstration that relevant risks are reduced to ALARP, and where relevant, I have raised two AFs (AF-UKHPR1000-0206, AF-UKHPR1000-0207. I consider these AFs apply equally to the UK HPR1000 RCPs.

Additional SI Provisions

965. The RP's position with respect to the post hydrotest inspections and periodic (10 year) in-service inspection proposals for SIC-1/2/3 components was unclear. I have raised an assessment finding to progress this matter at the site-specific stages (AF-UKHPR1000-0207).

966. The RP has implemented additional SI provisions in the reference plant that appear to be reasonable measures to reduce risk and based on OPEX further measures may be developed in China. I have raised an assessment finding (AF-UKHPR1000-0208) to ensure that the licensee considers these measures in developing the SI provisions for the UK HPR1000.

4.4.6 Conclusions

967. In response to the queries raised, in many cases the RP demonstrated an understanding of the code requirements and where additional evidence has been provided it has been sufficient to demonstrate code compliance and confirm that conservative assumptions were made. Whilst there are a number of points raised where either insufficient evidence was provided, or the technical argument needed further justification, these are judged to be a low risk to the design of the components in the UK HPR1000. This notwithstanding I have raised several assessment findings to ensure these are tracked to a satisfactory conclusion post GDA.

968. The RP provided an adequate explanation of its process for identifying relevant risks arising from supplementing the UK HPR1000 SG design and construction code (ASME) with additional measures from the RCC-M and RSE-M codes. There was therefore suitable and sufficient evidence to close RO-UKHPR1000-0033.

969. I am also content that the RP had developed an adequate process that would allow the totality of the SI provisions to be determined for the UK HPR1000 SG. Notably, the SI provision developed for the UK HPR1000 SG take cognisance of both Chinese practice and UK expectations relating to legislative requirements and where appropriate highest reliability.

4.5 Material Selection, Testing and Surveillance

970. The materials selection process adopted by the RP is an integral part of providing a structural integrity safety case. Materials selection plays a key role in ensuring that through life degradation can be minimised, and where degradation cannot be avoided it can be managed and mitigated appropriately. Moreover, it is necessary to have appropriate confidence in the mechanical properties of any selected materials, so as to support the defect tolerance analysis which fundamentally underlies the structural integrity safety case, especially four components of the highest reliability.

971. ONR has, therefore, an expectation that materials to be used in new build nuclear power plant in the UK shall be demonstrably capable of performing their safety duty throughout their life. This means that materials should be selected using a robust methodology that takes into account mechanical performance requirements, resistance to degradation, and the ability to test and inspect these materials through life in such a way as to support underlying safety case.

972. The UK HPR1000 materials selection methodology (Ref. 242) was assessed as part of the Step 3 GDA Structural Integrity report (Ref. 7). The methodology was considered to be, in and of itself, acceptable as a framework for the assessment and evaluation of materials suitability for use in the UK HPR1000. ONR noted, however, that the application of this methodology would still need to be tested and made provision for this at Step 4 of GDA. Moreover, in the GDA Step 3 report for structural integrity (Ref.

7), there was provision made to look specifically at the materials selection process as applied to the SG.

4.5.1 Assessment

973. I performed a sample review of materials selection reports and took an overview of the application of the materials selection methodology for these components. This focussed upon the application of the methodology outlined in the RP's 'Material Selection Methodology' (Ref. 242) and also took a view on whether the materials selected would be acceptable for the UK context.

974. To gain an appreciation of how the RP's materials selection methodology had been applied in response to different safety classifications, my sample consisted of several components, including HIC, SIC-1 and SIC-2 components. Where necessary, I raised a number of RQs as a result of my review, to better understand the RP's approach and application of the materials selection methodology. The components selected for sampling and the corresponding RQs (Ref. 28) are:

- Reactor Pressure Vessel Internals (RQ-UKHPR1000-1455).
- Steam Generators (RQ-UKHPR1000-1456).
- Reactor Pressure Vessel (RQ-UKHPR1000-1457).
- Main Feedwater System.

975. Within the 'Materials Selection Methodology' (Ref. 242), the RP commit to performing a preliminary selection of materials based upon Relevant Good Practice (RGP) and OPEX. The 'Materials Selection Methodology' (Ref. 242) further states that this process involves forming a longlist of candidate materials from 'evolution history' and thence a shortlist from 'RGP/OPEX and feedback'.

976. It was my expectation that these longlisting and shortlisting processes, as presented in Section 5.3 (Ref. 243), should be either referenced or presented in a suitable auditable way. This is not only to ensure that the RP is following its own internal process appropriately, but also to ensure that any learning gained may be passed on to the licensee. In all three of the components sampled, I did not consider that this process was presented in a meaningful way and it was not possible to ascertain whether the methodology had been followed.

977. To ensure that ONR's expectations are demonstrably met, I raised RQs against three of the four areas sampled. These RQs focused on ensuring that a suitable process had been followed and additionally whether or not the process had produced a result that was suitable for UK context. I used these results to assess the levels of confidence that I could have in the behaviours and competence of the RP.

4.5.1.1 Review of RPV

978. The materials selection report for the RPV is presented in 'Material Selection Report of Reactor Pressure Vessel' (Ref. 243). From my review of this document, I judged that the choices of materials are limited to a narrow band of materials and is not consistent in its groupings of materials with other materials selection reports. For example, the main RPV forgings are selected from SA-508 Grade 2, and SA-508 Grade 3 Class 1. The materials specification 16MND5, which is included in the RCC-M design code being used for UK HPR1000, is judged to be a subset of the SA-508 Grade 3, Class 1 materials specification. Whilst the two grades are extensively similar, they are not identical. As discussed later, the methodology used for grouping of grades was not consistent is used inconsistently between components. It was unclear, therefore, whether the RP had provided a suitable application of the 'Material Selection Methodology' (Ref. 242) for the RPV main forgings. I therefore raised RQ-UKHPR1000-1457 to address these shortcomings (Ref. 28).

979. The RP responded to RQ-UKHPR1000-1457. My primary concern in this area was that the methodology put forward in the 'Material Selection Methodology' (Ref. 242) must be demonstrably followed. This included provision of a long list of possible candidate materials, followed by a shortlist of acceptable materials, followed by the selection of the optimum material. It was clear from the response given that a suitable degree of consideration had been given to world experience of pressure vessel steels. This included an overview of not only the steels that generally used within the RCC-M design code, but also gives consideration to other steels from wider world experience and the history of the nuclear industry since its inception. The level of knowledge shown by the RP in this area was significant; this gave me confidence that behind the sparse choice of materials shown in the original submission, proper consideration had been given to worldwide experience.

980. The final selection of 16MND5 was made, giving the reasons behind why this was most appropriate material to be used. I judge that not only has the materials selection methodology presented by the RP been properly used in this instance, but also that the material selected is suitable and appropriate, and a defensible position of its selection has been presented.

981. The selection of 16MND5 in this instance falls in line with UK expectations especially given the use of the RCC-M design code. Use of any other material would have presented challenges in justifying a material outside of design code, and a material that could be potentially novel and introduce unknown factors.

4.5.1.2 Review of Steam Generator

982. In the 'Material Selection Report of Steam Generator' (Ref. 244), the RP presented its views on the materials selection for the Steam Generator. This was flagged by ONR in Step 3 of the GDA as a matter to follow up as part of the Step 4. Similar to the RPV and RVI materials selection reports, the process for long-listing and short-listing of materials was not well explained within the submission. Furthermore, although the SGs are to be built using the ASME design code, and not RCC-M as per the bulk of the rest of the reactor system, I note that the nozzle safe ends are to use a material specification from RCC-M and hence are not, strictly, compliant with the ASME code. The reason why the RP considered this to be ALARP was not discussed in the submission. Finally, and similar to the RPV materials selection report, for the highest reliability sections of the SG, a demonstration that the materials selected were suitable and ALARP was requested. These queries were presented to the RP as part of RQ-UKHPR1000-1456 (Ref. 28).

983. The RP responded to RQ-UKHPR1000-1456. Similar to the examples of the RPV and RVIs, given above, the RP was able to provide evidence that a full and broad range of materials had been considered during the long-listing process. The RP presented information on the history of SG tubing materials and the corrosion and degradation that has occurred around global experience. This included consideration of accepted practice from north American, western European, east Asian, and Russian designs and balanced the positives and negatives of each of the possible selections. Based on this information, I have confidence that the RP has implemented its materials selection methodology adequately. Moreover, based upon the selection of Alloy 690TT for the tubing material, which is a nickel-base alloy with a proven track record of integrity within nuclear steam generators, I judge that the outcome of the SG tube materials selection process is adequate.

984. Regarding the safe ends on the SG, the RP noted that the SG safe end will be a material specified to RCC-M, welded using procedures and consumables in accordance with ASME, onto an ASME specified buttering layer and an ASME specified main forging. The RP note that the safe ends are specified to be a Z2CND 18-12 CN material. This is an austenitic stainless steel similar an AISI type 316L steel

and equivalent steels exist for use as safe ends within the ASME design rules. I judge, therefore, that the use of this material type is adequate. Of note, in the 'Material Selection Summary Report' (Ref. 245) the only nickel-base welding consumable that is proposed for use in HICs is ERNiCrFe-7. This is equivalent to the American specification for UNS N06052, commonly called Alloy-52. This is in line with my expectations that, for the primary circuit, materials resistant to primary water stress corrosion cracking (PWSCC) should be used. In any site-specific phases, my expectation remains that PWSCC resistant materials should be used where appropriate.

985. The RP states in its response to RQ-UKHPR1000-1456, that the "design of the safe ends aims to ensure high quality and reliability of welding activities can be achieved on site, where the welding, inspection and heat treatment condition is limited". I interpret this to mean that the complexity of having to make a weld between an RCC-M and an ASME component is better controlled within a welding shop environment, rather than being performed on site. I judge that this is a reasonable argument but note that there is no discussion of the additional risk posed by procuring a what is nominally an ASME component containing RCC-M materials.

986. Given the similarity between materials specification from ASME and RCC-M in this instance, I am content that the increased risk is, from a metallurgical point of view, negligible. This notwithstanding, I recommend that the ONR Structural Integrity inspector ensure that sufficient oversight is paid to this weld during manufacturing to provide assurance.

987. Noting that the SG contains regions of highest reliability materials, designated as HIC by the RP, I note that the RP proposes to ensure that there are suitable and sufficient fracture toughness specimens taken to support the Safety Case, this is discussed by the RP in the report 'Supplementary toughness testing requirements for HIC Components' (Ref. 105), and is also supported by suitable and sufficient NDE, discussed separately in this report. Moreover, the Ageing and Degradation of the SG is discussed by the RP in the report entitled 'Ageing and Degradation of the SG' (Ref. 246). From my high-level review of the 'Ageing and Degradation of the SG' report (Ref. 246), I note that the RP has considered a wide range of physical and chemical degradation mechanisms. From the review performed I am satisfied that this has captured the expected degradation mechanisms suitably.

988. From the above I have confidence that the materials selection methodology has been suitably applied to the SG and that the materials selected are acceptable.

4.5.1.3 Review of Reactor Pressure Vessel Internals

989. As an example of a component important to Structural Integrity, but that does not have a pressure boundary role, I sampled the reactor internals. In contrast to the RPV, the reactor internals are not components of the highest reliability, being a Structural Integrity class 1 (SIC-1) but do perform an important Structural Integrity function of supporting the core. In contrast to the RPV, the reactor internal comprise a series of different components, each with different functions and requirements.

990. In the 'Material Selection Report of Reactor Vessel Internals' (Ref. 247), the RP presented its groupings of components, and applied the materials selection methodology to select materials. As was observed in the RPV materials selection report, it was not clear how the materials have been longlisted and shortlisted. The application of the materials selection report was not presented in a clear fashion, and some of the presented candidate lists of materials, in my judgement, were incomplete. I challenged this in RQ-UKHPR1000-1455 (Ref. 28).

991. The RP gave a full explanation of the materials discussed and presented evidence that the selection process had not simply followed the design code, but had considered materials from other design codes and properly considered the impact this might have in terms of potential materials improvements, balanced against the disbenefits of not complying with the design code used for these components (RCC-M). I gained confidence from these responses that the materials selection methodology outlined in the 'Material Selection Methodology' (Ref. 242) has been applied and, although it is disappointing that this was not presented through the materials selection report (Ref. 247), I was content that, from the point of view of Structural Integrity, the RP has provided an adequate explanation of why the different materials have been chosen for the RVIs. I consider that the materials selected are in line with internationally accepted practice and that the case presented by the RP that they are ALARP is also adequate.

4.5.1.4 Review of Main Feedwater Line Materials Selection (MFL)

992. The MFL forms part of the secondary circuit of the UK HPR1000. The primary purpose of the MFL is to control the flow of feedwater to the Steam Generator SG. The MSL then connect the SG and the steam consumers, making them a significant component of the pressure retaining boundary in the Main Steam System (MSS). Therefore, judicious material selection is vital to maintain high reliability throughout the design life of the plant and ensure the continuous removal of heat from the Reactor Coolant System (RCS).

993. The safety function of the MFL varies depending on the direct and indirect consequences of failure within each specific area that the pipe runs. The RP has assigned several SI classifications to various components depending on location, the highest being SIC-2 for the pipework located within the containment.

994. As part of the Step 4 GDA SI assessment, I have performed a review of the materials selection methodology applied to the MFL, to seek assurance that the RP has applied it consistently and proportionately for a component classified as SIC-2. I have completed this review in conjunction with the ONR chemistry specialist, who has assessed the adequacy of the materials selected with respect to ensuring the risk presented the secondary circuit from an aging and degradation perspective is reduced ALARP.

995. In accordance with the RP's 'Safety Case Methodology for HIC and SIC Components' (Ref. 26), argument 1-3 states that 'components are manufactured through judicious material selection'. For the MFS, the RP has followed it's material selection methodology (Ref. 242) and produced 'Material Selection Report of Main Feedwater Line' (Ref. 248).

996. The supporting evidence presented clearly identifies how the process of materials selection has been undertaken, seeking input and scrutiny from a wide range of system, chemistry and engineering specialists. The RP's multidiscipline review team has considered a comprehensive range of materials, shortlisted from design code materials that account for a broad range of OPEX and precedent from historic and modern nuclear plant designs. In my opinion, the RP has identified and prioritised the key factors for ensuring safe performance of the MFL, namely being mechanical performance, manufacturability and resistance to ageing and degradation.

997. Three materials were shortlisted and further scrutinised using a sensitivity study and optimisation approach to differentiate between the candidate materials. The RP has settled on the selection of RCC-M-code designated P280GH, with additional requirements to specify a minimum Cr level. This was identified from OPEX and considered a reasonable modification in order to reduce the risk of the primary ageing and degradation mechanism of flow assisted corrosion for the MFL.

998. From an SI perspective, I am satisfied that the RP has demonstrated that a judicious materials selection process has been applied for the MFL, such that the risk associated with structural integrity failure of the MFL has been reduced ALARP, commensurate with the assigned structural integrity classification. From the information I have sampled, I have confidence that the materials selected are acceptable and in line with ONR expectations.

4.5.1.5 Review of Containment Liner

999. The containment structure is a civil engineering structure and as such is largely outside the scope of the SI assessment. This notwithstanding, the containment liner is a metallic component which is part of the containment structure providing leak-tightness, but not performing a structural function. As part of the Step 4 GDA SI assessment, I have performed a review of the containment liner, and its fixtures, proportionate to the Structural Integrity safety classification of this component, and from the perspective of a Structural Integrity inspector. I note that concrete sections of the containment structure are considered within the Civil Engineering Assessment report (Ref. 78).

1000. The RP presented the analysis and design of the internal containment liner in the 'Analysis and Design of the Internal Containment Liner' (Ref. 249). I note that the liner, as a complete component, has been designed according to the ASME code for containment liners. This is in line with the design code for the RPV and other major components within the nuclear steam supply system and, as such, I consider it to be a suitable code for this purpose. The RP state that the liner is to be made from P265GH steel, a weldable pressure vessel steel. I note that there is no materials selection process presented for this steel. I note that this is a European specification of steel (EN 10028:2) and is to be used during construction to a USA design code. I further note that this steel is similar in composition to the American Society for Testing and Materials (ASTM) standard A516 Gr.60. I judge that the material selected in this instance is adequate. I note that, during the site-specific stages, it will be incumbent upon the licensee to ensure, once materials are procured, that the design calculations made during the GDA phase are supported by the materials properties of the steels utilised on site.

1001. I have not performed a comparative calculation for the stress analyses underpinning the RP's submission. This notwithstanding, I note that the ASME code forms the basis of the case proposed in terms of load combinations and acceptance criteria. I am content that this is an acceptable approach to the analyses performed.

1002. The analyses performed have utilised a finite element modelling technique, wherein the steel liner has a total of 15749 elements and the stiffeners 5090 elements. I have not chosen to sample the analyses performed, but note that the methodology applied is in line with my expectations for these components. I note that the studs on the containment liner have not been modelled, the RP claim that there is sufficient allowance made in the modelling of stress to compensate for this omission. I judge that this is acceptable for the purposes of GDA, but will need to be clarified during the site-specific stages to ensure that this assumption is applicable to the as-built condition.

1003. As part of this analysis, I have not sampled the hatches and manways into the containment structure.

1004. The anchor points form the connection between the steel liner and the structural concrete. Also presented in the 'Analysis and Design of the Internal Containment Liner' (Ref. 249), are the anchors for the polar crane. The RP has presented its analysis and testing results of the liner anchors in tension and in shear. Again, the materials selected for the anchor plates, stiffeners and bars have not been subject to a materials selection process. I note that the steels selected all are weldable grades with high levels of ductility and are widely used for these purposes. On this basis, I judge that the materials selected, P265GH, S235 and B500C respectively, are adequate.

1005. The RP has considered the possibility of the liner delaminating from the concrete containment under negative pressure and have provided a justification that this will not occur. I have not performed a full numerical review of this possibility. Through a high-level review of the information presented, I judge that the analysis performed is acceptable for GDA and falls in line with previous UK experience.

4.5.1.6 Review of Forging Considerations for Materials Selection

1006. Following on from the work done during Step 3 of GDA, I investigated the suitability of the RP's design in terms of fabricability and how the manufacturing process could be demonstrated to meet the RCC-M design code. I raised RQ-HPR1000-1088 and RQ-HPR1000-450 to address these matters.

1007. In its response to RQ-HPR1000-0450 and RQ-HPR1000-1088 (Ref. 28), the RP states that issues of carbon macro segregation, which have been observed in the international community, will be controlled by ensuring that suitable technical qualification is made during the manufacturing of these components. Specifically, there will be sufficient discards made on the ingots produced such that areas of chemical macro segregation will be discarded and not continue into the final product. This is on top of chemical control. I judged that this answer is adequate for GDA, but this does not obviate the need to ensure that the final product does not contain macro segregation or other similar metallurgical flaws. It remains ONR's expectation that any forged component shall be capable of fulfilling its nuclear safety function throughout the thickness of that component.

1008. As part of my review of the RP's methodology for material selection, I selected several components to assess how the consideration of reducing risk ALARP had been applied properly to the materials selected for the RPV, the PZR and the SG forgings. The details and scope of my review, along with minor shortfalls identified are provided in more detail later in this report under Sub-section 4.9 'Demonstration that Relevant Risks Have Been Reduced to ALARP'. In summary, I am content that within the remit of the GDA, the ALARP position from the RP is adequate broadly satisfied that the RP

4.5.1.7 Irradiation of Materials - Surveillance and Monitoring

1009. A flux of neutrons through a material can affect the properties of that material; notably, it can reduce the fracture toughness through-life. The fracture toughness of a material is an essential input to any defect tolerance analysis and, as such, forms a critical part of the Safety Case for any component of the highest reliability. As part of the defect tolerance analyses through-life, it is my expectation, therefore, that there should be suitable direct measurement of fracture toughness for highest reliability components.

1010. During Step 3 of GDA, I raised RQ-UKHPR1000-0508 (Ref. 28) to determine whether the RP has a conservative process for monitoring aging and degradation though the life of the RPV. In its response, the RP state that they have identified irradiation embrittlement and fatigue, thermal and mechanical, as the principal method of degradation for the RPV. They outline at a high level the steps taken to ensure that these effects have been addressed in design and operation of the plant. I judge that this is an adequate response to the RQ and note that these matters are considered specifically elsewhere in the GDA.

1011. In the response to RQ-UKHPR1000-0508, the RP also outlines how neutron fluence has been used to infer decreases in ductility (RTNDT) expected through the life of the reactor. This is based upon comparisons with international standards from the USA, France and Japan. I consider this acceptable for GDA, but note that the ONR's expectation is for direct measurement of fracture toughness through-life to underpin defect tolerance analyses.

1012. The RP's position on the irradiation surveillance process is described in the 'Irradiation Surveillance Requirement of RPV Core Region Material' (Ref. 250). This gives the location, number and orientation of samples to be taken for irradiation surveillance purposes. Irradiation surveillance capsules are samples of original forged materials placed into, or near, the core such that they have neutron doses in advance of the actual bulk pressure boundary forgings, whilst in a representative environment. I performed an in-depth review of the 'Irradiation Surveillance Requirement of RPV Core Region Material' (Ref. 250) as part of my assessment of the RP's GDA submissions.

1013. At a high level, I note that the RP is intending to perform irradiation toughness surveillance on samples from the RPV only. I judge that this is acceptable and is in line with UK expectations and experience. I note further that the RP has proposed performing surveillance on samples from the RPV forging material, weld material and Heat Affected Zone (HAZ) material. The surveillance capsules are proposed to contain materials for three difference testing regimes: Charpy V-notch, tensile tests and fracture toughness specimens. Moreover, the structure of the testing programme is based upon the American standard ASTM E185, which is entitled "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels". This is an internationally recognised standard, and I judge that it is not in conflict with the French design code used for the nuclear plant. This means that, overall, the surveillance programme contains the materials expected and follows a recognised international standard.

1014. The positions of the samples are outlined in Section 6.1 of the 'Irradiation Surveillance Requirement of RPV Core Region Material' (Ref. 250). Whilst they are not referenced back to ASTM 185, I recognise these are being in accordance with that standard. These samples reflect accepted practice used in other highest reliability applications within the UK. This includes the positioning of the notches in the materials to follow the lines of metallurgical lowest toughness, for example, along the forging lines. I judge that these are appropriate to pick up the directions of the lowest toughness and will, therefore, provide bounding values of toughness for use in the defect tolerance analyses.

1015. In Appendix 2 to the 'Irradiation Surveillance Requirement of RPV Core Region Material' (Ref. 250), the RP present a recommended capsule insertion and removal schedule. This covers a nominal 64.8 year reactor life-span. I note that this is a recommended schedule and would be capable of meeting the needs of the current GDA.

1016. The numbers of specimens and their locations are given in Appendix 3 of the 'Irradiation Surveillance Requirement of RPV Core Region Material' (Ref. 250). I note that there is currently no intention to pursue fracture toughness testing for the HAZ in the irradiated condition. I note that there was no intention in initial submissions to test HAZs in the main fracture toughness testing programme. This was based upon the RP's statement that the base materials and weld material would have limiting properties compared with the HAZ.

1017. Regarding the testing of irradiated samples, it may be possible to prove, through the fracture toughness testing of unirradiated samples, that there is sufficient margin between the HAZ and the other regions such that they can be considered bounding. This remains to be demonstrated and, given the highest reliability claim for this component, will require evidence to demonstrate this that is specific to the materials used in all deployments of this technology in the UK. I judge that this is a minor shortfall in the RP's Safety Case to be addressed during the site-specific stages. The licensee should make provision for the testing of irradiated HAZ samples or provide materials-specific evidence as to why these data will be bounded by the proposed irradiation embrittlement surveillance scheme.

4.5.1.8 Summary of Materials Selection

1018. As part of RQ-UKHPR1000-0508 (Ref. 28), I questioned the RP to ensure that the RPV, and other highest reliability components, not only met code, but considered ALARP measures above code, where reasonable. In the response made to this RQ, the RP has provided evidence that the experiences and information from previous GDAs has been taken into account, notably in the chemical composition of the main forgings and how this composition controls elements related to through-thickness hardenability and negative effects from tramp elements (discussed in Sub-section 4.4.3.3 above, with respect to meeting UK legislative requirements). These additional controls meet ONR expectations, as outlined in previous GDAs, that ALARP measures above code compliance be considered, especially in the case of highest reliability components. I am satisfied that, from an SI perspective, the RP's proposed control over the chemical composition of the primary circuit products is capable of satisfying ONR's expectations through the lifecycle of the plant. This notwithstanding it remains my expectation that, at any site-specific stage, the licensee will need to ongoing materials data in line with the plant's operating safety case.

1019. Through the sampling of the materials selected for the SG, I have gained assurance the nickel-base alloys used in that component are suitably resistant to PWSCC. This has been further reinforced by the materials selection summary report, which makes provision only for the use of ERNiCrFe-7, a PWSCC-resistant grade, as a filler material for the welding of nickel-base alloys. My expectation for the use of PWSCC-resistant materials has been met for GDA. This expectation applies equally to materials choices to be made during site-specific phases and for areas not sampled directly as part of this GDA.

1020. The RP has proposed a neutron irradiation embrittlement surveillance scheme for the RPV. No similar schemes have been proposed for other components within the reactor plant, and I consider this to be appropriate given that it is only the RPV that will undergo significant neutron irradiation through-life. This scheme, which includes specimens to measure both conventional mechanical properties as well as fracture toughness, has been produced augmenting upon an accepted international standard; I judge that it is capable of supporting the safety case for the RPV through the proposed plant life. The fitness and appropriateness of the surveillance programme will need to be reviewed through the plant lifetime, this forms part of normal business for operation of the nuclear plant.

1021. Overall, from the information I have sampled, the materials selection processes, and the materials selected for the plant, have met my expectations. This is in terms of both materials properties and aging and degradation resistance, insofar as this applies to SI. I have identified instances where the RP's materials selection methodology has been applied inconsistently within the GDA; which I consider to be a minor shortfall that can be resolved as part of normal business during the site-specific stages. It is my expectation that monitoring of materials properties, aging and degradation will still need to be performed through-life in accordance with ONR expectations presented within the ONR SAPs EAD.3, EAD.4 and EMC.3 (Ref. 2).

4.5.2 Strengths

1022. The RP has developed a comprehensive and robust process for the identification, assessment and selection of materials for the metal SSCs important for safety. For the sampled components, the RP's process has demonstrated how a multidiscipline team of SI specialists have considered materials selection using a wide range of fundamental considerations, accounting for OPEX, challenges with manufacture, construction, inspection and through life operability.

1023. The RP has demonstrated how ONR expectations for reducing risks ALARP has been included in the materials selection process, and how design modifications have been implemented, where reasonable, to improve safety.

1024. The RP has developed an adequate programme for material surveillance and testing following a recognised international standard, to demonstrate through life materials integrity.

1025. The RP has considered relevant OPEX associated with the manufacture of heavy forgings. Assurance has been provided that issues of carbon macro segregation that have been observed in the international community, will be controlled through suitable technical qualification during the manufacturing of these components.

4.5.3 Outcomes

1026. The RP has responded well to the challenges raised by the structural integrity assessors as part of Step 4 of the GDA. The methodology presented for selection of materials has been applied, though this has not always been consistent, in a satisfactory manner. Where challenged, extra evidence has been provided to support and capture the decisions made by the RP. The RP has demonstrated an understanding of world experience in this area, how best to interpret this within a UK context, and how this might be used to make a suitable selection for any given material.

1027. The process, whilst not being consistent across components, has led to selection of materials that are within ONR’s general expectations. I have no generic concerns over any of the materials selected within the information I have sampled as part of this GDA.

1028. The application of the UK expectation to maintain risks ALARP has not been well developed, particularly in so far as demonstrating gross disproportionality, but the considerations presented demonstrate that the processes and methodologies applied are suitable.

1029. I have raised a minor shortfall related to the provision for the testing of irradiated HAZ samples and materials-specific evidence within the scope of the proposed irradiation embrittlement surveillance scheme.

4.5.4 Conclusion

1030. The RP has provided an overview of its materials selection process, notably through a materials selection methodology and a number of examples of how this has been applied. Although the methodology has not always been applied consistently, and the application of the UK's expectation to minimise risks to ALARP has not always been fully understood by the RP, in each of the sampled cases, the RP has been able to demonstrate how it has taken into consideration all reasonable OPEX and thought through what the failure modes and consequences might be for any given system. The materials chosen in each specific case lie within my expectations for the components sampled; the level of novelty in these materials chosen is low and materials similar to these exist within worldwide experience in other nuclear applications, for the sampled components. I am content that, for the components sampled, the materials selected are appropriate for use in the UK HPR1000.

1031. The RP has provided a surveillance strategy to monitor and assess the level of neutron irradiation damage to the reactor pressure vessel through the life of the component. This is broadly based upon international codes and standards but has been amended so as to address the UK expectation for direct measurement of fracture toughness. I judge that the scheme proposed is satisfactory, but that the licensee should ensure that the components with heat affected zones are suitably addressed. I consider this to be a minor shortfall.

4.6 Inspection

4.6.1 In-Process Manufacturing NDE

1032. The objective based NDE that is performed at the end of manufacture (Sub-section 4.3.5) is to support the HIC avoidance of fracture claim by providing high reliability detection and rejection of defects of structural concern. NDE is also performed at various stages of manufacture as defined by the RCC-M code, or in the case of the SG and RCP, by ASME III. I note that RCC-M code prescribes NDE during manufacture as a quality control tool to ensure the welding is being implemented correctly. Both codes require that any defect that is detected by the in-process inspection and is assessed as planar is repaired irrespective of the defect size.

1033. RCC-M opts for radiography over ultrasonic inspection for the volumetric inspection through coarse-grain austenitic stainless steel and Inconel alloy welds. Presumably RCC-M considers that the ultrasonic attenuation and beam distortion in these welds renders the reliability of ultrasonic inspection to be lower than radiography for manufacturing quality control. I sampled the in-process manufacturing NDE of the pressurizer SL (classified as a SIC-1) as an example of a non-HIC austenitic steel pipe with austenitic welds. The SL CSR (Ref. 35) states that radiography is the sole volumetric inspection of the welds. I note, however, that the RP does specify an ultrasonic volumetric inspection for the pressuriser surge nozzle to safe-end weld as a supplement to the code based radiographic inspection. Consequently, the RP recognises the benefits of ultrasonic inspection of coarse grain welds and also accepts that effective ultrasonic inspection can be developed. Furthermore, ultrasonic inspection of the nozzle to safe-end weld is more complex than the SL welds.

1034. I sampled the manufacturing inspection requirements for the dissimilar metal weld joining the accumulator injection line nozzle to safe-end as a further example of a coarse grain weld. I note that here too, radiography is only specified for the manufacturing inspection in the ‘Accumulator Component Safety Report’ (Ref. 36).

1035. Reliable detection of planar defects using radiography requires accurate alignment of the radiographic beam with the defect plane and relies on the defect having a sufficient gape (width) to present an image of the defect with an observable contrast. Radiography is good however at detecting volumetric defects. In my opinion, radiography alone is insufficient for the reliable detection of defects of structural concern and ultrasonic inspection should be considered as a supplementary tool.

1036. I reviewed the in-process manufacturing inspection arrangements for the MCL pipe-pipe and pipe-safe-end welds as examples of coarse grain HIC welds. Here, ultrasonic NDE has been included as a supplement to the RCC-M specified radiographing (Ref. 158).

1037. As a further example of a coarse grain HIC weld, I sampled the inspection arrangements for the steam generator primary nozzle to safe-end welds. The design and manufacture of the UK HPR1000 steam generator is defined by ASME III and similar to RCC-M, ASME opts for radiography over ultrasonic inspection for coarse grain welds. Unlike the MCL welds, a supplementary ultrasonic inspection had not been included for the primary nozzle to safe-end welds. I note that this weld is subject to high reliability ultrasonic inspection at the end of manufacture in support of the avoidance of fracture claim and the likely capability of this inspection has been demonstrated in GDA (Ref. 158).

AF-UKHPR1000-0209 – The licensee shall, as part of detailed design, justify whether it is reasonably practicable to use ultrasonic inspection for:

- austenitic stainless steel and Inconel alloy non-high integrity component welds; and
- those high integrity component welds for which the in-process inspection does not include ultrasonic inspection.

1038. The end of manufacture NDE of HICs will ultimately be subject to formal rigorous qualification by applying the principles of the ENIQ methodology (Ref. 146) (Ref. 165). The RP has presented evidence within GDA (see Sub-section 4.3.) to demonstrate that proposed inspection techniques when fully developed will be capable of detecting defects of structural concern. I note that qualified manufacturing inspections have been prescribed mainly for welds (along with a small number of welded items) and not for parent forgings or castings. I consider this is appropriate due to the substantially higher risk of weld failure due to manufacturing defects and that, in general, the defects that arise in parent material are more readily detected by the standard NDE techniques. Notwithstanding this, it is important that there is an appropriate level of confidence in the capability of the manufacturing inspections of the parent material to detect defects of concern. I therefore expect evidence is presented to support the capability of these inspections and supplementary techniques added where the capability is seen as falling short of the requirements.

1039. Similarly, through GDA, ONR has targeted its attention at the end of manufacturing of NDE for HICs. While the structural integrity claims for non-HICs are less onerous than HICs it remains important that an appropriate level of confidence is demonstrated in the capability of the manufacturing NDE to detect defects of concern. I therefore expect evidence is presented to support the capability of these inspections and supplementary techniques added where the capability is seen as falling short of the requirements.

1040. I have captured my expectation regarding the demonstration of manufacturing NDE capability for HIC parent material and non-HICs in the following assessment finding.

AF-UKHPR1000-0210 – The licensee shall, as part of detailed design, demonstrate that the in-process manufacturing non-destructive examination is capable of detecting the defects of concern. This should include both high integrity component parent material and non-high integrity component welds.

4.6.2 Pre-service inspection/In-service inspection

1041. PCSR Chapter 17 (Ref. 79) describes the principles of how in-service inspection (ISI) is to be used to provide early warning of degradation and to detect service induced defects before they grow to a size to be of structural concern. Pre-service inspection (PSI) is to be performed prior to start of operation to provide a baseline by which future ISI can be compared. The PSI is performed under the same conditions as the ISI and with similar equipment to confirm that the ISI can be applied in practice. It also provides a final demonstration that the component is free of significant defects at the start of operations.

1042. The PSI/ISI will be performed in accordance with RSE-M (Ref. 251). The licensee is responsible for determining a through-life examination maintenance inspection and testing (EMIT) programme that includes ISI. Consequently, the detailed scope of the PSI/ISI is outside of the scope of the GDA. There are, however, some important principles that I consider are within scope and which I have assessed here.

1043. In the 'Outline of PSI and ISI Requirements for UK HPR1000' document (Ref. 251), the RP states that within the UK regulatory context, ultrasonic inspection is preferred over radiography as the volumetric NDE method when searching for structurally significant (planar) defects. Radiographic detection of planar defects requires accurate alignment of the source with the plane of the defect and the detection performance falls off as the defect gape decreases. Consequently, the capability for tight (small gape) tilted defects may not be sufficient to reliably detect service induced defects. Furthermore, radiographic inspection presents a hazard during reactor outages and should be avoided if it is reasonably practicable to do so.

1044. RSE-M requires that ultrasonic inspection is to be applied as the volumetric inspection method for low alloy ferritic steel welds where the material has a fine, isotropic grain structure and the ultrasonic attenuation is low. In contrast, the code considers that ultrasonic inspection of austenitic stainless steel and Inconel alloys, where the weld material structure is coarse grained and anisotropic, is unreliable. Consequently, radiography is specified for PSI/ISI in these cases. The exception is the RPV where special ultrasonic techniques using large diameter focussed probes in immersion are applied.

1045. The RP has considered the UK context and has defined supplementary ultrasonic inspections for HIC austenitic and Inconel alloy welds. I note that these are specified in addition to radiography and not as a replacement. The RP's process (Ref. 251) does not include any supplementary inspections for non-HIC austenitic and Inconel welds and radiography is maintained as the sole volumetric method. For example, the ISI of the steam generator primary nozzle to safe-end HIC weld includes radiography (RSE-M) and ultrasonic (supplementary) methods, whereas ISI of the pressuriser surge line to safe-end (SIC-1) includes only the RSE-M code requirement of radiography. I note that ASME XI, the equivalent ISI code to RSE-M, differs in this respect and requires ultrasonic inspection for a much wider scope of austenitic and Inconel alloy welds. In addition, from my assessment of the SG codes and standards (Sub-section 4.4.2.2) it is important to consider the risks at the code interfaces when combinations of codes are used. This should inform the PSI/ISI strategy for the SG and RCP components.

1046. The UK experience is that effective and reliable ultrasonic NDE can be developed for austenitic and Inconel alloy welds and is applied to UK PWR. I therefore expect the licensee to consider the wider use of ultrasonic inspection as a replacement to radiography for the ISI of austenitic and Inconel alloy welds. Furthermore, where ultrasonic inspection is specified as supplementary to radiography for HIC welds, the licensee should consider ultrasonic inspection as a replacement to radiography.

AF-UKHPR1000-0211 – The licensee shall, as part of detailed design, justify the application of ultrasonic inspection to replace radiography in the pre-service/in-service inspections for high integrity and non-high integrity components and austenitic and Inconel alloy welds, including austenitic and Inconel alloys and dissimilar metals at code interfaces.

1047. Assessment of the RP's ALARP assessment report states how the design has been considered and, in some cases modified, to promote effective ISI and this is discussed in Sub-section 4.7

1048. I note that the PSI/ISI of HIC welds will be qualified using ENIQ principles similar to those applied for the objective based end of manufacture ultrasonic inspections. I am satisfied that this will provide an appropriate level of confidence for the capability and reliability of the ISI to forewarn of failure from service induced degradation. It is also important that an appropriate level of confidence is derived for the capability of the ISI of non-HICs. It is accepted that this may not attract the full rigour of qualification and the level of demonstration be linked to the structural integrity classification.

AF-UKHPR1000-0212 – The licensee shall, as part of detailed design, demonstrate that the pre-service/in-service non-destructive examination for non-high integrity component welds is capable of detecting the defects of concern.

4.6.3 Design for Access and Inspectability

1049. Access for NDE and 'design for inspectability' are prominent nuclear safety considerations. Design for access and inspectability is a concept where the requirements for NDE are explicitly considered in the design of the item to be inspected. Reference designs that are submitted for a GDA may not have anticipated this concept and it is expected that, where appropriate, design modifications will be considered to improve the performance of the inspection. The extent to which these modifications are considered will depend upon the structural integrity classification of the structure or component and the role of the NDE in assuring structural integrity. The designer should take a balanced view such that any design modifications should not be to the detriment of the overall demonstration of structural integrity.

1050. It is expected that the concept of design for access and inspectability is applied to structures and components of all safety classes in a proportionate or risk informed manner, with the requirements for highest reliability (HICs in the RP's SI classification scheme), being the most onerous. These expectations are supported by the following ONR SAPs (Ref. 2):

- EMC.8: Geometry and access arrangements should have regard to the need for examination.
- EMC. 13: Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated examined and maintained throughout the life of the facility.
- EMC.27: Provision should be made for examination that is capable of demonstrating with suitable reliability that the component or structure has been manufactured to an appropriate standard and will be fit for purpose at all times during future operations.

1051. The RP provided a review of the available access for manufacturing NDE, PSI and ISI of HICs (Ref. 118) and (Ref. 252) during Step 3 of GDA. These reviews were based largely on the UK HPR1000 reference design of Fangchenggang-3. However, the ONR SI assessor highlighted that these reports did not fully demonstrate that all reasonably practicable options had been considered as part of the design for inspectability. Consequently, regulatory observation RO-UKHPR1000-0022 (Ref. 49) was raised to explain the expectations regarding design for access and inspectability and to ensure there is sufficient confidence that the requirements for manufacturing NDE and PSI/ISI have been adequately considered.

1052. The regulatory expectation is that the design for inspectability principle should be considered for all components where NDE supports nuclear safety. Embedded in this principle is the recognition that the extent to which the design may be adapted for NDE should be commensurate with the item's nuclear safety classification and the role that NDE plays in the structural integrity safety case. Consequently, it is expected that HICs would attract the greatest attention when considering design options and the RO was structured in the form of two actions:

- RO-UKHPR1000-0022.A1 – Considerations to enhance the reliability of NDE for high integrity components in UK HPR1000.
- RO-UKHPR1000-0022.A1 – Considerations to enhance the reliability of NDE for non-HICs in UK HPR1000

1053. A more complete assessment of the RPs submissions in response to RO-UKHPR1000-0022 is provided in the 'Assessment of the Response to RO-UKHPR1000-0022 – Design for Access and Inspectability' (Ref. 253), where I have summarised the main findings and conclusions from my assessment.

1054. The RP addressed each action in a similar way with a series of submissions for HICs and a parallel set of documents for non-HICs, and for each action, the RP produced criteria by which the access and inspectability should be assessed. This was followed by an assessment of the component against these criteria which concluded whether NDE was impeded or not (Ref. 254) (Ref. 255) (Ref. 256). In those cases where the access or the design did impede the NDE, the RP considered whether it was reasonably practicable to modify the generic design to overcome or reduce any impediment to the NDE (Ref. 256) (Ref. 257) (Ref. 258) (Ref. 259) (Ref. 260). Finally, where design improvements were considered to be reasonably practicable, the RP performed an optioneering exercise to identify the preferred solution.

1055. My assessment of the RP's response to the regulatory observation was targeted towards HICs and considered one example of a non-HIC to provide evidence of the RP's overall approach.

1056. I reviewed revision B of the requirements and guidance for HIC components (Ref. 254) that defined the criteria the RP would apply in its assessment of whether NDE would be impeded by the extant design of HICs. I clarified some aspects of the criteria through RQ-UKHPR1000-0917 (Ref. 28) to which the RP responded positively and updated the requirements and guidance document (Ref. 256).

1057. Overall, I was satisfied that the RQ response and the revised document provided a sound basis for assessing the access and inspectability conditions for HICs in relation to the high reliability claims made for NDE.

1058. The requirements and guidance for non-HICs (Ref. 256) were similar to those for HICs and essentially contained the same criteria regarding physical access and scanning conditions. It is in the application of the guidance and the degree to which design modifications are considered, that reflects the differences in NDE reliability sought for HICs and non-HICs.

1059. The conditions for access and inspectability are presented in the 'Application of Weld Ranking Procedure' (Ref. 107) (for the end of manufacturing inspections) and in the 'Access and Inspectability Assessment' (Ref. 261) (for PSI/ISI). In general, the conditions for PSI/ISI are worse than those for the manufacturing inspection where more surfaces are available and access is less restrictive. I note that the 'Access and Inspectability Assessment' (Ref. 261) concludes that any limitations are mostly due to the component design, rather than the physical access to the component.

1060. The RP took forward those items for which access or scanning restrictions have been identified in a series of 'ALARP justification reports' (Ref. 257) (Ref. 258) (Ref. 259) (Ref. 260). Here it considers options for improving the generic design to facilitate NDE. Table 11 presents those HICs for which scanning limitations were identified along with the design improvements arising from the optioneering studies. These design modifications described below have been accepted for inclusion within the UK HPR1000 GDA (Ref. 80) (Ref. 262).

1061. In reviewing the RP's response to RO-UKHPR1000-0022 (Ref. 49), I concluded that appropriate guidance and criteria had been developed for assessing access and inspectability and that suitable design improvements had been identified to overcome scanning restrictions (Ref. 253).

Table 11: HICs for which scanning limitations were identified

Component	Limitation	Design Improvement
Main coolant line welds	The length of flat surface adjacent to some welds was insufficient to	The orientation of the primary nozzles is modified to allow a

Component	Limitation	Design Improvement
	allow continuous scanning of ultrasonic probes.	longer straight section of the pipes adjacent to welds
	The position of the counterbore was too close to the weld, thereby limiting inspection from the bore and generating confusing echoes	The counterbore region was extended thereby eliminating these restrictions
RPV upper dome to flange weld	Lifting lugs on the upper dome restrict scanning at some circumferential position.	The welded dome to flange was replaced by a single piece forging thereby removing the need for a weld inspection.
Main steam line welds	A radiographic access hole restricted the scanning of welds 10,11, 12 and 13.	The access hole was removed thereby removing the scanning obstruction
	The counterbore region was too short for welds outside of the containment, which could generate confusing signals.	The position of the counterbore transition was moved to 70mm from the weld centreline.
	There was a tapered region on the valve side of the MSIV to pipe weld, that would interfere with probe scanning.	The tapered section of the surface was moved away from the weld to facilitate scanning from the valve side.
	Inspection of the steam generator secondary outlet nozzle safe-end to MSL pipe weld was restricted due to a change of section.	The straight section adjacent to the weld was increased thereby improving the scanning access.
Pressuriser upper/low dome to shell weld	There was no straight section on the dome side of the weld, thereby presenting difficulties for the detection of transverse defects.	A flat region of 80mm was applied to the dome surface to facilitate circumferential scans.
Pressuriser nozzle to dome welds	The surge line nozzle to dome weld inspection is complicated by the geometry.	The dome to surge line nozzle weld was replaced by a single piece forging, eliminating the requirement for a weld inspection.
Steam generator feedwater nozzle to transition piece weld (non-HIC weld)	The available scanning distance did not satisfy the criteria of the guidance document	The nozzle to transition piece was replaced with a single nozzle forging thereby removing the need for a weld inspection.

4.6.4 Strengths

1062. The RP has described a comprehensive set of NDE inspections during manufacture and in-service to support its structural integrity claims.

1063. The manufacturing scope of NDE includes code-based in-process inspections and a final objective-based inspection performed at the end of manufacture to support the avoidance of fracture claim for HICs. The RP has demonstrated that it understands the UK regulatory expectation of qualifying the objective based inspections to demonstrate that the reliability of NDE is commensurate with the high integrity claim. The proposed use of the ENIQ methodology to qualify NDE is recognised in the UK as good practice.

The RP's proposed arrangements for inspection qualification were also aligned with recognised UK practice.

1064. I noted that the RP chose to present evidence for the likely performance of the qualified NDE using ENIQ style documents. While this was not necessary for GDA, this exercise has provided the RP with a sound basis of taking the inspections through future qualifications post GDA. While the ENIQ methodology will be used for qualifying these inspections it will be the responsibility of the licensee to ensure that the detailed arrangements, including the selection of the qualification body, are effective.

1065. The detailed arrangements for PSI/ISI lie with the licensee to take forward and are out of scope for the GDA. I did note however, a commitment with the RP's submissions that the PISI/ISI of HICs will be formally qualified using the ENIQ methodology.

1066. The RP developed a structured process for identifying restrictions to implementing reliable NDE (principally scanning restrictions for ultrasonic inspection) and assessing whether it was reasonably practicable to improve the design to facilitate NDE. This has resulted in several design modifications for the UK HPR1000 that will improve the overall reliability of NDE.

4.6.5 Outcomes

1067. In spite of the RP's process for the review and approval of the GDA technical justifications, the documents provided to ONR for assessment had internal inconsistencies, errors and inappropriate application of NDE techniques. It appears that the self-checking and verification by the RP's contractors and the subsequent approval by the RP was inadequate in these cases. Ultimately, the objective based inspection systems for HICs will be assessed and qualified by an independent qualification body. In the early stages, the RP employed a recognised qualification body to review the GDA technical justifications and act as a limited GDA qualification body. I noted that in these cases, the qualification body played a significant role in enhancing the quality of the documents.

1068. The RP's submissions did not fully address some aspects of the NDE of non-HICs by:

- not including ultrasonic inspections in both manufacturing NDE and PSI/ISI in some cases for detecting defects of concern; and
- not describing how it would demonstrate the capability of the NDE for non-HICs and the parent material of HICs.

1069. I have produced four assessment findings (AF-UKHPR1000-209 to AF-UKHPR1000-212) for the licensee to take these matters forward.

4.6.6 Conclusion

1070. Overall, I am satisfied that the RP has presented sufficiently detailed proposals for NDE inspection that support the structural integrity claims in the PCSR and supporting documents. These proposals will need to be implemented by the licensee and I have defined several assessment findings that will assist the licensee in reducing risks to be ALARP.

1071. The RP has presented a tiered approach to NDE development and qualification that considers the structural integrity classification and the role of the NDE in supporting the integrity.

1072. I am satisfied that the RP has paid due attention to the 'design for inspectability' concept and has committed to several design modifications to improve NDE reliability.

4.7 Demonstration that Relevant Risks Have Been Reduced to ALARP

1073. Within Step 2 of GDA (Ref. 6), I was broadly content that the RP was developing a reasonable understanding of ONR's expectations with respect to reducing risks ALARP for the SI safety case. Within Step 3 of GDA, I continued to engage with the RP, to track its progress in developing the ALARP demonstration.

1074. At a high level, the RP has produced the 'ALARP Demonstration Report of PCSR Chapter 17' (Ref. 27), which presents a holistic ALARP assessment of the SI aspects of the UK HPR1000 design, including a review against UK and international RGP and OPEX. The RP states that this report provides evidence to support Claim 3.4.8 of the PCSR:

"Claim 3.4.8 All reasonably practicable options to improve nuclear safety have been adopted, demonstrating that the risk is ALARP."

1075. Within this document the RP states that it has developed a methodology for the ALARP demonstration for SSC structural integrity for UK HPR1000. This is to demonstrate an understanding of the nuclear safety risk associated with structural failure of metallic components and demonstrate that the risk is appropriately controlled. The RP states that the ALARP demonstration is set in three tiers:

- Tier 1 -The top tier is the Pre-Construction Safety Report (PCSR) chapter. Sub-chapter 17.8 of the PCSR presents the safety case claims as well as the methodology for demonstrating that the risk of structural failure is ALARP.
- Tier 2 -The second tier is the ALARP demonstration report for PCSR Chapter 17. A high-level review of the UK HPR1000 design against RGP and OPEX in order to identify potential gaps.
- Tier 3 -The third tier are the lower level supporting documents which address the gaps against RGP and OPEX.

1076. The RP recorded any gaps between the design and RGP/OPEX and considered potential enhancements to address these gaps. The gaps identified included:

- optioneering to determine whether it is reasonably practicable to avoid classifying components as HIC by providing additional protection or mitigation measures.
- ALARP optioneering is required in order to determine whether it is reasonably practicable to reduce the number and the length of welds.
- It is necessary to identify any significant changes which have been introduced via the more recent versions of the codes and standards and determine whether these have any implications for the design.
- It is proposed that the design, operation and maintenance of the SG for UK HPR1000 will make use of a combination of different codes. An ALARP review is required to determine whether the combination of codes for the design operation and maintenance is acceptable to identify any additional measures may be warranted.

1077. The RP identified the need to perform optioneering on the high-risk areas identified in the ALARP report, with a number of options identified, evaluated and implemented to optimise the design and further reduce risk.

1078. I reviewed the development of this throughout Step 3 of GDA and noted that the methods that have been applied in the review of OPEX and RGP appear adequate, but there are areas where this appears to be comprehensive (e.g. design challenges for major vessels and piping).

1079. At the close of Step 3 of GDA I was broadly content that the RP appeared to be using a reasonable process for reducing risk ALARP, with continued effort to develop its ALARP position within the structural integrity area. It was clear that the RP had made positive improvements to the ALARP demonstration, however I consider that the detail of this demonstration should be reviewed across the range of SI components, to ensure the information presented is complete, proportionate and there is a holistic ALARP position. This has been the focus of my assessment of the RP's application of its methodology to reduce risks ALARP.

4.7.1 Assessment

1080. The ALARP position initially presented by the RP was predicated on compliance with code and other guidance and publications from external bodies. I note that the demonstration of ALARP and the demonstration of code compliance are not equivalent. Whilst it is ONR's expectation that relevant codes and standards are complied with, either directly or through equivalence, the concept of ALARP is such that all reasonably practicable measures, that are not grossly disproportionate, should be taken to reduce risk.

1081. From the conclusions drawn at Step 3 of GDA, it was recognised that there was a need to address significant ALARP considerations in the UK HPR1000 design and safety case, specifically associated with the design of the RPV, PZR, SG, MCL and RCP (Ref. 7). It was expected that this may involve the need for complex balances between minimising the number and length of welds, whilst retaining adequate material properties in thick section forgings.

1082. It is generally accepted that the areas where gross failure would most likely lead to intolerable consequences are mainly large diameter welds. ONR expects that there is a balance between reducing the number and length of welds and ensuring adequate material properties (ONR SAP EMC. 9 vs. EMC.13 & 14, Ref. 2). There is also an expectation that where welds important for safety cannot be avoided, then they are readily accessible for pre- and in-service inspections (ONR SAP EMC.27, Ref. 2).

1083. As part of my assessment strategy, I have reviewed the application of the RP's methodology for reducing risks ALARP across several technical topics within SI. These are primarily associated with:

- Structural integrity classification – I consider the RP submissions relating to the structural integrity classification of key SSCs, which has included the results of consequences assessments and companion ALARP studies to ensure the structural integrity classification is justified.
- General ALARP considerations – I consider the RP's application of ALARP to the demonstration that risks from structural integrity have been reduced ALARP for areas such as code selection, design features and materials selection.

Reducing Risks ALARP Through SI Classification

1084. This matter has been addressed through the regulatory observation process and sampling of certain components. Maintaining risks ALARP, ensuring that all options have been considered and implemented where reasonably practicable, have been considered through the actions raised under Regulatory Observations RO-UKHPR1000-0008 and RO-UKHPR1000-0058 (Ref. 49). These refer to classification of the main coolant lines and main steam lines, respectively. The RP has presented sufficient evidence to address the actions raised over consideration of ALARP in the SI classification process for the UK HPR1000. The ROs were successfully closed, with a number of residual matters raised. These residual matters have been discussed accordingly in Sub-section 4.2 above, where the reasoning and judgements made with respect to ALARP have been presented.

1085. In summary, I am satisfied that the RP has demonstrated how reducing risks ALARP has been considered in the SI classification of highest reliability components, in accordance with the ONR expectations (Ref. 2) (Ref. 5).

1086. For components of a lower safety significance (i.e. non-highest reliability), ONR expectations are that risk should still be reduced ALARP. As part of my assessment of the RP's approach for SI classification (Sub-section 4.2 above), I raised RQ-UKHPR1000-0670 (Ref. 28) asking the RP whether all reasonably practicable steps to minimise risk have been taken into account when considering the Steam Generator Blowdown lines, and how ALARP decisions involving differing levels of risk will be captured in the structural integrity case.

1087. The RP referred to the company document provisions on the optioneering process (Ref. 263). I noted that this is a cross-discipline submission and hence I consider it appropriate to test the applicability of this process through specific applications. In this case, the RP has presented the application of the 'Provisions on Optioneering Process for UK HPR1000 Generic Design Assessment GDA Project' (Ref. 263) to the SG blowdown lines in its response to RQ-UKHPR1000-0670. I consider this to be a relevant sample of how the optioneering process is applied.

1088. For the SG blowdown lines, the RP presented three options in its response to RQ-UKHPR1000-0670 (Ref. 28), which are presented in (Ref. 47). Comments made against (Ref. 47) within this report are from the Structural Integrity discipline viewpoint only and do not prejudice the views of other disciplines. The three options referenced are:

- Option 1 - Retain the extant design.
- Option 2 - Rearrange the blowdown lines.
- Option 3 - Modify the isolation valves.

1089. The RP concludes in (Ref. 47) that classification of the SG blowdown lines is SIC-2 for the selected design Option 3. This has been performed by comparing benefit and detriment for the three options above in terms of Safety (including nuclear safety), Environmental, Technical and Cost & Schedule. I have not sampled this analysis in detail but I am content that this is an example of how the methodology might be applied for a lower classification (SIC-2) component. I note that the balance of Safety against Cost & Timescale was 5:1 (50% Safety consideration against 10% Cost and Timescale consideration). I judge that this is appropriate for this classification of component, but that a greater demonstration of disproportion will be needed for higher classification components.

General ALARP Considerations

1090. As part of the ONR SI GDA Step 3 report (Ref. 7) it was noted that the scoring in the optioneering processes was judgemental and, therefore, could only provide an indication of the preferred option in any optioneering exercise. Using technical judgement within an optioneering process is normal and accepted by ONR. It is, however, ONR's expectation that, where an optioneering process returns two or more options with similar scores, the duty holder should ensure that the selected outcome is robust. This is commonly done using sensitivity studies, where reasonable variations are made to the input scorings in the optioneering study to see if this has a significant effect on the output.

1091. I raised RQ-UKHPR1000-0670 (Ref. 28) asking the RP to provide clarity on the how the ALARP principles have been applied and actioned in the following areas:

- Comparing options offering different risk reductions and what is required to select an option of higher risk.

- Making ALARP decisions relating to materials selection.
- How ALARP decisions involving differing levels of risk will be captured in the structural integrity case.

1092. The RP noted that different options offering risk reduction would be delt with through the RP's ALARP methodology programme (Ref. 264). This is intended to balance competing areas of risk and provide an overall position where the risk in maintained ALARP for the system as a whole. I judge that this is acceptable as a process and provides sufficient assurance for the Structural Integrity review. This notwithstanding, I note that the overall acceptability of the process needs to be sampled and tested, and that an acceptable result within the Structural Integrity discipline does not prejudice the assessments in other disciplines, where its application may be different.

1093. Optioneering has been used across the SI discipline to provide a view on the materials and designs to be employed. I have sampled the optioneering used in the materials selection process, the specifics of which are discussed further in Sub-section 4.5 of this report. In this section of the report, the generic ALARP position only is considered, with details being available in Sub-section 4.5. The link between the ALARP demonstration and the overarching methodology for ALARP (Ref. 264) is not made in any detail. Notably, the need to demonstrate gross disproportionality for highest reliability components is not made and the weightings used are neither explicitly in line with ONR's expectations for highest reliability components nor used consistently. I have made this observation when sampling several components specific assessments for ALARP consideration in component design, as discussed below from paragraph 1124ff of my report. This notwithstanding, in Sub-section 4.5, I have judged the materials selection process to be adequate, despite these shortcomings.

1094. Given that the ALARP methodologies have produced acceptable results in the materials and SG blowdown analyses, I judge shortfalls in these areas constitute a minor shortfall. It remains that my expectation for demonstrations that risks are reduced to ALARP, including the demonstration of gross disproportion where appropriate, need to be made robustly and explicitly, especially for components of the highest reliability.

Consideration of ALARP in Design Code Selection

1095. It is important that the design of SSCs accounts for the burden of proof required to meet safety case expectations, particularly for instances where highest reliability is considered unavoidable. Therefore, it is a fundamental expectation that appropriate OPEX and RGP has been considered in the design of components, a significant proportion of which is dependent on the use of relevant design codes.

1096. I have sampled the RP's approach for selection and application of design codes and standards in the context of reducing risk ALARP under Sub-section 4.4 of this report.

Consideration of ALARP for Design Features

1097. Even with selection of an appropriate design code, ONR expects that the design of an SSC can be demonstrated to have considered and sentenced options to reduce risk of gross failure ALARP. From an SI perspective, this is more significant in instances where a claim of highest reliability is being made, such that failure of the SSC is considered so remote it can be discounted. To support this, it is important that the choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component (ONR SAP EMC.9)

1098. RO-UKHPR1000-0022 (Ref. 49) relates to design for inspectability, which includes aspects of minimisation of the need to inspect and hence, implicitly, the drive towards minimisation of the number and length of welds. The assessment of RO-UKHPR1000-0022 is discussed in more detail in Sub-section 4.6 above, with a brief summary of the modifications implemented during GDA provided in Table 11.

1099. In addition to these design changes, several other modifications have been bought about following ALARP assessments conducted by the RP relevant for SI safety significant components, which are described below.

RCP Casing Design

1100. The RP had stated that it intended to use an austenitic casting for the RCP casing. ONR has a preference for the use of forged components where reasonably practical (ONR SAP EMC.9). This is because of the potential difficulties in the achievement of adequate material properties in castings; the challenges associated with achieving confidence in inspection qualification; and the justification of weld repairs, with their attendant residual stresses. In practice, this means that developing the structural integrity cases including avoidance of fracture demonstrations for RCP casings (and other components e.g. valve bodies) using cast materials has proven challenging for claims of highest reliability.

1101. From a review of available OPEX and the application of the RP's ALARP review process, the RP concluded that the use of forged carbon steel pump casing with welded support lugs (Option 2) is appropriate for the UK HPR1000. This represented a change from the reference position of a cast stainless steel case (Option 1). The result of the ALARP process was documented in 'ALARP Assessment Report of Reactor Coolant Pump Casing' (Ref. 265).

1102. Whilst I was content with the preferred option identified by the RP, I had a query over the completeness of the ALARP study. It is ONR's expectation that all benefits and dis-benefits of each option are considered within the ALARP optioneering. This ensures that a complete representation of the risks posed by the identified options can be compared. During Step 3 of GDA, I was also not certain that the RP had considered the true through life burden of welds in the decision-making process.

1103. I was of the opinion that an additional risk had been introduced through the selection of the forged carbon steel RCP casing that has not been accounted for within the comparison. Specifically, with the selection of a carbon steel material instead of an austenitic material an additional six dissimilar metal welds will be introduced at the connection between the safe end and MCL. For emphasis, the focus of my challenge was to ensure that the decision-making process is robust and all relevant risks have been identified, such that the option selected reduces risk ALARP to meet the legal expectation. This challenge was conveyed to the RP in RQ-UKHPR1000-0418 (Ref. 28).

1104. In response, the RP acknowledged that the characteristics of dissimilar metal welding had not been evaluated as an analysis dimension. The RP committed to update the RCP ALARP report and explained that in view of the two (six in total for the UK HPR1000) additional HIC dissimilar metal welds, the replacement of welds between cast casing nozzles and forging reactor coolant pipes, reduces the difficulty of NDE and associated technical justification of welds. Hence, it was the RP's position that the consideration of the risk posed by dissimilar metal welds should not affect the final conclusion of the selection of pump casing manufacturing process.

1105. Overall, the RP argued that the proposed design represents a balance of various influence factors such as minimising the numbers and length of welds, OPEX, maximum forging/rolling capability, maximum heat treatment capability, and homogeneous material properties. Whilst these appear to be predominantly associated with manufacture and pre-service issues, I noted from the ALARP assessment that the

RP had considered the burden associated with through life inspections of welds, such as operator dose, confined space working and the financial costs associated with completing ISI.

1106. ONR received the revised 'ALARP Assessment Report of Reactor Coolant Pump Casing' (Ref. 266) which showed an updated consideration of technical and safety factors within the ALARP scoring and weighting criteria. The assessment takes account of the additional of dissimilar metal welds and scores this with respect to burden of additional welds to control and inspect. The updated ALARP assessment also includes a sensitivity study, by where weighting factors associated with safety and technical criteria are increased compared to cost and time scale factors. The outcome shows that the forged component is still identified as the highest scoring option.

1107. Overall, I am satisfied that the RP has conducted a thorough assessment of the RCP casing and has adequately identified and selected design opportunities to reduce risk ALARP. The assessment has been conducted by a multidiscipline team of experts, who have considered relevant OPEX and technical/engineering factors to determine the benefits/disbenefits of the options available. Through a process of engineering judgement and consideration of available evidence, the options have been scrutinised and rejected to that which the RP considers reduces risk ALARP.

Core Barrel Design

1108. In Step 3 of GDA, the RP was still considering the SI classification of the RVI, the outcome of which was needed in order to set expectations for demonstrating that risks have been reduced ALARP. The adequacy of the final classification (SIC-1) is discussed under Sub-section 4.2 of my report.

1109. From the perspective of reducing risk ALARP, in Step 3 of GDA the RP had considered two options for manufacturing the core barrel:

- Option 1: Rolled and welded plate segments.
- Option 2: Forged and welded cylinders.

1110. The latter option was noted to be beneficial through removal of the longitudinal weld, however it introduced challenges associated with the size of forging capacity required and risks associated with managing material properties during manufacture.

1111. I concluded that further investigation was necessary to determine whether the RP had completed a robust review of the benefits/disbenefits of both options to confirm that risks have bene reduced ALARP.

1112. The RP has completed this task and produced its findings within 'ALARP Assessment Report of Core Barrel Longitudinal Weld' (Ref. 267)

1113. In brief, the ALARP assessment covers a wide range of considerations, that take into account ONR expectations (Ref. 2) for product form (ONR EMC.9) and weld position (ONR EMC.10). This document applies the RP's methodology for multi-discipline input for the consideration of OPEX, safety aspects, capability, cost and time. Taking cognisance of my previous comments, I consider this ALARP review is similarly devoid of considering OPEX outside of the RP's native experience.

1114. In some instances, the RP refers to the length and number of welds between the two options through comparison to the EPR™ design at Taishan. The RP claims that the forged approach will require an additional circumferential weld, which will nullify any benefit made from removing the three longitudinal welds associated with Option 1. The RP recognises the disbenefit of having longitudinal welds in the core barrel, by stating that there is "3700mm longitudinal weld in core active region". The RP also considers the similar disbenefit for Option 2, claiming that "according to the OPEX of Taishan project, for the forging is difficult to manufacture, there is a significantly high risk of adding a circumferential weld in the core active region".

1115. From my review, I am unclear as to how relevant this is as a metric for the ALARP assessment, given the comparison appears to be made directly between the EPR™ core barrel and the reference design core barrel. As such, this would result in design-specific measurements being used, which is not a true comparison given the increased overall height and diameter of the EPR™ core barrel. It is also not immediately apparent how detrimental the additional circumferential weld is for Option 2, by direct comparison of the additional length of weld added.

1116. The outcome of the RPs scoring favours Option 1 (score – 93) as preferential to Option 2 (Score 89). In my opinion, this is not significant difference and does not warrant an instance of gross disproportion. Whilst there may be some ambiguity about the robustness of the scoring criteria and balanced comparison between options, I consider the outcome still aligns with ONR's expectations for ONR SAP EMC.9 (Ref. 2). My judgement is compounded with the outcome of the RP's classification of the RVI as SIC-1, which identifies failure of the longitudinal welds as being bounded by circumferential weld failure, such that removal of the longitudinal weld is unlikely to reduce the risk significantly for the remaining, bounding case.

1117. I am therefore satisfied that for the sampled case of the Core Barrel design, the RP's consideration of reducing risk is appropriate and proportionate with the assigned safety classification.

Fuel Transfer Tube Design

1118. The FTT design was initially considered in response to the RP's approach to SI classification, where the RP applied it's ALARP methodology. My assessment of this is provided in Sub-section 4.2 above, which describes how the FTT design was optimised to reduce the likelihood and consequences of FTT failure, through the minimisation of welds and improvements to the civil structure. This enabled the RP to revise the safety claim made on the FTT, reducing risk to ALARP.

1119. In conclusion, I am satisfied that the RP has completed a fair and balanced assessment of the options identified to reduce the risk ALARP through component design modification, resulting in revision of the SI safety classification.

Consideration of Engineered Protective Measures Within the Safeguards Building

1120. As has been discussed within RO-UKHPR1000-0058 and RO-UKHPR1000-0046 (Ref. 49), the RP has proposed additional physical protective measures to reduce the consequences of pipe whip and jetting causing damage to HICs in the safeguards building.

1121. From the consideration of ONR guidance regarding the choice of product form of metal components enabling examination and minimising the number and length of welds, I note that the RP's ALARP process has resulted in the SG Main Feedwater Nozzles now being constructed as a single piece forged nozzle, in contrast to the previously proposed multi-piece welded nozzle. Also, the RPV upper dome has changed from the RP's standard design to be a single piece forging. These two examples align directly with ONR SAP EMC.9 (Ref. 2) and give direct evidence that the RP has given due consideration to the minimisation of number and length of welds in the components sampled.

1122. I have confidence the submissions and designs from the RP are adequate for the purposes of GDA. This is based upon the analyses made in this report, and in response to RO-UKHPR1000-0008, RO-UKHPR1000-0058 and RO-UKHPR1000-0022 (Ref. 49), demonstrating that, for the components sampled, the welds have been designed to adequately facilitate inspection. I note that this has not been the subject of a direct challenge from ONR but can reasonably be inferred from related submissions.

1123. I judge, therefore, that ONR SAP EMC.9 (Ref. 2) has been addressed adequately within GDA. I recommend that, during the site-specific stages, the ONR Structural Integrity inspector should ensure that similar considerations have been made for other high safety significance components and that this is recorded appropriately.

1124. Within RO-UKHPR1000-0022 (discussed under Sub-section 4.6 above), the RP has demonstrated how the design of various SI classified components have been evaluated and modified to improve access for inspectability, thus reducing the risks associated with manufacture and through life degradation.

Consideration of ALARP in Materials Selection

1125. I sampled three components from the UK HPR1000 to assess whether the RP's ALARP process had been applied properly to the materials selected for the UK HPR1000. These were the RPV, the PZR and the SG forgings, covered by RQ-UKHPR1000-1237, RQ-UKHPR1000-1400 and RQ-UKHPR1000-1401 respectively (Ref. 28). These RQs were in addition to those raised in Step 2 of GDA e.g. RQ-HPR1000-0165.

1126. The ALARP position of the forgings were assessed by the RP using a conventional method balancing benefits and detriments of different design options in terms of safety and manufacture. This involved a series of relative rankings, which were subsequently summed using weightings. This is within ONR's expectations but is noted to have elements of subjectivity within the process.

1127. In all three cases, I noted that the methodology used to apply weightings for different sections of the component was not well explained, and the final weightings used were not demonstrably in line with ONR's expectations in terms of showing gross disproportionality between cost and benefit for highest reliability components. For all three components, I challenged the weighting process used. For the RPV I noted that the waiting for nuclear safety was 45%; Conventional safety 5%; technical 30%; Economic 10%; time scale 10%. The RP stated that these weightings have been arrived at in conjunction with its UK based consultant, and are intended to balance not just structural integrity but also other areas of reactor plant safety. They state that the inclusion of economic and time scale considerations at only 10% is considered appropriate, as this is significantly lower than safety and technical.

1128. I judge that, although the strict criteria of a 10-to-1 test of gross disproportionality has not been explicitly demonstrated here, the overlap between safety and technical areas in terms of delivering nuclear safety means that safety considerations are considered significantly more than time scale and cost. The RP has performed some degree of sensitivity study in this area, bringing economic and time scale considerations down to 5% each. This considered, along with the demonstration by the RP that even in its sensitivity cases the outcome of the ALARP consideration has not changed, I judged that the RP has made an adequate case.

1129. Across all three of the components sampled, a similar argument was presented by the RP regarding the weightings applied. In my opinion, that there is no apparent link back to a more structured process on how to perform ALARP assessments, especially within the structural integrity discipline, but do not consider that this is a significant shortfall within the case presented by the RP.

1130. Moreover, the RP has provided evidence that they have considered how ALARP has been treated in previous GDAs and applied it within its considerations in this case. I do not have any concerns over the scorings arrived at and do not believe any to be counterintuitive. Therefore, when combining them with the weightings discussed above, I am content that within the remit of the GDA and from an SI perspective, the RP has adequately demonstrated that risks have been reduced ALARP.

4.7.2 Strengths

1131. Throughout the GDA, the RP has developed its understanding of ONR SI expectations for reducing risks in the design of the UK HPR1000 metallic SSC. This has resulted in an improved methodology for consideration of ALARP, taking into account UK specific OPEX of how previous RP's have demonstrated consideration of ALARP within GDA.

1132. The RP has provided multiple examples of ALARP assessments that show good use of multi-discipline input at the design stage, underpinned by sound reasoning and balanced judgement. The RP has also demonstrated a proactive approach to reducing risks ALARP, by implementing improvements to the design of SSCs outside of those discussed initially with ONR.

1133. The RP's ALARP methodology has made good use of sensitivity studies for higher safety significant components, to provide further assurance that the outcome identified reduces risk ALARP

4.7.3 Outcomes

1134. Whilst the RP has made significant progress with improving its understanding of ALARP and UK expectations for reducing risk, I have identified several topics that I consider will benefit the licensee with respect to demonstrating SI risks are reduced ALARP.

1135. I consider that the methodology used by the RP to apply weightings for different sections of the component is not well explained in the provided information. The weightings used in the ALARP assessments are not fully in line with ONR's expectations in terms of showing gross disproportionality between cost and benefit for highest reliability components. In my opinion, that there is no clear link back to a more structured process on how to perform ALARP assessments or designation of criteria weightings, especially within the structural integrity discipline. Nevertheless, I am satisfied that the outcome of the ALARP assessments seem reasonable, therefore I do not consider that this is a significant shortfall within the case presented by the RP.

1136. From the information that I have sampled, I am broadly satisfied that the ALARP methodology and its subsequent application to SSCs of SI interest is adequate for the purposes of GDA. This does not necessarily mean that they will be adequate for use throughout the site-specific stages, especially for components of the highest reliability, which I consider to be a minor shortfall.

4.7.4 Conclusion

1137. The RP has considered the ALARP implications in terms of classification, component design, design code selection and materials selection. I note that the RP has made significant changes to the design and manufacture, as compared with the reference design, during the course of the GDA as a direct output from ALARP considerations. For example, this has included a change to the reference design in terms of materials and production route for the reactor coolant pump casing. I consider this to be a demonstration that the RP holds a positive attitude towards meeting ONR's expectations in terms of nuclear safety.

1138. Notwithstanding the reservations above regarding demonstration of gross-disproportionality, from the documents that I have sampled and my engagements with the RP, I am content the RP understands ONR expectations for reducing risks ALARP, through appropriate consideration of SI aspects.

4.8 Consolidated Safety Case

1139. I have performed a high-level review focused on the consolidation of changes identified and proposed for implementation during the GDA process for Chapter 17 of the PCSR. During this part of my assessment, I have not performed any detailed technical analysis but have simply looked for consistency between the support documents and the latest version of Chapter 17 of the PCSR, as presented at the time of assessment (Ref. 3). Adequacy of the RP's approach to SI safety case development, structure and content is discussed in more detail in Sub-section 4.1 above.

1140. Chapter 17 represents the top-level safety case describing the demonstration of structural integrity for the UK HPR1000. This provides a head document, drawing together the claims arguments and evidence at the top level of the structural integrity safety case. It acts as an overarching document to bring together or signpost claims, arguments, and evidence pertinent to the safety case.

1141. As part of the RP's process, modifications have been made to the PCSR to reflect updates and progress in the work done in subordinate documents. A list of 28 items have been identified by the RP in the 'Update Plan of PCSR Chapter 17 V2 and the Main Modification for Draft 1' (Ref. 268).

1142. I have performed a sample to check of this document to determine whether these planned amendments have been included in the latest version of the PCSR available at the time of writing. This review has been performed on Revision H of the PCSR (Ref. 3) and the sampled items I have selected are listed below in Table 12:

Table 12: List of modifications to the safety case identified by RP to demonstrate consolidation of the SI safety case within GDA.

Number	Section	Modifications
4	17.2.1 Objective 17.2.2 Scope 17.2.3 Chapter route map 17.2.4 Chapter structure	There is no modification identified within the technical contents, except some text refinement, after review of the latest plant route map and GDA scope.
6	17.2.6 Interface with other chapters	Modify the relevant description of partial chapter according to their feedback, to present the interfaces more clearly.
10	17.5.2 Result of structural integrity classification	Modify the candidate HIC list and the finalised HIC list according to the latest contents of Equipment Structural Integrity Classification List and the results of RO08 and RO58

Number	Section	Modifications
13	17.6.1.1.3 Material selection	1)Add the relevant description of material selection and ageing and degradation for MCL, RCP and MSL, as well as the corresponding references. 2) Add the relevant description ageing and degradation for RVI and main feedwater line (such as FAC), as well as the corresponding references.
17	17.6.1.2 Avoidance of fracture	1) Refine the relevant step description for implementing avoidance of fracture demonstration, according to the assessment comments and completed works of avoidance of fracture demonstration. 2) Modify the description of reconciliation of DTA and TJ according to the latest contents of Avoidance of Fracture Reconciliation Strategy and the response to RO-UKHPR1000-0006 and RQs. 3) Add the DTA/TJ/reconciliation works completed in Step 4 of GDA.
19	17.6.2 Structural integrity class 1 component	Add the sampled SIC-1 CSRs, and modify the relevant description according to GDA Step 4 works.

1143. From my sample review of the above items listed in Table 12, I am satisfied that appropriate changes have been made to (Ref. 3).

1144. In addition to the above sample, Sub-section 4.1 of my assessment identifies several instances where document updates and amendments to reflect emergent work have not been fully implemented in the safety case at the time of writing. it is important that the transition between GDA and site-specific stages is managed properly, for which a robust and unambiguous safety case is essential. I consider the lack of document updates completed within GDA to be a minor shortfall, which will be addressed as part of the safety case development during site specific stages.

1145. In addition to the above samples, throughout my assessment I have noted several instances where the RP has made a commitment to update documentation in response to RQ's raised by ONR. To ensure these have been consolidated into the safety case, I have sampled a number of these instances, for which my findings have been recorded below in Table 13:

Table 13: List of changes sampled to demonstrate consolidation of the SI safety case within GDA.

Commitment	Document	Status
In RQ-UKHPR1000-1247 (Ref. 28), the RP committed to revising defect tolerance assessments to improve traceability and ensure that the reason for conducting the DTAs could be tracked back to the SI safety claims. This commitment was made after several DTA's had already been produced for resolution of RO-UKHPR1000-0006 (Ref. 49). I therefore selected these to check whether the changes had been applied retrospectively via a revision update.	GHX00100010DPLX44GN - Defect Tolerance Assessment of Flange-nozzle Shell to Core Shell Weld - Rev F - 1 September 2020 GHX00100007DPLX44GN - Defect Tolerance Assessment of RPV Inlet Nozzle to Safe End Weld - Rev A - 27 February 2020 GHX00100015DPLX44GN - Defect Tolerance Assessment of RPV Core Shell to Transition Ring Weld - Rev A - 28 June 2019 GHX44100005DNLX44GN - Defect Tolerance Assessment of Main Steam Isolation Valve Casing crotch - Rev B - 31 January 2021 GHX00100034DPLX44GN - Avoidance of Fracture Reconciliation for RPV Inlet Nozzle to Flange-Nozzle Shell Weld - Rev C - 4 June 2021	None of the committed changes have been made to these documents, so the documents sampled remain non-compliant with the requirements of RO-UKHPR1000-0006 (Ref. 49) and RQ-UKHPR1000-1247 (Ref. 28). Overall, this is only a small number of the total number of documents produced by the RP and does not significantly undermine the technical evidence presented in the safety case.
In RQ-UKHPR1000-1406 (Ref. 28), the RP committed to revise a key document to address an action raised under RO-UKHPR1000-0033 (Ref. 49). I therefore selected this to check whether the changes had been applied as per the RP's commitment in the RQ.	GHX44300011DPZS03GN - QA Requirements for SG Design, Manufacture and Plant Construction - Rev C - 8 April 2021	The document has been updated in the latest version.
In RQ-UKHPR1000-1420 (Ref. 28), the RP committed to revise a key document to address an action raised under RO-UKHPR1000-0033 (Ref. 49). I therefore selected this to check whether the changes had been applied as per the RP's commitment in the RQ.	GHX44300013DPZS03GN - The SG Codes Relevant Risks Analysis and Assessment report - Rev B - 9 April 2021	The document has been updated in the latest version.

Commitment	Document	Status
In RQ-UKHPR1000-1620 (Ref. 28), the RP committed to update its methodology documentation to improve understanding of how different safety classifications (safety/mechanical/code /SI) are linked.	GHX30000002DOZJ03GN - Methods and Requirements of Structural Integrity Classification	The document has been updated in the latest version.
In RQ-UKHPR1000-1406 (Ref. 28), the RP committed to a number of document changes and additions to address ONR queries and ensure that this is recorded in the safety case to align with ONR expectations.	QA Requirements for SG Design, Manufacture and Plant Construction, GHX44300011DPZS03GN,	The document has been updated in the latest version.

4.8.1 Strengths

1146. The RP has developed and implemented a process to ensure that key amendments to the technical evidence for the SI safety case are appropriately reflected through revision of the higher tier documents.

1147. The consolidated safety case for structural integrity as presented within GDA has been managed and updated to include references to key judgements, safety significant changes and design modifications applied to the UK HPR1000 as result of regulatory engagements and RP self-assessment.

4.8.2 Outcomes

1148. Whilst every effort has been taken to ensure that key safety significant claims have been updated to reflect changes in the supporting evidence, there are still instances where the documentation has not been updated within the GDA. This has resulted in a lack of traceability of evidence within the safety case and ambiguity when trying to reconcile evidence supporting some safety claims. It should be noted that in my opinion, these instances are generally minor in nature and appear to be more of an administrational shortfall in the completion of due process, rather than through lack of fundamental evidence being available.

4.8.3 Conclusion

1149. I am generally satisfied that the RP has fulfilled its commitment to issue a consolidated SI safety case within the GDA, to demonstrate how risks associated with metal SSCs of the UK HPR1000 have been reduced ALARP. A number of instances have been identified where traceability of technical evidence in the safety case has been linked to absent or outdated documents, however these are not considered to be significant within the scope of GDA and are expected to be resolved during the site-specific stages.

4.9 Comparison with Standards, Guidance and Relevant Good Practice

1150. The standards, guidance and relevant good practice used in my assessment are referenced in context throughout Section 4, and are listed in Sub-section 2.4 and in Annex 1. A summary of my judgement against the most relevant of these is as follows:

- Relevant ONR SAPs from EMC series that relate to the structural integrity aspects of safety cases. I am satisfied that the RP appropriately considers these expectations during GDA, in particular with regard to EMC.1-3 on demonstration of highest reliability.
- Relevant ONR SAPs from SC series that relate to the production of an adequate safety case. I am content that the RP has met the intent of these assessment principles to the extent expected for GDA. I have also gained confidence that appropriate consolidation of the structural integrity safety case has been achieved, as noted in Sub-section 4.8. There are a number of areas where further work will be required by the licensee, but the generic case presented represents a suitable basis for this future development and broadly meets my expectations for GDA.
- Relevant ONR SAPs from EAD series that relate to material ageing and degradation and the provision of through life measurement of materials properties. As described throughout this report, there is a key relationship between materials and the demonstration of through life integrity.
- Relevant ONR SAPs from the EHA series have been considered with respect to the classification of components. Where referenced in the body of the report, I have sought advice from Internals Hazards specialists to inform my judgements on the adequacy of consequence analysis. From my engagement with the ONR IH specialists, I am broadly satisfied that the RP's safety case has met the expectations laid down with the EHA series of ONR SAPs.

1151. In summary, from the information that I have sampled and assessed, I have no reason to consider that the SI safety case as presented within GDA will not meet the intent of the codes and standards listed above, and is in line with my expectations for what constitutes UK relevant good practice from a structural integrity perspective.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

1152. This report presents the findings of my Structural Integrity assessment of the generic UK HPR1000 design as part of the GDA process.

1153. Based on my assessment, undertaken on a sampling basis, I have concluded the following:

- In my opinion, the RP has developed an adequate safety case methodology and structure for the UK HPR1000, which demonstrates how the risks associated with structural integrity of the plant are identified, assessed and managed.
- I am satisfied that the RP has demonstrated an adequate approach for classification of the UK HPR1000 Systems, Structures and Components (SSCs) important for safety. This approach recognises that the level of SI demonstration should be commensurate with the importance of the SSC to maintaining nuclear safety, enabling the RP to identify instances for which highest reliability claims need to be demonstrated, where unavoidable.
- I am broadly content that for the purposes of the GDA, the RP has provided a sound basis for confidence that adequate avoidance of fracture demonstrations can be made for the most safety significant components of the UK HPR1000. In my opinion, the RP has generally applied conservative methods in its Defect Tolerance Assessments and appropriate methods in the development of its GDA technical justifications, which provides confidence in the future qualification of manufacturing inspections.
- The RP has demonstrated an understanding of relevant design and construction code requirements, with sufficient evidence to demonstrate code compliance based on conservative assumptions. The RP has also provided an adequate explanation of its process for identifying relevant risks arising from supplementing the UK HPR1000 SG and RCP design and construction code (ASME) with additional measures from the RCC-M and RSE-M codes. Notably, the SI provision developed for the UK HPR1000 SG and RCP take cognisance of both Chinese practice and UK expectations relating to legislative requirements, and where appropriate, highest reliability. Whilst ONR does not encourage the combining of codes and standards for safety justifications, I am satisfied that the RP has provided sufficient reason to explain why this approach is justified for the UK HPR1000 and has broadly demonstrated how these codes are mutually compatible for the SG and RCP.
- The RP has developed and applied an adequate materials selection and testing strategy, which I consider provides sufficient evidence to underpin safety case claims of high-quality components and consideration of through life ageing and degradation.
- I am satisfied that the RP has presented sufficiently detailed proposals for NDE inspection that support the structural integrity claims in the PCSR and supporting documents. The RP has presented a tiered approach to NDE development and qualification that considers the structural integrity classification and the role of the NDE in supporting the integrity. I am satisfied that the RP has paid due attention to the ‘design for inspectability’ concept and has committed to several design modifications to improve NDE reliability.
- In general, the RP has completed a comprehensive ongoing review of the safety case, to ensure that improved safety justifications and plant modifications have been consolidated and incorporated into the most recent version of the PCSR. Whilst I have identified instances where committed changes are yet to be made, I do not consider these significant in the context of the overall safety demonstration, which can be resolved post-GDA.

- The RP's safety case considers the ALARP implications from a structural integrity perspective in terms of classification, component design, design code selection and materials selection. This has resulted in a number of component design changes to reduce risk. I have identified several opportunities for improvement in the demonstration of reducing risks to be ALARP. I am content that the RP understands ONR expectations for reducing risks to ALARP, through appropriate consideration of SI aspects and that further development of this aspect for safety can be managed post-GDA.
- In the course of my assessment of the above structural integrity topic areas, I have raised 29 assessment findings in total, which are provided in Annex 2.

1154. Overall, based on my sample assessment of the safety case for the generic UK HPR1000 design undertaken in accordance with ONR's procedures, I am satisfied that the case presented within the PCSR and supporting documentation is adequate. On this basis, I am content that a DAC should be granted for the generic UK HPR1000 design from a Structural Integrity perspective.

5.2 Recommendations

1155. Based upon my assessment detailed in this report, I recommend that:

- **Recommendation 1**: From a Structural Integrity perspective, ONR should grant a DAC for the generic UK HPR1000 design.

- **Recommendation 2**: The 29 Assessment Findings identified in this report should be resolved by the licensee for a site-specific application of the generic UK HPR1000 design.

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Annex 1

Relevant ONR Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
SC.4	The regulatory assessment of safety cases. Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose
EKP.3	Engineering principles: key principles. Defence in depth.	Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression
EMC.1	Integrity of metal components and structures: highest reliability components and structures. Safety case and assessment	The safety case should be especially robust and the corresponding assessment suitably demanding, in order that an engineering judgement can be made for two key requirements: a) the metal component or structure should be as defect-free as possible; b) the metal component or structure should be tolerant of defects.
EMC.2	Integrity of metal components and structures: highest reliability components and structures. Use of scientific and technical issues	The safety case and its assessment should include a comprehensive examination of relevant scientific and technical issues, taking account of precedent when available.
EMC.3	Integrity of metal components and structures: highest reliability components and structures: Evidence	Evidence should be provided to demonstrate that the necessary level of integrity has been achieved for the most demanding situations.
EMC.4	Integrity of metal components and structures: general. Procedural control	Design, manufacture and installation activities should be subject to procedural control.
EMC.5	Integrity of metal components and structures: general. Defects	It should be demonstrated that safety-related components and structures are both free from significant defects and are tolerant of defects.
EMC.6	Integrity of metal components and structures: general. Defects	During manufacture and throughout the operational life the existence of defects of concern should be able to be established by appropriate means.
EMC.7	Integrity of metal components and structures: design. Loadings	For safety-related components and structures, the schedule of design loadings (including combinations of loadings), together with conservative estimates of their frequency of occurrence should be used as the basis for design against normal operating, plant transient, testing, fault and internal or external hazard conditions.

SAP No	SAP Title	Description
EMC.8	Integrity of metal components and structures: design. Requirements for examination	Geometry and access arrangements should have regard to the requirements for examination.
EMC.9	Integrity of metal components and structures: design. Product form	The choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component.
EMC.10	Integrity of metal components and structures: design. Weld positions	The positioning of welds should have regard to high-stress locations and adverse environments.
EMC.11	Integrity of metal components and structures: design. Failure modes	Failure modes should be gradual and predictable.
EMC.12	Integrity of metal components and structures: design. Brittle behaviour	Designs in which components of a metal pressure boundary could exhibit brittle behaviour should be avoided.
EMC.13	Integrity of metal components and structures: manufacture and installation. Materials	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility.
EMC.17	Integrity of metal components and structures: manufacture and installation. Examination during manufacture	Provision should be made for examination during manufacture and installation to demonstrate the required standard of workmanship has been achieved.
EMC.21	Integrity of metal components and structures: operation. Safe operating envelope	Throughout their operating life, safety-related components and structures should be operated and controlled within defined limits consistent with the safe operating envelope defined in the safety case.
EMC.23	Integrity of metal components and structures: operation. Ductile behaviour	For metal pressure vessels and circuits, particularly ferritic steel items, the operating regime should ensure that they display ductile behaviour when significantly stressed.
EMC.24	Integrity of metal components and structures: monitoring. Operation	Facility operations should be monitored and recorded to demonstrate compliance with the operating limits and to allow review against the safe operating envelope defined in the safety case.

SAP No	SAP Title	Description
EMC.27	Integrity of metal components and structures: pre- and in-service examination and testing. Examination	Provision should be made for examination that is reliably capable of demonstrating that the component or structure is manufactured to the required standard and is fit for purpose at all times during service.
EMC.28	Integrity of metal components and structures: pre- and in-service examination and testing. Margins	An adequate margin should exist between the nature of defects of concern and the capability of the examination to detect and characterise a defect.
EMC.29	Integrity of metal components and structures: pre- and in-service examination and testing. Redundancy and diversity	Examination of components and structures should be sufficiently redundant and diverse.
EMC.30	Integrity of metal components and structures: pre- and in-service examination and testing. Control	Personnel, equipment and procedures should be qualified to an extent consistent with the overall safety case and the contribution of examination to the Structural Integrity aspect of the safety case.
EMC.32	Integrity of metal components and structures: analysis. Stress analysis	Stress analysis (including when displacements are the limiting parameter) should be carried out as necessary to support substantiation of the design and should demonstrate the component has an adequate life, taking into account time-dependent degradation processes.
EMC.33	Integrity of metal components and structures: analysis. Use of data	The data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and the contribution to the safety case.
EMC.34	Integrity of metal components and structures: analysis. Defect sizes	Where high reliability is required for components and structures and where otherwise appropriate, the sizes of crack-like defects of structural concern should be calculated using verified and validated fracture mechanics methods with verified application.
EAD.1	Ageing and degradation. Safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD.2	Ageing and degradation. Lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components that are important to safety.

SAP No	SAP Title	Description
EAD.3	Ageing and degradation. Periodic measurement of material properties	Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties.
EAD.4	Ageing and degradation. Periodic measurement of parameters	Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.
ECS.1	Safety classification and standards. Safety categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety.
ECS.2	Safety classification and standards. Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regard to safety.
ECS.3	Safety classification and standards. Standards	Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.

Annex 2

Assessment Findings

Number	Assessment Finding	Section
AF-UKHPR1000-0006	The licensee shall, as part of detailed design, demonstrate that the derivation of design loads is compliant with the provisions of the relevant design codes and standards. This should include, but not be limited to, the basis for design against normal operation, fault and accident conditions.	Section 4.1.5.7. Para 117
AF-UKHPR1000-0116	The licensee shall, as part of detailed design, demonstrate the impact of geometric alignment on fulfilling safety functions for components in situations where misalignment cannot be discounted within the safety case.	Section 4.1.7.4. Para 148
AF-UKHPR1000-0186	The licensee shall, as part of detailed design, demonstrate that the evidence necessary to underpin the claims and arguments in the safety case is robust, traceable and supported by a clear narrative. This should include, but not be limited to, the provision of structural integrity classification to reflect its importance within the system design manuals and clear linkage within component safety reports to the consequence analysis undertaken for structural integrity classified components.	Section 4.1.9. Para 154
AF-UKHPR1000-0187	The licensee shall, as part of detailed design, demonstrate how additional provisions applied for Structural Integrity Class-1 components for the UK HPR1000 reference design are addressed within the UK HPR1000 safety case. This should include, but is not limited to, justification of the associated criteria and additional provisions for the reactor vessel internals.	Section 4.2.1.1, Para. 170
AF-UKHPR1000-0188	The licensee shall, as part of detailed design, demonstrate that the structural integrity classification process identifies and confirms the limiting faults during all plant states.	Section 4.2.1.1. Para 172
AF-UKHPR1000-0189	The licensee shall, as part of detailed design, compare the design code (pressure equipment) class assigned via the M class with relevant operational experience. For structural integrity, if this comparison results in the assignment of a different design code (pressure equipment) class for the component, the licensee shall either upgrade the nuclear pressure equipment design class or justify the use of a lower design class.	Section 4.2.1.2. Para 198

Number	Assessment Finding	Section
AF-UKHPR1000-0190	The licensee shall, as part of implementing the ESPN Order during detailed design, review any resultant changes in the classification of structures, systems and components and either upgrade or justify the allocated design code (pressure equipment) class.	Section 4.2.1.2. Para 208
AF-UKHPR1000-0191	The licensee shall, as part of detailed design, demonstrate that closure components are appropriately classified and underpinned by the safety case. This should include, but not be limited to, the consequence of failure of covers, studs and bolts for high integrity components.	Section 4.2.1.3. Para 322
AF-UKHPR1000-0192	The licensee shall, as part of detailed design, justify that consistent approaches to lifetime fatigue crack growth are employed in its defect tolerance assessment methodologies, such that judgements can be made on the level of defect tolerance and the development of the avoidance of fracture demonstration cases.	Section 4.3.7.4. Para 425
AF-UKHPR1000-0193	The licensee shall, as part of detailed design, undertake defect tolerance assessment covering all relevant weld locations and defect orientation for high integrity components, including the vulnerable areas of the parent forging. This should expand the scope of assessments undertaken during GDA and should demonstrate that the limiting locations have been assessed.	Section 4.3.7.5. Para 427
AF-UKHPR1000-0194	The licensee shall, as part of detailed design, demonstrate that consistent approaches are implemented in the defect tolerance assessment, in accordance with the relevant defect tolerance assessment procedure. This should include, but not be limited to, consistent use of weld residual stress values, appropriate yield stress in failure parameters and identification of the limiting conditions for plastic collapse.	Section 4.3.7.5. Para 435
AF-UKHPR1000-0195	The licensee shall, as part of detailed design, develop and implement a robust verification process to underpin the defect tolerance assessment work in support of the avoidance of fracture demonstrations for high integrity components.	Section 4.3.7.8. Para 471
AF-UKHPR1000-0196	The licensee shall, as part of detailed design, justify the assumed manufacturing defect descriptions in inspection specifications for high integrity components. This should include, but not be limited to, the use of suitable expert elicitation.	Section 4.3.10.1. Para 491

Number	Assessment Finding	Section
AF-UKHPR1000-0197	The licensee shall, as part of detailed design, justify the position for any high integrity component welds with low defect size margins identified during GDA to confirm the end of life limiting defect size, and underpin the avoidance of fracture demonstrations. This should include, but not be limited to, the main steam isolation valve weld repair.	Section 4.3.13.1. Para 639
AF-UKHPR1000-0198	The licensee shall, as part of detailed design, justify within the safety case where Design by Analysis is being claimed as evidence of code compliance in lieu of Design to Rule assessments.	Section 4.4.1.1. Para 720
AF-UKHPR1000-0199	The licensee shall, as part of detailed design, justify that suitable tri-axial stress and thermal stress ratchet analyses are used to underpin the safety case. Where these analyses are not considered necessary, this should be justified.	Section 4.4.1.1. Para 720
AF-UKHPR1000-0200	The licensee shall, as part of detailed design, demonstrate that the rules of ASME III Appendix A are applied for tube sheets.	Section 4.4.1.1. Para 720
AF-UKHPR1000-0201	The licensee shall, as part of detailed design, document within the safety case the key assumptions and uncertainties applied in the finite element modelling methodologies, including where axisymmetric modelling of asymmetric nozzles is undertaken.	Section 4.4.1.1. Para 720
AF-UKHPR1000-0202	The licensee shall, as part of detailed design, develop and implement a process for the evaluation of small margins against design code limits for high integrity components and in situations where fatigue usage factors are high. This should include, but not be limited to, the reactor pressure vessel components, such as the closure head flange, closure studs, core shell, inlet and outlet nozzles, and the main coolant loop charging line weld.	Section 4.4.1.2. Para 734
AF-UKHPR1000-0203	The licensee shall, as part of detailed design, justify and implement approaches to account for the environmental effects of primary circuit coolant water, in its evaluation of the design life of structures, systems and components for the UK HPR1000. This should include, but not be limited to, the safety significant vessels and piping designed to the RCC-M code and, where combinations of codes are justified such as the steam generator and reactor coolant pump, to ASME III.	Section 4.4.1.2. Para 738

Number	Assessment Finding	Section
AF-UKHPR1000-0204	The licensee shall, as part of detailed design, justify the application of fatigue initiation analysis to RCC-M Annex ZD to components, and demonstrate the validity and veracity of the test data that underpin the specific fatigue initiation (S-N) curves.	Section 4.4.1.2. Para 742
AF-UKHPR1000-0205	The licensee shall, as part of detailed design, demonstrate that all reasonably practicable measures have been implemented to minimise risks to the integrity of locations with cumulative fatigue usage factors of ≥ 0.75. This should include, but not be limited to, operational experience, the uncertainties and conservatism in the analyses, where component repair or replacement option is not available and where additional measures to afford forewarning of failure could be implemented.	Section 4.4.1.2. Para 750
AF-UKHPR1000-0206	The licensee shall, as part of the detailed design and where the design approach of combining design codes is justified, demonstrate that all relevant design and construction code developments are implemented, taking cognisance of the need to reduce risks to as low as reasonably practicable.	Section 4.4.1.3. Para 760
AF-UKHPR1000-0207	The licensee shall, as part of detailed design, demonstrate the inspections and testing of interface regions for dissimilar metals welds reduce risks to as low as reasonably practicable, where the approach of combining codes and standards is justified.	Section 4.4.2.2. Para 819
AF-UKHPR1000-0208	The licensee shall, as part of detailed design, identify any additional non-destructive examination requirements beyond the design code, imposed by the designer for structural integrity class 1 to 3 components, post the workshop hydrotest in the UK HPR1000 reference design. The licensee shall also establish whether it is commensurate with reducing risks to as low as reasonably practicable to implement any additional non-destructive examination for the UK HPR1000.	Section 4.4.3.1, Para 860
AF-UKHPR1000-0209	The licensee shall, as part of detailed design, justify whether it is reasonably practicable to use ultrasonic inspection for: ▪ austenitic stainless steel and Inconel alloy non-high integrity component welds; and ▪ those high integrity component welds for which the in-process inspection does not include ultrasonic inspection.	Section 4.6.1. Para 1037

Number	Assessment Finding	Section
AF-UKHPR1000-0210	The licensee shall, as part of detailed design, demonstrate that the in-process manufacturing non-destructive examination is capable of detecting the defects of concern. This should include both high integrity component parent material and non-high integrity component welds.	Section 4.6.1. Para 1040
AF-UKHPR1000-0211	The licensee shall, as part of detailed design, justify the application of ultrasonic inspection to replace radiography in the pre-service/in-service inspections for high integrity and non-high integrity components and austenitic and Inconel alloy welds, including austenitic and Inconel alloys and dissimilar metals at code interfaces.	Section 4.6.2. Para 1046
AF-UKHPR1000-0212	The licensee shall, as part of detailed design, demonstrate that the pre-service/in-service non-destructive examination for non-high integrity component welds is capable of detecting the defects of concern.	Section 4.6.2. Para 1048