Office for Nuclear Regulation

Civil Nuclear Reactor Build - Generic Design Assessment

Step 2 Assessment of the Structural Integrity of Hitachi GE's UK Advanced Boiling Water Reactor (UK ABWR)

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EXECUTIVE SUMMARY

This report presents the results of my assessment of the Structural Integrity of the Hitachi General Electric Nuclear Energy Ltd (Hitachi-GE) UK Advanced Boiling Water Reactor (UK ABWR) undertaken as part of Step 2 of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA).

The GDA process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments getting increasingly detailed as the project progresses. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety or security shortfalls that could prevent the proposed design from being licensed in Great Britain. Therefore during GDA Step 2 my work has focused on the assessment of the key claims in the area of Structural Integrity to judge whether they are complete and reasonable in the light of our current understanding of reactor technology.

Structural Integrity, within the context of this GDA, is primarily concerned with the integrity of metal components and structures, for example pressure vessels and piping, their supports and vessel internals. In the most general sense the, Structural Integrity safety claim is based on the identification of the integrity level claimed for a component or structure in order to support the overall safety case for the reactor.

An important aspect of the Structural Integrity safety claim is the identification of those components which form a principal means of ensuring nuclear safety and where the safety case needs to claim that the likelihood of gross failure is so low that the consequences of gross failure can be discounted from the deterministic safety analysis i.e. the identification of those components needing a highest reliability claim. These components require an in-depth explanation of the measures over and above normal practice that support and justify the claim that the likelihood of gross failure is so low that it can be discounted.

I have, therefore, sought to confirm that the UK ABWR safety case is based on identifying the integrity claims necessary to support the overall safety case and that those components, which will need a claim that the likelihood of gross failure is so low that it can be discounted, will be identified and that a suitable approach will be developed to justify such claims.

I have also considered the through-life degradation mechanisms that could, potentially, affect a UK ABWR as an implicit structural integrity claim will be related to the 60 year design life of the plant.

The principal standards I have used to judge the adequacy of the design fundamentals and claims in the area of Structural Integrity have been ONR's Safety Assessment Principles (SAPs), in particular SAPs EMC.1 to EMC.34 on the Integrity of Metal Components and Structures (EMC.1 to EMC.3 having specific relevance to the highest reliability claim); EAD.1 to EAD.4 on Ageing and Degradation; ECS.1 to ECS.3 on Safety Classification and Standards; and ONR's Technical Assessment Guide NS-TAST-GD-016 on the Integrity of Metal Components and Structures.

My GDA Step 2 assessment work has involved continuous engagement with the RP in the form of technical exchange workshops and progress meetings. In addition, my understanding of the ABWR technology and, therefore, my assessment, has benefited from a visit to the Hitachi Reactor Internal Pump test facility and the Hitachi Rinkai Works.

My assessment has been based on the RPs Preliminary Safety Report (PSR) and its references relevant to Structural Integrity. The RPs preliminary safety case related to Structural Integrity, as presented in those documents, can be summarised as providing:

- the basis for the Structural Integrity Classification process including the identification of the components needing a highest reliability claim;
- the approach to providing a beyond design code compliance justification to support a highest reliability claim;
- the basis for an avoidance of fracture justification bringing together material properties, fracture analysis and qualified manufacturing inspections;
- an overview on the approach to mitigating the threat from the stress corrosion cracking degradation mechanism; and
- design summaries for the main components in the reactor circuit.

During my GDA Step 2 assessment of the UK ABWR aspects of the safety case related to Structural Integrity I have identified the following areas of strength:

- the RP has adopted an approach to Structural Integrity classification that identifies the integrity claims needed to support the overall safety case;
- the RP has adopted an approach to systematically identifying those components requiring a claim that the likelihood of gross failure is so low that it can be discounted;
- the beyond design code compliance justification proposed by the RP using an avoidance of fracture demonstration for the highest reliability components appears consistent with ONR's expectations;
- a multi-faceted approach is being taken to mitigate the threat from the stress corrosion cracking degradation mechanism; and
- the design summaries show that the main components of the reactor are generally of a conventional nature which gives confidence that their integrity claims will be justifiable.

I have not identified any important shortcomings during my GDA Step 2 assessment of the UK ABWR, but I have raised four Regulatory Observations to aid the RP in meeting regulatory expectations during Step 3 and Step 4 of GDA:

- Avoidance of Fracture – Margins based on the size of Crack-Like Defects
- CRD Penetration Design
- RPV Design (use of forgings and plate materials)
- Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary

My Step 2 assessment recognises and accepts that the RP's safety case needs to be developed in many areas in order to provide the evidence to support the claims related to structural integrity, however, the areas below were specifically noted for follow-up:

- Sufficiency of low integrity claims for the Balance of Plant safety case ie the reactor circuit downstream of the Main Steam Isolation Valves
- Provision of a material selection justification taking into account UK ABWR specific water chemistry
- Optimised material choice for the Reactor Water Clean Up System
- Inclusion of the potential for chloride ingress, including protection measures and consequences, in the safety case

In relation to my interactions with Hitachi GE's Subject Matter Experts (SME) in Structural Integrity, I have found the RP to be receptive to ONR's approach and accepting of the need to provide beyond design code compliance justifications for the highest reliability components in line with ONR's expectations. The RP appears to have been well resourced, has consistently delivered good quality documentation to the agreed programme, and has made good use of UK contractors to provide specialist advice.

Overall, I see no reason, on Structural Integrity grounds, why the UK ABWR should not proceed to Step 3 of the GDA process.

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineers
ASME III	ASME Boiler and Pressure Vessel Code Section III
ASME VIII	ASME Boiler and Pressure Vessel Code Section VIII
BMS	Business Management System
BWR	Boiling Water Reactor (general sense)
CRD	Control Rod Drive
DAC	Design Acceptance Confirmation
EA	Environment Agency
Hitachi-GE	Hitachi General Electric Nuclear Energy Ltd
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
JSME	Japanese Society of Mechanical Engineers
JSW	The Japan Steel Works company, Japan
MSIV	Main Steam Isolation Valve
NDE	Non Destructive Examination
ONR	Office for Nuclear Regulation
PCSR	Pre-construction Safety Report
PSR	Preliminary Safety Report
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP(s)	Safety Assessment Principle(s)
SCC	Stress Corrosion Cracking

LIST OF ABBREVIATIONS

SSC	Structures, Systems and Components
SME	Subject Matter Expert
TAG(s)	Technical Assessment Guide(s)
TAGSI	UK Technical Advisory Group on the Structural Integrity of High Integrity Plant
TSC	Technical Support Contractor
WENRA	Western European Nuclear Regulators' Association

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1 INTRODUCTION

1.1 Background

1. The Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA) process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments getting increasingly detailed as the project progresses. Hitachi General Electric Nuclear Energy Ltd (Hitachi-GE) is the RP for the GDA of the UK Advanced Boiling Water Reactor (UK ABWR).

2. During Step 1 of GDA, which is the preparatory part of the design assessment process, the RP established its project management and technical teams and made arrangements for the GDA of its ABWR design. During Step 1 Hitachi-GE also prepared submissions to be evaluated by ONR and the Environment Agency (EA) during Step 2.

3. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety or security shortfalls that could prevent the proposed design from being licensed in Great Britain.

4. This report presents the results of my assessment of the Structural Integrity of Hitachi-GE's UK ABWR as presented in the UK ABWR Preliminary Safety Report (PSR) (Ref. 9) and supporting documentation (Refs 10 to 21).

1.2 Methodology

5. My assessment has been undertaken in accordance with the requirements of the Office for Nuclear Regulation (ONR) How2 Business Management System (BMS) procedure PI/FWD (Ref. 1). The ONR Safety Assessment Principles (SAPs) (Ref. 2), together with supporting Technical Assessment Guides (TAG) (Ref. 3) have been used as the basis for this assessment.

6. My assessment has followed my GDA Step 2 Assessment Plan for Structural Integrity (Ref 6) prepared in December 2013 and shared with Hitachi-GE to maximise openness and transparency.

2 ASSESSMENT STRATEGY

7. This section presents my strategy for the GDA Step 2 assessment of the Structural Integrity of the UK ABWR (Ref 6). It also includes the scope of the assessment and the standards and criteria that I have applied.

2.1 Scope of the Step 2 Structural Integrity Assessment

8. The objective of my GDA Step 2 Structural Integrity assessment for the UK ABWR was to review and judge whether the claims made by the RP related to Structural Integrity that underpin the safety, security and environmental aspects of the ABWR are complete and reasonable in the light of our current understanding of reactor technology.

9. In the most general sense, the Structural Integrity safety claim is based on indentifying the integrity levels necessary to support the overall safety case for the ABWR. This results in the identification of integrity claims on individual components and structures.

10. A fundamental aspect of the Structural Integrity safety claim is the identification of those components which form a principal means of ensuring nuclear safety and the likelihood of gross failure is claimed to be so low that the consequences of gross failure can be discounted, ie the highest reliability components. These components require an in-depth explanation of the measures over and above normal practice that support and justify the claim that the likelihood of gross failure is so low that it can be discounted. A typical example would be the main pressure boundary of the reactor pressure vessel.

11. The Step 2 Structural Integrity assessment has therefore sought to confirm that the UK ABWR safety case is based on identifying structural integrity claims necessary to support the overall safety case and that those components which will need a claim that the likelihood of gross failure is so low that it can be discounted will be identified and that a suitable approach will be developed to justify such claims.

12. The Step 2 Structural Integrity assessment has also considered the through-life degradation mechanisms that could potentially affect an ABWR as an implicit claim will be related to the 60 year design life of the plant.

13. During Step 2, I have also evaluated whether the claims related to Structural Integrity are to be supported by a body of detailed technical documentation sufficient to allow me to proceed with GDA work beyond Step 2.

14. Finally, during Step 2 I have undertaken the following preparatory work for my Step 3 assessment:

- preparation of longer term Regulatory Observations to aid the RP in meeting regulatory expectations during Step 3 and Step 4;
- review of the level of the technical support contracts needed during Step 3 and Step 4; and
- review of boiling water reactor operating experience (including ABWR experience) in terms of the types of material degradation mechanisms seen through life.

2.2 Standards and Criteria

15. The goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety, security and environmental case. For this purpose, within ONR, assessment is undertaken in line with the requirements of the How2 Business Management System (BMS) document PI/FWD (Ref. 1). Appendix 1 of Ref. 1 sets down the process of assessment within ONR; Appendix 2 explains the process associated with sampling of safety case documentation.

16. In addition, the Safety Assessment Principles (SAPs) (Ref. 2) constitute the regulatory principles against which duty holders' safety cases are judged, and, therefore, they are the basis for ONR's nuclear safety assessment and therefore have been used for GDA Step 2 assessment of the UK ABWR. The SAPs 2006 Edition (Revision 1 January 2008) were benchmarked against the IAEA standards (as they existed in 2004). They are currently being reviewed.

17. Furthermore, ONR is a member of the Western Regulators Nuclear Association (WENRA). WENRA has developed Reference Levels, which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors.

18. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and enlarged on in the Technical Assessment Guide on Structural Integrity (Ref. 3, and see

2.2.2 below). This guide provides the principal means for assessing the Structural Integrity aspects in practice.

2.2.1 Safety Assessment Principles

19. The SAPs (Ref. 2) of relevance to this assessment are SAPs EMC.1 to EMC.34 on the Integrity of Metal Components and Structures (EMC.1 to EMC.3 having specific relevance to the highest reliability claim); EAD.1 to EAD.4 on Ageing and Degradation; ECS.1 to ECS.3 on Safety Classification (see also Table 1 for further details).

2.2.2 Technical Assessment Guides

20. The following Technical Assessment Guide has been used as part of this assessment (Ref. 3):

- NS-TST-GD-016 Revision 4. March 2013. Integrity of Metal Components and Structures.

2.3 Use of Technical Support Contractors

21. I have not engaged Technical Support Contractors (TSC) to support my assessment of Structural Integrity for the UK ABWR during Step 2.

2.4 Integration with Other Assessment Topics

22. Early in GDA I recognised that during the project there would be a need to consult with other assessors (including Environment Agency's assessors) as part of the Structural Integrity assessment process. Similarly, other assessors will seek input from my assessment of the Structural Integrity for the UK ABWR. I consider these interactions very important to ensure the prevention of assessment gaps and duplications, and, therefore, are key to the success of the project. Thus, from the start of the project I made every effort to identify as many potential interactions as possible between the Structural Integrity and other technical areas, with the understanding that this position would evolve throughout the UK ABWR GDA.

23. It should be noted that the interactions between the Structural Integrity and some technical areas may need to be formalised since aspects of the assessment in those areas constitute formal inputs to the Structural Integrity assessment, and vice versa. At this stage, however, interactions have been on an informal basis covering the following:

- Structural Integrity inspectors provide input to the missile generation, pipe-whip and internal flooding aspects of the Internal Hazards assessment. During Step 2 there have been significant informal interactions on the failure modes to be assumed for nuclear classified medium energy piping and on the failure locations for pipe-whip assessment. This informal interaction has included joint meetings with the RP and internal meetings to establish the ONR position on these aspects.

- The Structural Integrity inspectors work with the Reactor Chemistry inspectors in assessing the potential for through-life degradation. During Step 2, there have been a number of informal interactions to ensure that a consistent and integrated approach is been taken both by the Regulators and the RP. This has involved joint meetings with the RP, internal meetings on the topic and joint inputs on related Regulatory Queries (RQs).

- The Fault Studies inspectors provide advice on the structural integrity claims needed to support the overall safety case for the plant. During Step 2 there have been a number of informal interactions to ensure that the structural integrity classification process is consistent with the overall classification approach being taken for the plant.

- The Structural Integrity inspectors provide input on the metallic components used in the containment structure. The overall assessment of the containment structure is lead by the Civil Engineering inspectors. During Step 2 there have been informal interactions on the design code to be used for these metallic structures.

24. In addition to the above, during GDA Step 2 there have been interactions between Structural Integrity and the rest of the technical areas. Although these interactions, which are expected to continue thorough GDA, are mostly of an informal nature, they are essential to ensure consistency across the technical assessment areas.

3 REQUESTING PARTY'S SAFETY CASE

25. This section presents a summary of the RP's preliminary safety case in the area of Structural Integrity. It also identifies the documents submitted by Hitachi-GE which have formed the basis of my assessment of the UK ABWR Structural Integrity during GDA Step 2.

3.1 Summary of the RP's Preliminary Safety Case in the Area of Structural Integrity

26. The aspects covered by the UK ABWR preliminary safety case in the area of Structural Integrity can be broadly grouped under 8 headings which can be summarised as follows:

- Overall Approach

The Structural Integrity PSR (Ref. 9) accepts that it will be necessary to provide a safety case to show how the structural integrity of the components can be assured over the design life of the plant to a level of structural reliability and degree of rigour commensurate with the consequences of gross failure.

It therefore proposes a structural integrity specific classification methodology based on the direct and indirect consequences of failure so that structured arguments can be presented for the major components tailored to the reliability claimed for that component.

It then acknowledges that for components where the consequences of gross failure are not acceptable, and that physical safeguards and barriers cannot be provided, then it needs to be shown that the likelihood of gross failure needs to be so low that it can be discounted. These components will then require a highest reliability claim. The arguments and evidence to support such a claim will require a high burden of proof and will require measures to be taken over and above nuclear design code requirements.

- Structural Integrity Classification

The structural integrity classification process starts with the overall classification system, and then identifies a sub-set of components within the Class 1 safety components and structures which require a higher reliability claim than can be demonstrated by code compliance. These components are identified as either Very High Integrity (VHI) components or High Integrity (HI) components in the RP's safety case depending the consequence of gross failure. For VHI there is no protection from gross failure, for HI there would be some protection.

The approach to making these decisions will be based on a failure modes and effects criticality assessment using an expert panel approach with relevant subject areas taking part.

- Safety Case Strategy

For components requiring a higher reliability claim (VHI or HI), a four legged safety case will be developed based on the ideas expressed in Ref 23 by the UK Technical Advisory Group on the Structural Integrity of High Integrity Plant (TAGSI).

For standard Class 1, Class 2 and Class 3 components the safety case will claim that design and manufacture to recognised nuclear and non-nuclear design codes will provide the evidence to support the reliability claims necessary.

- Avoidance of Fracture

The Structural Integrity PSR (Ref. 9) recognises the important contribution to any claim of high structural reliability made by a demonstration that the Non-Destructive Examination (NDE) techniques applied during manufacture can reliably detect defects below a size that they are considered structurally significant.

This will involve undertaking a detailed fracture mechanics based defect tolerance assessment to determine the limiting defect sizes on the VHI and HI components at the start of life taking account of any potential to grow the defects through life. The qualified non-destructive examinations being proposed for the components can then be shown to be able to reliably detect such postulated start of life defects with a suitable margin.

The documentation outlines the approach to the defect tolerance assessment, the approach to demonstrating the capabilities of the NDE techniques to be applied and the process which will be adopted to formally qualify the end of manufacture NDE.

This work is recognised as being a significant part of the beyond nuclear design code compliance needed to demonstrate the highest reliability claim as part of the four legged safety case strategy.

- Applicable Code and Standards

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III will be used for the for the Class 1 and Class 2 components and the ASME Boiler and Pressure Vessel Code Section VIII for Class 3 components, supplemented by other recognised international standards as appropriate.

- Material Choices and Degradation Mechanisms

The Structural Integrity PSR (Ref. 9) provides a brief summary of the materials selected for the UK ABWR design and the degradation threats. Stress Corrosion Cracking (SCC) and Irradiation Assisted SCC (IASCC) are two important threats and Ref. 15 describes the background and countermeasures taken to mitigate these specific threats. For example the external stainless steel reactor coolant recirculation loops seen in earlier Boiling Water Reactor (BWR) designs which are potentially susceptible to Stress Corrosion Cracking do not exist on the ABWR design.

- Design Summaries for Major Components

Design summaries, from a structural integrity perspective have been provided for the RPV, Main Steam Piping, Feedwater Piping and Main Steam isolation Valves (MSIVs), Refs. 10, 11, 12, and 13.

These provide an overview of the main design features, the functional requirements, the design requirements and diagrams of the main features of these components. The ASME Boiler and Pressure Vessel Code Section III will be used as the starting point for the design and manufacture of these components, supplemented as necessary by additional measures where components require a higher reliability claim.

They provide an important reference point on the design in advance of the PCSR.

- Specific characteristics of the ABWR Balance of Plant (BOP)

A design summary of the BOP from a structural integrity perspective is provided in Ref. 21. It describes the systems, the main components and gives advance information on the probable classification of the main components.

In general non-nuclear codes and standards are proposed for the BoP as the majority of the system as the majority of the component in the system are Class 3. No higher integrity claims such as HI or VHI are, currently, thought necessary for the BoP.

3.2 Basis of Assessment: RP's Documentation

27. The RP's documentation that has formed the basis for my GDA Step 2 assessment of the safety claims related to the Structural Integrity for the UK ABWR is:

- UK ABWR PSR Chapter on Structural Integrity “Preliminary Safety Report on Structural Integrity” (Ref. 9). This document outlines the RP's overall strategy for structural integrity; their approach to identifying those components which need a highest reliability claim and their proposals for justifying a highest reliability claim.

- UK ABWR Report “Summary of the Design of the Reactor Pressure Vessel for UK ABWR” (Ref. 10). This document provides a design summary for the Reactor Pressure Vessel in relation to pressure boundary integrity.

- UK ABWR Report “Summary of the Design of Main Steam Piping for the UK ABWR” (Ref. 11). This document provides a design summary for the Main Steam Piping up the first Main Steam Isolation Valve outside of the Primary Containment Vessel.

- UK ABWR Report “Summary of the Design of Feedwater Piping for the UK ABWR” (Ref. 12). This document provides a design summary for the Feedwater Piping from the first Feedwater isolation valve outside of the Primary Containment Vessel through to the Reactor Pressure Vessel nozzles.

- UK ABWR Report “Summary of the Design of Main Steam Isolation Valves for UK ABWR” (Ref. 13). This document provides a design summary for the pressure boundary of the Main Steam Isolation Valves.

- UK ABWR Report “Outline of the PSI and ISI Plan for ABWR” (Ref. 14). This document provides a high level description of the general approach that will be used for the Pre-Service Inspection (PSI) and In-Service inspection (ISI) of the UK ABWR.

- UK ABWR Report “Approach for the Avoidance of SCC” (Ref. 15). This document summarises the RP's approach to avoiding Stress Corrosion Cracking (SCC) in austenitic stainless steels and nickel-base alloys.

- UK ABWR Report “Structural Integrity Classification Procedure” (Ref. 16). This document summarises the RP's methodology for structural integrity classification, and the process for identifying those components requiring a highest reliability claim.

- UK ABWR Report “Weld Ranking Procedure” (Ref. 17). This document describes the RP's methodology for identifying the limiting areas on the highest reliability components for detailed assessment during GDA.

- UK ABWR Report "Defect Tolerance Assessment Plan" (Ref. 18). This document describes the RP's methodology for undertaking the fracture mechanics assessment to determine limiting defects sizes as part of the avoidance of fracture demonstration used in support of justifying a highest reliability claim.

- UK ABWR Report "Inspection Assessment Plan" (Ref. 19). This document describes the RP's methodology for demonstrating that the end of manufacture non-destructive examination proposals for the highest reliability components can reliably detect defects of structural significance with a suitable margin.

- UK ABWR Report "Inspection Qualification Strategy" (Ref. 20). This document describes the RP's methodology for qualifying the end of manufacture non-destructive examinations to be undertaken on the highest reliability components.

- UK ABWR Report "Summary of the Design of BOP Components for UK ABWR" (Ref. 21). This document provides a design summary of the main components in the Balance of Plant Components from a pressure boundary perspective.

- UK ABWR GDA tracking sheet (Ref. 8)

- Responses to Regulatory Queries (RQs) (Ref. 8):

  - RQ-ABWR-0134 "Chloride Ingress Protection"
  - RQ-ABWR-0163 "UK ABWR SCC and IASCC Claim"
  - RQ-ABWR-0164 'Structural Integrity Claims on the Balance of Plant"

- Resolution plans proposed by the RP to respond to Regulatory Observations (ROs) (Ref. 26):

  - RO-ABWR-0001 "Avoidance of Fracture – Margins based on the size of Crack-Like Defects"
  - RO-ABWR-0002 "CRD Penetration Design" (CRD – Control Rod Drive Mechanism)
  - RO-ABWR-0003 "RPV Design" (RPV – Reactor Pressure Vessel)
  - RO-ABWR-0004 "Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary".

28. The RP also submitted the following documentation during the later stages of Step 2. Although submitted during Step 2, these documents were not intended to form part of my Step 2 assessment and will be taken into account during the Step 3 assessment:

- UK ABWR Report "Structural Integrity Supporting Report - Load Combinations for Systems and Components" (Ref. 30). This report describes the basis for the load combinations which will be used in the structural integrity assessment of the UK ABWR.

- UK ABWR Report "Structural Integrity Supporting Report – Seismic Design for Systems and Components" (Ref. 31). This report describes the basis for the seismic analysis methods used for systems and components.

- UK ABWR Report "Structural Integrity Classification Report" (Ref. 32). This report describes the outputs from the application of the structural integrity classification procedure where those components requiring a highest reliability claim are identified.

- UK ABWR Report "Weld Ranking Application Report" (Ref. 33). This report describes the outputs from the application of the weld ranking procedure to identify the limiting areas on the highest reliability components.

- UK ABWR Report "Proposed Topic Report Structure for the components relating to Structural Integrity" (Ref. 34). This report provides an outline of the safety case structure that will be used for the structural integrity topic reports which support the Pre-Construction Safety Report (PCSR).

29. In addition, in May 2014 Hitachi-GE has submitted to ONR for information an advance copy of the UK ABWR PCSR. The advance copy described Structural Integrity in Chapter 5.5 (Ref. 22), but this chapter number may change in the formal issue of the PCSR. Although I have not covered this report in my GDA Step 2 formal assessment, it has been useful to start planning and preparing my GDA Step 3 work.

4 ONR ASSESSMENT

30. My assessment has been carried out in accordance with ONR How2 BMS document PI/FWD, “Purpose and Scope of Permissioning” (Ref. 1).

31. My GDA Step 2 Structural Integrity assessment has followed the strategy described in Section 2 of this report.

32. My Step 2 assessment work has involved continuous engagement with the RP’s Structural Integrity Subject Matter Experts (SME), ie, 3 Technical Exchange Workshops (2 in Japan and 1 the UK) and 4 progress meetings (by video conferences) have been held. I have also visited:

- Hitachi Works (reactor internal pump test facility), where I was shown the arrangement of the reactor internal pump in respect of the reactor pressure vessel pressure boundary.
- Hitachi Works (reactor internals workshop), where I was shown various components being manufactured for use inside the reactor pressure vessel and the arrangement of the Control Rod Drive (CRD) mechanism.
- I was also given a tour of the Muroran Plant of Japan Steel Works (JSW) by JSW staff. This JSW plant will manufacture the large ferritic forgings used in the production of the UK ABWR reactor pressure vessel. There were some nuclear reactor pressure vessel forgings undergoing manufacture at the time of the visit, but no ABWR forgings.

33. During my GDA Step 2 assessment I raised 3 RQs where I needed to formalise my requests for additional information. I also issued 4 ROs during my GDA Step 2 assessment to aid the RP in meeting regulatory expectations during Step 3 and Step 4 of GDA.

34. Details of my GDA Step 2 assessment of the UK ABWR preliminary safety case in the area of Structural Integrity including the areas of strength that I have identified, as well as the items that require follow-up and the conclusions reached are presented in the following sub-sections.

4.1 Overall Approach

4.1.1 Assessment

35. The assessment of the overall approach to structural integrity starts with a consideration of the structural integrity ‘safety claim’ in its most general sense, and whether the RP’s approach to structural integrity is based on identifying the integrity levels necessary to support the overall safety case. This is linked to the ONR SAPs (Ref. 2) ECS.2 and ECS.3 on the safety classification of structures, systems and components (SSCs) and that SSCs important to safety should be designed, manufactured, installed, maintained etc to appropriate standards.

36. In general the integrity levels for normal components and structures will be justified primarily through compliance with internationally accepted nuclear and non-nuclear design and construction codes covering components such as pressure vessels, pipework, supports, reactor internal structures, etc. These codes provide for a graded approach to link integrity levels to the overall safety case. This aspect is discussed further under Safety Case Strategy and Applicable Codes and Standards below.

Further consideration is, however, required in the case of components where the consequences of gross failure cannot be shown to be acceptable.

37. ONR's SAPs (Ref. 2) acknowledge that there will be some components where the consequences of gross failure cannot be shown to be acceptable and no further protection or measures can be put in place against such a failure. Under these circumstances the emphasis falls on the arguments and evidence to support the claim that the likelihood of gross failure is very low, and so unlikely that it can be discounted from the deterministic safety assessment (ONR SAPs paragraphs 238 to 257). Similar claims have featured in safety cases for operating nuclear power stations in Great Britain.

38. The SAPs (paragraph 243) note that this is an onerous route to constructing a safety case, and there will need to be an in depth explanation of the measures over and above normal practice that support and justify the highest reliability claims.

39. Thus the identification and justification of these highest reliability components is a fundamental aspect of considering the 'safety claim' relating to structural integrity. As well as being an onerous route to constructing a safety case, this approach will be new to the RP and will require new work to provide such a justification. It will, therefore, form a significant focus of the structural integrity assessment during GDA.

40. During Step 2 I have sought to confirm that the RP is proposing an approach that will identify those components which need a claim that the likelihood of gross failure is so low that it can be discounted from deterministic safety assessments, and that a suitable approach will be developed to justify such claims.

4.1.2 Link with Internal Hazards and Fault Studies

41. The corollary of this approach is that where components are not in the highest reliability category, there needs to be a robust consequences case against gross failure. In general I will consider this aspect in later steps in GDA once the structural integrity classification process is complete, liaising with Fault Studies inspectors in terms of the direct consequences of failure and Internal Hazards inspectors in terms of the indirect consequences of failure as necessary.

42. However, previous experience has shown that ONR's expectations in terms of the pipework failure modes assumed in the Internal Hazard assessments of flooding and pipe-whip can differ from the approaches previously adopted by RPs. In particular the following approaches have been challenged:

- the failure mode for medium energy nuclear classified pipework in internal flooding assessments
- the failure locations for nuclear classified high energy pipework in pipe-whip assessments

43. In terms of the failure mode for medium energy nuclear classified pipework in internal flooding assessments the approach can be to assume only a crack like failure mode with a consequentially small leak area. ONR does not accept that this can be the only failure mode and that much larger leak areas, generally full bore ruptures, will need to be considered in the internal flooding assessment.

44. In terms of the failure locations for nuclear classified high energy pipework in pipe-whip assessments the approach can be to discount failure at welds away for the terminal ends if certain stress and fatigue criteria are met. ONR does not accept that this will always be the case and expects that the consequences of failure at these intermediate locations will also need to be considered in the pipe-whip assessments.

45. The challenge to these approaches affects the Internal Hazards safety case. I therefore worked with the Internal Hazards inspector by attending joint meetings with the RP to explain ONR's position and expectations on these aspects. This is written up in the Step 2 Internal Hazards Assessment Report (Ref. 24), however as the source of these challenges is from the Structural Integrity topic area, I have included this note in my report.

46. The direct and indirect consequences of failure are also taken into account in the Structural Integrity Classification process, and it was identified at the outset that the failure locations assumed for pipe-whip assessments would need to include intermediate weld locations. This is discussed in the Structural Integrity Classification section below.

4.1.3 Strengths

47. The RP has recognised the need to provide a structural integrity safety case that meets the expectations of ONR's SAPs. The Structural Integrity PSR (Ref. 9) proposes a structural integrity classification methodology considering the direct and indirect consequences of failure to identify those components requiring a highest reliability claim. It then accepts that arguments and evidence over and above nuclear design code requirements will be required to support such a claim.

48. I am encouraged that the RP has accepted from the outset of the GDA the need to identify those components needing a highest reliability claim, and that they will need a justification beyond design code requirements. This will allow the RP to commence the work on this aspect early in GDA, which I consider helpful, as whilst this form of claim has featured in the safety cases for other operating reactors in Great Britain (Gas cooled reactors and pressurised water reactors), it will be the first time such an approach has been adopted for a Boiling Water Reactor (BWR) design.

4.1.4 Items that Require Follow-up

49. I have not identified any aspects during my GDA Step 2 assessment of the "Overall Approach" that require to be specifically noted for follow up during the Step 3 structural integrity assessment.

50. Note that previous GDAs have required a Regulatory Observation (RO) to be raised in terms of identifying the those components needing a highest reliability claim, but in the case of this assessment of the UK ABWR I considered the RP to have been sufficiently advanced on this aspect in Step 2 not to require an equivalent RO.

4.1.5 Conclusions

51. Based on the outcome of my assessment of the "Overall Approach" I have concluded that the RP is proposing an approach to structural integrity based on identifying the integrity levels necessary to support the overall safety case. I am satisfied that their approach will include the identification of the components which need a claim that the likelihood of gross failure is so low that it can be discounted from deterministic safety assessments, and that a suitable approach will be developed to justify such claims.

52. I am therefore satisfied with the approach described, and that it meets ONR's expectations in the general sense of the structural integrity 'safety claim'.

4.2 Structural Integrity Classification

4.2.1 Assessment

53. The Structural Integrity PSR (Ref. 9) explains that the structural integrity classification will start from the main safety classification scheme being developed for the UK ABWR, based on three broad classes of SSC; Class 1, Class 2 and Class 3. It then explains that Class 1 will need to be further sub-divided for structural integrity according to the consequences of gross failure. This will identify a sub-set of components which will need a higher reliability claim than can be demonstrated by code compliance.

54. This leads to a position where the identification of the Class 1, Class 2 and Class 3 components is taken directly from the main safety classification. The only structural integrity specific classification will be for Class 1 components to identify those requiring a higher reliability claim. I believe this is suitable approach to the structural integrity classification.

55. The structural integrity classification of the Class 1 components is outlined in the PSR (Ref. 9) and described in more detail in the Structural Integrity Classification Procedure, Ref. 16. This is based on the direct and indirect consequences of gross failure of the components. Where the need for higher integrity claims are required, then the components are identified as either Very High Integrity components (VHI) or High Integrity Components (HI) depending on the consequences of gross failure. For VHI there is no protection from failure, for HI there would be some. VHI components are judged to have a gross failure rate below 10^{-7}/year, whereas High Integrity is judged to have a failure rate between 10^{-7}/year and 10^{-5}/year. There is a recognition that gross failure of a component may need to be considered by region for a complex components such as the RPV.

56. The overall approach to structural integrity classification falls in line with my expectations, essentially concentrating on the Class 1 components and the consequences of gross failure to identify the components needing a higher integrity claim. The ONR SAPs refer to these components as the highest reliability components, but given that the RP is proposing two higher reliability categories. I will, generally, refer to these as the higher integrity components. I also accept that there may be benefit in considering specific regions where it comes to complex vessels.

57. The PSR identifies failure rates for these higher integrity components, which is always difficult for these very high levels of reliability as the actuarial data to support the numbers does not exist. There is therefore an element of judgement in such numbers, however, I accept that the failure rates assumed are in line with previous approaches adopted in Great Britain.

58. The RP has decided to have two higher reliability classes available, but notes that the HI category may or may not be used depending on the outcome of the classification process. Such a two category approach is used on the Gas Cooled Reactor Fleet in Great Britain, but is not used on the Pressurised Water Reactor in operation in Great Britain. I will be interested to see if and how such an approach will be implemented later in GDA, but it is reasonable for the RP to keep the option for two higher reliability classes open at this stage.

59. An initial schedule of components that are likely to be of significance will be used for the structural integrity classification procedure during Step 2 of GDA (Section 3 of Ref. 16) as the actual schedule of Class 1 components is still being developed for the UK ABWR. This schedule will be compared back when the full classification is complete, but I have reviewed this list and am satisfied that it appears comprehensive, noting with it includes non-pressure boundary components that are important to safety for example the reactor internals and RPV support skirt.

60. The RP has applied the classification procedure during Step 2 to identify the components requiring a higher integrity claim, and this is reported in Ref. 32. The application of the procedure took place during the later stages of Step 2 and it was not intended to take this into account in my Step 2 assessment. This is consistent with a step wise approach to GDA, and was recognised in the assessment plan for Step 2 where the plan shows that the classification procedure rather than the classification itself would be assessed during Step 2. Thus for Step 2 my assessment is focussed on the classification procedure described in Ref. 16. The review of the application of the procedure and the identification of the higher integrity components (Ref. 32) will be undertaken during my Step 3 assessment.

61. The classification procedure described in Ref. 16 meets my expectations in terms of considering the consequences of failure. Importantly it considers both the direct and indirect consequences the potential for gross failure in all circumstances and, for example, does not limit the failure location to the terminal ends in pipe-whip assessments. There are a number of simplifications described for the assessments, but these appear conservative. The only aspect where I did raise some questions was in terms of considering the consequences of pipe-whip in a single plane based on the initial jet force direction. I questioned whether the consequences could be worse if a different plane of travel was assumed. The RP agreed to address this aspect in a subsequent assessment considering the worst case scenario in planes up to 30 degrees from the initial direction, and this will be assessed during my review of the classification itself during Step 3 of GDA.

62. The RP recognises that a number of different disciplines will be required to reach conclusions on the structural integrity classification, and a failure modes and effects criticality assessment (FMECA) approach is proposed. This will be subject to audit and review, in particular in terms of the assumptions and judgements, with the outputs subject to review by an expert panel. I accept this is a suitable approach to the classification process provided the assumptions and judgements on which the conclusions are reached are clearly recorded and subject to scrutiny. I will consider this aspect in my review of the outputs from classification during the Step 3 assessment.

4.2.2 Items that Require Follow-up

63. I have not identified any aspects during my GDA Step 2 assessment of the “Structural Integrity Classification” that require to be specifically noted for follow up during the Step 3 structural integrity assessment.

64. A review of the outputs from the classification to identify the components requiring a higher integrity claim will take place during GDA Step 3, but this is part of normal assessment business.

4.2.3 Conclusions

65. Based on the outcome of my assessment of the “Structural Integrity Classification” I have concluded that the RP’s approach to structural integrity classification will be suitable and the associated procedure will allow the RP to identify those components or regions of a component that will require a higher integrity claim.

66. The identification of the components requiring a higher integrity claim has been undertaken by the RP during the latter stages of GDA Step 2 based on the classification procedure. This has not been reviewed as part of my Step 2 assessment, but a review will be undertaken during the GDA Step 3 assessment.

4.3 Safety Case Strategy

4.3.1 Assessment

67. The Structural Integrity PSR (Ref. 9) explains the safety case strategy and sets out how the integrity claims are to be justified. The safety assessment principles (SAPs) on the integrity of metal components provide a framework to assess the safety case in EMC.1 to EMC.34, noting that EMC.1 to EMC.3 are specifically for the highest reliability components (See Table 1).

68. For standard Class 1, Class 2 and Class 3 components the RP is proposing that design and manufacture to recognised nuclear and non-nuclear design codes will provide the primary evidence to support the reliability claims necessary, but that known degradation mechanisms will also be addressed. SAPs EMC.4 to EMC.34 apply in these situations, but I accept that compliance with an appropriate design code can form the main basis of demonstrating compliance with these SAPs. ONR will still take an active interest in these components, particularly where there are novel design features or specific degradation threats, but also in terms of sampling the code compliance aspects in subsequent steps of GDA.

69. For the highest reliability components SAPs EMC.1 to EMC.3 apply, and this will require a demonstration beyond design code compliance. EMC.1 states that the safety case must be especially robust in order that an engineering judgement can be made on two key requirements:

- The metal component is as defect free as possible
- The metal component or structure should be tolerant of defects

70. EMC.2 and EMC.3 state that the safety case should include a comprehensive examination of relevant scientific and technical issues, and provide evidence that the necessary level of integrity has been achieved. Compliance with a recognised nuclear design code will be an important starting point, but it will not provide the full justification.

71. The RP proposes to use the structure developed by TAGSI for the demonstration of 'Incredibility of Failure' in structural integrity safety cases, Ref. 23, in order to provide the necessary demonstration for the higher integrity components. Whilst ONR does not use the term 'Incredibility of Failure', ONR does recognise that the approach proposed by TAGSI can provide a suitable framework for justifying a higher reliability claim. In particular the conceptual defence in depth provided by the four legs of the TAGSI approach can be useful in demonstrating these very high levels of reliability.

72. I am therefore satisfied that a safety case based on the TAGSI approach is suitable, way forward for the higher integrity components and will allow a judgement to be reached on whether the highest reliability claims have been justified.

4.3.2 Regulatory Observations related to the Safety Case Strategy

73. ONR anticipates that the RP will need to develop a programme of new work to deliver the beyond design code justifications needed to support this case. In line with previous GDAs ONR has generated Regulator Observations (ROs) where it would be useful to aid the RP in meeting regulatory expectations in Step 3 and beyond. I have raised three ROs related to the Safety Case Strategy and these are associated with the justification of the highest reliability components (Ref. 26):

- RO-ABWR-0001 - Avoidance of Fracture - Margins based on the size of Crack-Like Defects

- RO-ABWR-0003 - RPV Design (use of forgings and plate materials)
- RO-ABWR-0004 - Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary

74. RO-ABWR-0001 is discussed in the next section 'Avoidance of Fracture', but RO-ABWR-0003 and RO-ABWR-0004 are discussed in this section.

75. As explained in the previous section on 'Structural Integrity Classification', the actual identification of those components requiring a highest reliability claim has not yet been reviewed at this stage of my GDA assessment. However, previous experience has shown that the RPV will come into this category in terms of its main pressure boundary and possibly its supports. I have therefore raised RO-ABWR-0003 and RO-ABWR-0004 at an early stage assuming that the RPV would be identified as a highest reliability component, in order to present some of the beyond design code regulatory expectations for this component to show that it is as defect free as possible and it is tolerant of defects.

4.3.2.1 RO-ABWR-0003 - RPV Design

76. RO-ABWR-0003 – 'RPV Design' is intended to provide guidance on ONR's expectations in terms of justifying the design of the RPV and forgings that make up the RPV.

77. The RPV will be manufactured for a number of major component parts that are welded together (shells, domes, support skirts, nozzles etc). There is a need to choose a product form for these major component parts which minimises the number and length of welds, it should have good material properties, and the product form should avoid placing the welds in high stress locations or adverse environments

78. In order to satisfy this need, there is a general expectation that the RPV will, where possible, be manufactured from low alloy ferritic forgings which will be chosen to minimise the number and length of welds in the vessel, and that the weldments will, where possible, avoid locations of high stress or neutron irradiation.

79. This will need to be demonstrated, and where welded plate material is proposed for either the pressure boundary, for example the RPV head, or the support skirt, a detailed justification will be required to demonstrate why the proposal is adequate taking into account properties, propensity to include defects and relative weld lengths.

80. The RP has proposed a credible Resolution Plan to address the RO (Ref. 26) which will provide the necessary responses in later stages of GDA. My assessment of these responses will commence during Step 3 of GDA.

4.3.2.2 RO-ABWR-0004 - Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary

81. 4.3.2.2 RO-ABWR-0004 – 'Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary' is intended to provide guidance on ONR's expectations on the manufacture of the forgings forming the RPV pressure boundary and subsequent construction of this pressure boundary.

82. The specifications provided in the nuclear pressure vessel design codes will define the basis for the parameters involved in the manufacturing processes such as chemical compositions; forging processes; quench and temper heat treatments; welding and cladding procedures etc. However, ONR's experience has shown that it is necessary for the RP to show that they understand the detailed interaction of these parameters in

order to apply controls over and above those specified in the design codes to ensure satisfactory finished forgings and vessels.

83. This RO is therefore intended to provide guidance on ONR's expectations in terms of these interactions including control of the: chemical composition in the forgings; casting and forging process; and welding and cladding processes.

84. The intent is to provide a framework to show how the controls will achieve the necessary initial properties and homogeneity in the base forgings; that welds and cladding will be to the required quality; and that these properties are maintained through life.

85. Again, the RP has proposed a credible Resolution Plan to address the RO (Ref. 26) which will provide the necessary responses in later stages of GDA. My assessment of these responses will commence during Step 3 of GDA.

4.3.3 Items that Require Follow-up

86. I have not identified any aspects during my GDA Step 2 assessment of the "Safety Case Strategy" that need to be specifically noted for follow up during the Step 3 structural integrity assessment.

87. I have raised three ROs related to this topic, two of which are described in this section. Credible resolution plans have been developed by the RP to address these and they will be followed up as part of normal assessment business during Step 3.

4.3.4 Conclusions

88. Based on the outcome of my assessment of the "Safety Case Strategy" I have concluded that the RP has developed an approach that should allow the appropriate integrity levels to be demonstrated, and has proposed a suitable structure for the beyond design code demonstration for the highest reliability components that should meet ONR's expectation's.

4.4 Avoidance of Fracture

4.4.1 Assessment

89. ONR's expectation for the highest reliability components is that the component or structure should be as defect free as possible and is demonstrated to be tolerant of defects (ONR SAPs EMC.1). In particular the limiting defect size needs to be shown to be larger that the defect size that can be reliably detected by the applied examination techniques. This is provided through an Avoidance of Fracture demonstration.

90. This involves a detailed fracture mechanics based defect tolerance assessment, using verifiable material properties, to determine the limiting defect sizes for these components at the start of life taking into account any potential for through-life crack growth. The non-destructive examinations being proposed for the components then need to be shown to be able to reliably detect such start of life defects by a suitable margin. Such a demonstration is beyond the design code compliance required for these components

91. The need to provide such a demonstration is reflected in the PSR provided by the RP, Ref. 9. The PSR recognises the importance of such a contribution to any high reliability claim. It proposes a detailed fracture mechanics assessment for the VHI and HI components, to determine limiting start of life defects taking account of the potential for through-life growth, and then demonstrating that the qualified non-destructive

examinations undertaken at the time of manufacture can reliably detect these start of life defects. I am therefore satisfied that the RP is proposing a suitable Avoidance of Fracture demonstration.

92. During Step 2 the RP submitted documentation to describe their proposed approach, with the main work needed to provide the demonstration being undertaken during Step 3 and into Step 4:

- the Weld Ranking Procedure (Ref. 17) report describes the methodology for identifying the limiting areas on the higher reliability components for detailed assessment during GDA
- the Defect Tolerance Assessment Plan (Ref. 18) report describes the methodology for undertaking the fracture mechanics assessment
- the Inspection Assessment Plan report (Ref. 19) describes the RP's methodology for demonstrating capability of the end of manufacture non-destructive examination proposals
- the Inspection Qualification Strategy report (Ref. 20) describes the methodology for qualifying the end of manufacture non-destructive examinations
- and finally Section 6 of the PSR (Ref. 9), Safety Case strategy, describes, in effect, the process of bringing together the overall Avoidance of Fracture demonstration within the Safety Case.

93. The RP was also intending to complete the weld ranking on the higher reliability components and commence the defect tolerance assessment work towards the end of GDA Step 2, but the deliverables for this would occur too late in Step 2 for me to take this into account in my Step 2 assessment. Thus my Step 2 assessment has concentrated on a review of the proposed approach in line with my assessment plan, with the main work needed to provide the Avoidance of Fracture demonstration being considered during the Step 3 and Step 4 assessments.

94. I have therefore reviewed the documentation describing their proposed approach and overall I am satisfied that it should provide a sound basis for an Avoidance of Fracture demonstration for the purposes of GDA. Detail comments on the documentation are provided in the next few paragraphs.

4.4.1.1 Weld Ranking Procedure (Ref. 17)

95. The RP is proposing to undertake detailed avoidance of fracture demonstrations on what it believes will be the limiting regions of the higher integrity components. Previous GDAs have accepted that it is not necessary to provide an avoidance of fracture demonstration for every region of each higher integrity component during GDA, but it is necessary to provide one for what are expected to be the limiting regions of the components, with any remaining demonstrations taking place after GDA has finished. Thus I am satisfied with the RP's overall approach.

96. The Weld Ranking Procedure, Ref. 17, provides the RP's approach to identifying the limiting regions in the higher integrity components. It provides a structured approach to identifying the limiting regions by semi-qualitatively taking into account aspects related to the size of the limiting defect and the difficulty in detecting such a defect in order to identify those areas which are likely to be limiting in an avoidance of fracture demonstration.

97. Whilst there is inevitably an element of subjectivity in such a semi-quantitative ranking process, I am satisfied that the process should provide a suitable approach to identifying the limiting regions. The ranking process uses weighting factors to combine

the various aspects considered in the process. These weighting factors will affect the list of regions to be considered and are not established in the report. However, the procedure includes a review of the ranking by an expert panel to address any anomalies from the ranking process and I will review the weighting factors and final list of limiting regions once the weld ranking has been completed. It should be noted that although the procedure is termed a weld ranking procedure, it will also include areas of the parent forgings subject to high stresses or inspection difficulties. This is important, as whilst the limiting regions in the components are generally the welded regions, the potential for parent forgings to be limiting cannot be excluded.

98. The actual application of the weld ranking procedure to the higher integrity components was completed in the later stages of Step 2, and is reported in Ref. 33. My Step 2 assessment does not include the review of the Ref. 33, and this will be undertaken during Step 3 in line with the Step 2 assessment plan.

99. Thus for Step 2 I conclude that I am content with the weld ranking procedure described in Ref. 17. Consideration of the limiting regions identified by the procedure for the detailed avoidance of fracture demonstrations, and whether they provide sufficient coverage of the higher integrity components, will be addressed in the review of Ref. 33, the application of the procedure, during Step 3.

4.4.1.2 Defect Tolerance Assessment Plan (Ref. 18)

100. Ref. 18, establishes the basic approach that will be used in the fracture mechanics assessments of the selected regions of the higher integrity components to establish the limiting defect sizes. The RP proposes to use the R6 defect assessment procedure (Ref. 27) to undertake this work. The R6 defect assessment procedure 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. I am therefore satisfied with the choice of this procedure as the basis for the fracture mechanics assessment.

101. Ref. 18 includes details of the important parameters which will be needed for the proposed assessments including: the treatment of primary and secondary stress; the treatment of residual stress; the failure assessment diagrams to be used; the material properties adopted; the postulated defect aspect ratios; crack growth assumptions; use of ductile tearing. My review did not identify any particular areas of concern in these details, and I am satisfied that the approaches should lead to a conservative assessment. Ref 18 also notes the intent to show a margin of at least two between the size of defect that can be reliably detected by the qualified examination and the limiting deft size taking account of through-life crack growth, which is consistent with the approach established in previous GDAs.

102. Whilst I am satisfied with the RP’s proposals, it was noticeable that the RP had to refer back to public domain information from previous GDAs to establish ONR’s expectations on a number of aspects. Since the publication of Ref. 18 I have raised Regulatory Observation RO-ABWR-0001 on the Avoidance of Fracture to define ONR’s expectations in terms of the UK ABWR GDA, and I would expect this RO to be referenced in subsequent submissions. RO-ABWR-0001 is discussed in more detail below.

4.4.1.3 Inspection Assessment Plan (Ref. 19)

103. Ref. 19 describes the approach that will be applied during GDA to provide confidence that the proposed end of manufacturing inspections can reliably detect defects of structural concern. It proposes a methodology for demonstrating the reliability of the

end of manufacturing inspections including producing Technical Justifications (TJs) for the proposed inspections, independent reviews of the TJs by a GDA specific qualification body; and anticipates that ultrasonic inspection approaches will be used in the main.

104. I consider these to be an important set of commitments. Whilst it is important to identify the limiting defect sizes in a component, it is the actual inspections which will give the confirmation that the components are free of structurally significant defects. The established code based inspections during manufacture are predominantly focussed on radiographic examination. These are important in ascertaining the general quality of manufacture, but are not necessarily effective at detecting crack like defects. I therefore consider the beyond design codes qualified ultrasonic inspections being proposed by the RP to be the most important examinations in support of the avoidance of fracture demonstration.

105. The demonstration of the capability and reliability of the proposed inspections through a Technical Justification subject to independent assessment by a GDA specific qualification body should ensure that the proposals are robustly based, and I support such an approach. The Technical Justification will effectively only give a partial qualification, and more work will be required post GDA including the provision of representative test pieces, but I consider the RP's proposals to be sufficient for GDA purposes.

106. Ref. 19 is specifically written in support of the Avoidance of Fracture demonstration, and focuses on the end of manufacture inspection. What also need to be recognised is that the other more code based inspections undertaken during manufacture are important in demonstrating the quality of manufacture. This should be addressed through the overall safety case and I will consider this further during later stages of GDA.

4.4.1.4 Inspection Qualification Strategy (Ref. 20)

107. Ref. 20 described the methodology which will be adopted by the RP for the full qualification of the end of manufacture non-destructive examination techniques (as opposed to the partial qualification undertaken for the GDA demonstration). The strategy is to use an ENIQ based methodology (Ref. 28). ENIQ is the European Network for Inspection and Qualification, and it is a recognised authority on inspection qualification.

108. I consider ENIQ's approaches to be well founded, and capable of meetings ONR's expectations. The main elements of the methodology are to develop an inspection specification to define defect types and performance requirements, develop inspection techniques to meet the requirements of that specification, and then qualification of the inspection procedures and personnel through a combination of technical justifications and practical trails. I support the RP's proposals on this aspect and note that they intend to use an independent third party for the qualification aspects.

4.4.1.5 Safety Case Strategy (Section 6 of Ref. 9)

109. Section 6 of the PSR (Ref. 9) on the Safety Case Strategy brings together the limiting defect sizes and qualified inspection aspects into an avoidance of fracture demonstration in support of the highest reliability claims. As discussed in Section 4.3.1, I am satisfied that the approach being proposed by the RP based on the TAGSI structure of Ref. 23 will allow a judgement to be reached on whether the highest reliability claims have been justified.

4.4.2 Regulatory Observation related to the Avoidance of Fracture

110. As noted in Section 4.3.2, ONR has generated Regulatory Observations where it would be useful to aid the RP in meeting regulatory expectations in step 3 and beyond. Three ROs are related to the Safety Case Strategy and the justification of the higher integrity components, but RO-UKABWR-0001 on ‘Avoidance of Fracture’ is discussed in this section rather than the section on Safety Case Strategy.

4.4.2.1 RO-UKABWR-0001 - Avoidance of Fracture

111. RO-UKABWR-0001 on ‘Avoidance of Fracture’ is intended to provide guidance on demonstrating that the highest reliability components should be tolerant of defects. The demonstration brings together the three important aspects of: demonstrable material toughness properties; limiting defect size calculations; and qualified inspection in order to show that the qualified manufacturing inspections can detect defects of structural concern.

112. This is an important piece of work needed to support the highest reliability claims, and it is anticipated that it will be an extensive piece of new work for the RP. It is therefore important that the RP has a good understanding of ONR’s expectations.

113. The RO has four Regulatory Observation Actions (ROAs) covering:

- Material Properties
- Fracture Assessment
- Manufacturing Inspection
- Overall Avoidance of Fracture Demonstration

114. It is clear from the preceding discussion on the Avoidance of Fracture Demonstration that the RP has used public domain information from previous GDAs to inform themselves of ONR’s expectations. This is welcomed as they have proposed an approach that is in line with ONR general expectations, but the RO and associated ROAs are specific to the UK ABWR GDA, and the RP can now refer to these as appropriate rather than previous GDAs information in the public domain.

115. I also consider that the RO is important to ensure that all aspects are suitably addressed. For example the RP’s proposals during Step 2 on Avoidance of Fracture were potentially quite limited in terms of the material property aspects needed to support the demonstration. This would have been picked up on by assessment during subsequent steps of GDA, but this aspect can now be addressed by the RP in response to the RO.

116. The RP has proposed a credible Resolution Plan to address the RO (Ref. 26) which will provide a sequenced set of responses through Step 3 and into Step 4. This is a large piece of work, and my assessment will commence in Step 3 and extend into Step 4.

4.4.3 Strengths

117. The provision of an avoidance of Fracture demonstration is a significant piece of new work for the RP. I am encouraged that the RP has accepted that this is an important part of the overall justification of a highest reliability claim. I am also encouraged that the RP has looked back to the public domain information available from previous GDAs to understand ONR’s general expectations in this area to allow them to make a set of proposals that are in line with those expectations.

4.4.4 Items that Require Follow-up

118. I have not identified any aspects during my GDA Step 2 assessment of the “Avoidance of Fracture” that require to be specifically noted for follow up during the Step 3 structural integrity assessment.

119. The assessment of the Avoidance of Fracture demonstration itself will take place during GDA Step 3 and continue into Step 4, but this is part of normal assessment business.

4.4.5 Conclusions

120. Based on the outcome of my assessment of the “Avoidance of Fracture” I have concluded that the RP’s approach should be suitable for supporting the justification of a highest reliability claim.

4.5 Applicable Codes and Standards

4.5.1 Assessment

121. As noted in Section 4.3 on safety case strategy, the RP is proposing for standard Class 1, Class 2 and Class 3 components that design and manufacture to recognised nuclear and non-nuclear design codes will provide the primary evidence to support the reliability claims necessary. For the higher reliability components compliance with a recognised nuclear design code will be an important starting point, but further beyond design code demonstrations are required.

122. The PSR (Ref. 9) provides information on the types of design code being proposed. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III (ASME III) is proposed for the Class 1 and Class 2 components and the ASME Boiler and Pressure Vessel Code Section VIII (ASME VIII) for Class 3 components, supplemented by other recognised international standards as appropriate.

123. ONR has long experience of these internationally recognised codes, and I would not envisage any problems with using these codes as the basis for design and manufacture. ASME III is a nuclear specific code and is therefore appropriate for Class 1 and Class 2 components and ASME VIII and the other standards are normal industrial codes which are suitable for Class 3 components, so this is in line with ONR SAP ECS.3 and paragraph 158.

124. Although not mentioned in the PSR, I understand that the RP will, in principle, use the latest version of the ASME III code for its design. This will be confirmed during later stages of GDA, but is potentially helpful as it removes the need to consider code developments between the code edition used and the current code.

125. I also understand that the RP’s original ABWR plant would have designed to meet the equivalent requirements from the Japanese Society of Mechanical Engineers (JSME), so the pressure vessel, piping, valves etc will also need to be assessed against the ASME code requirements. I do not believe that this should cause any fundamental difficulties, due to similarities between the codes, but the RP will need to show compliance with the code it has chosen for the UK ABWR. The RP may, however, decide to show code compliance based on the design of the limiting locations in components for GDA purposes rather than the components as a whole. I would not object to such an approach providing the locations could be shown to be limiting and full code compliance was committed to post GDA.

126. ASME III has a graded approach to the design and manufacture of nuclear pressure components using three classes, ASME III Class 1, ASME III Class 2 and ASME III

Class 3. Class 1 is the highest class and Class 3 the lowest class. The application of these classes to individual components is not defined in the design codes themselves, but by local requirements related to quality classification. I consider this to be an important link between the integrity levels and safety case for the Class 1 and Class 2 components which use ASME III (noting that the Class 1 and 2 in this sentence is nuclear safety class).

127. Appendix A of Ref. 9 provides an early indication of the proposals for quality classification the UK ABWR. My initial review of these proposals suggests that they generally appear reasonable, but there are some potential anomalies. For example Figure A1 suggests a quality class B for the main steam pipework down stream of the MSIVs, but the associated valves are quality class D. I will therefore undertake a more detailed review of these proposals during Step 3 once they have been finalised to ensure that they meet ONR's expectations.

4.5.2 Strengths

128. The RP's decision to use ASME Boiler and Pressure Vessel Code, as the basis for the main components in the design, assists in ONR's assessment due to ONR's long experience with these codes.

4.5.3 Items that Require Follow-up

129. I have not identified any aspects during my GDA Step 2 assessment of the "Applicable Codes and Standards" that require to be specifically noted for follow up during the Step 3 structural integrity assessment.

130. Further work will be needed on the quality classifications to be adopted during GDA Step 3 once the RP's proposals have been finalised, but this is part of normal assessment business.

4.5.4 Conclusions

131. Based on the outcome of my assessment of the "Applicable Codes and Standards" I have concluded that the RP's approach of using the nuclear specific ASME Boiler and Pressure Vessel Section III design code for Class 1 and Class 2 Components and normal industrial standards for Class 3 components should be acceptable.

4.6 Material Choices and Degradation Mechanisms

4.6.1 Assessment

132. I have considered the material choices and potential degradation mechanisms that could affect the UK ABWR as there is an implicit structural integrity claim related to the 60 year design life of the reactor.

133. There are a number of aspects to this; some will be addressed under other topics. Fatigue loading from mechanical or thermal hydraulic source will generally be considered through design code assessment and the potential for fatigue crack growth will be considered in the calculations associated with the Avoidance of Fracture demonstration. Irradiation damage to the belt line of the RPV pressure shell will be considered within the design and manufacture of the RPV, for example control on material composition for the RPV forgings. This section, however, is concerned mainly with materials chosen for the design and the environmentally assisted cracking phenomena that could affect the design. These aspects affect the structural integrity topic area but are intrinsically linked with the reactor chemistry area in terms of the

environment. This has been reflected in joint meetings with Reactor Chemistry and Structural Integrity SMEs.

134. The RP has provided an overview of material selection and potential degradation mechanisms in the PSR (Ref. 9). This is a high level description of the issues with some very generic statements and it is of limited use for my assessment. It does, however, acknowledge the threat that Stress Corrosion Cracking (SCC) has posed to BWR plant, and a further document on the approaches used to mitigate the threat from SCC in the UK ABWR has been provided (Ref. 15).

135. In terms of material choices, the individual design summaries (Ref. 10, 11, 12 and 13) do provide some information on the materials that have been selected for the major components. The RPV will be manufactured from low alloy steel, clad with stainless steel and a nickel-base alloy in its lower regions. The main steam lines will be made from carbon steel, and the RP is proposing to use carbon steel for the feedwater piping. The MSIVs will be manufactured from carbon steel castings or forgings. Nuclear grade low carbon stainless steel will be used for the reactor internals along with niobium stabilised grades of nickel-base alloys (Ref. 9). This information gives an appreciation that the design uses established materials that are generally suitable for their purpose, but that they are not necessarily immune from degradation, and more detailed review will be needed in subsequent stage of GDA.

136. SCC requires a combination of susceptible material, stress and environment and has affected light water reactor components made from stainless steels and nickel-base alloys (pressurised water designs and boiling water reactor designs). It is generally recognised that earlier designs of BWR plant have suffered extensively from SCC on their austenitic stainless steels and nickel-base alloys, for example on recirculation loop pipework, core internal structures, CRD stub tubes. IASCC is a subset of the mechanism where high neutron irradiation can make austenitic stainless steel susceptible to SCC and BWR designs have suffered from Irradiation Assisted SCC (IASCC) on near core components. These damage mechanisms were recognised at the time the ABWR was being designed and Ref. 15 provides an overview of the approaches taken to mitigate this threat in the UK ABWR design.

137. I have reviewed Ref. 15 and consider it to provide a useful high level description of the approaches being taken. Firstly it describes the fundamental improvements incorporated in the ABWR design that eliminate some of the historical difficulties associated with BWR designs. Most significant of these is the elimination of external recirculation loops by using Reactor Internal Pumps which I consider to be an important improvement.

138. There are then the detailed material choices and compositions used for the stainless steels and the nickel-base alloys. For example nuclear grade low carbon stainless steels will be used, niobium stabilisation in the nickel-base Alloy 600, and use of Alloy 82 as the weld material for the nickel-base alloy rather than Alloy 182. There are then the material processing and conditioning to reduce the chance of sensitisation and residual stress levels. I consider these to be useful approaches.

139. Finally, and very importantly, there is the water chemistry aspect which controls the environment. At the start of Step 2 the water chemistry for the UK ABWR had not been chosen, as reflected in the PSR (Ref. 9), but later in Step 2 a decision was reached to use hydrogen water chemistry in conjunction with platinum injection (noble metal chemistry) and zinc injection, as reflected in Ref. 15. This is intended to further reduce the susceptibility to SCC during normal operation. The choice of water chemistry is the subject of much more extensive discussion with the Reactor Chemistry GDA Step 2 Assessment Report (Ref. 29), but its significance is noted here.

140. Whilst more detailed review and assessment will be required later in GDA, overall I can conclude that a multi-faceted approach is being taken to mitigate the threat from the stress corrosion cracking degradation mechanism. However, I remained uncertain on whether the RP was claiming that their approach would eliminate the potential for SCC over the 60 year life of the reactor or whether the RP was claiming that their approach will minimise the likelihood of SCC, but that it cannot be entirely ruled out. I therefore raised RQ-ABWR-0163 (Ref. 8).

141. The RP's response to RQ-ABWR-0163 states that whilst substantial efforts are made to eliminate SCC/IASCC from the UK ABWR, the onset cannot be entirely ruled out for the 60 year design life of the plant. A specific ISI programme will therefore be applied to detect SCC/IASCC long before it can threaten structural integrity. Further detail will be provided in a report titled 'Material Selection Report' which will be provided early in GDA Step 3.

142. I consider this to be a suitable response. Clearly a range of measures have been put in place to reduce the likelihood of it occurring, but it would be very difficult to substantiate a claim that SCC/IASCC had been eliminated. Given that the most safety significant areas susceptible to SCC have actually been eliminated from the ABWR design (the austenitic stainless steel recirculation loop pipework) I consider that it is reasonable for the RP to suggest a safety case based on a claim of minimising the likelihood of occurrence supplemented by an ISI programme to detect any problems before they become significant. The 'Material Selection Report' identified in the response to RQ-ABWR-0163 would appear to provide the main basis for the RP's case, and this will be subject to detail assessment during Step 3.

143. In addition to my review of Ref. 15 I commissioned another Structural Integrity inspector within ONR to undertake a wider review of the Material Degradation Mechanisms that may affect the UK ABWR both to inform my assessment during Step 2 and future assessment during Step 3. The internal ONR report (Ref. 25) confirms that SCC and IASCC are the main threats to the integrity of the plant, and concludes that whilst mitigation measures may reduce the likelihood of occurrence, it would be difficult to substantiate that it will have been eliminated.

144. Ref. 25 notes the improvement that have been seen on existing plant by applying hydrogen water chemistry and noble metal chemistry, but that this chemistry is not necessarily effective in all operating mode, and specific attention will need to be played to start up and shut down operations. This chemistry will also not be effective in terms of components susceptible to IASCC, so there may need to be further consideration of the material choice for components subject to high levels of irradiation. The report also asks questions on the effectiveness of surface stress improvement techniques when subject to stress cycles, and whether the nickel-base alloys proposed for the design may be affected by the use of a hydrogen water chemistry.

145. The points raised by Ref. 25 cannot be addressed at this stage of GDA, but will be taken into account during the assessment of the 'Material Selection Report' to be provided early in Step 3 of GDA. One aspect to note is that the justification of the material selection will need to take into account the UK ABWR water chemistry,.

146. As well as the general consideration of material choices and degradation mechanisms discussed above, there were two more specific considerations identified during Step 2, the material choice to the reactor water clean up system and the potential for chloride ingress.

4.6.1.1 Reactor Water Clean Up System – Material Choice

147. The Reactor Water Clean Up System is connected to the RPV via a bottom drain line and a mid-height connection. I understand that Japanese specification ABWRs use carbon steel for this system, but this can lead to radiological protection difficulties due to the high dose rates in the vicinity of the pipework. An alternative approach, where hydrogen water chemistry with zinc addition is employed, is to use stainless steel for this system as the zinc addition will reduce the dose rates. However, using stainless steel pipework brings back the potential for SCC that did not exist with the carbon steel design.

148. The RP has yet to make a decision on the material choice on the system for the UK ABWR, but during joint Reactor Chemistry/Structural Integrity meeting ONR provided clear advice that this would need to be based on an As Low As Reasonably Practicable (ALARP) decision taking into account radiological protection, reactor chemistry and structural integrity concerns. The RP has not yet declared when such a decision will be made, nor on the form of the justification, but I intend to progress this matter further during Step 3.

149. Associated with this aspect is RQ-ABWR-0082 raised by the Reactor Chemistry inspectors which question the purpose of the RPV drain line on the Reactor Water Clean Up System and for an ALARP justification for the presence of the drain line. The RP's response gave an explanation for its presence and suggested that it was ALARP to keep the drain line, but this response is not yet considered to be sufficient and will be discussed further during Step 3. More detail is available in the Reactor Chemistry Step 2 report (Ref. 29).

4.6.1.2 Potential for Chloride Ingress

150. Control of chloride is an important factor in the protection of the reactor circuit against stress corrosion cracking. The UK ABWR will use sea water as the heat sink for the steam turbine condenser which operates at sub-ambient pressure, so there is the potential for an ingress of sea water should there be a leak or failure of the condenser tubes. I therefore raised RQ-ABWR-0134 (Ref. 8) to ask about the level protection available against chloride ingress should there be a tube leak or tube failure.

151. I asked questions about small scale tube failures in terms detection systems used to protect the plant against such events; their safety classification; any required operator action; and the response time necessary to protect the plant. I also asked about the main condenser failure on the Hamaoka-5 ABWR in Japan in May 2011 as this suffered a very large ingress of sea water into the coolant system of that plant. I asked about the safety significance of such an event; the potential for long term damage to the reactor system; and what design changes are being considered to prevent such a failure occurring on a UK ABWR condenser.

152. The RP's response describes the protection, detection and time for operator action in terms of a small condenser failure. Protection against small leaks is provided via the condensate demineraliser and detection through conductivity monitoring (both Class 3 systems) with operator action required within 30 minutes. In terms of the large condenser failure, there was not thought to be any significant influence on the safety functions, and periodical inspection will be performed to ensure long term integrity. The RP's response also states that the UK ABWR condenser design will take account of the lessons learnt form the Hamaoka-5 failure to prevent a similar failure.

153. The response provides useful background information, but I am not yet fully satisfied that this aspect has been satisfactorily addressed, and the measures will need to be incorporated into the safety case. I intend to take these matters forward during the Step 3 assessment.

4.6.2 Items that Require Follow-up

154. I have identified three aspects during my GDA Step 2 assessment of “Material Choices and Degradation Mechanisms” that require to be specifically noted for follow up during the Step 3 structural integrity assessment. These are associated with the provision of a material selection justification taking into account UK ABWR water chemistry; the optimisation of the material choice for the Reactor Water Clean Up system; and the inclusion of the potential for chloride ingress in the safety case.

155. The remaining assessment work in this area in Step 3 will be part of normal assessment business.

4.6.3 Conclusions

156. Based on the outcome of my assessment of the “Material Choices and Degradation Mechanisms” I accept that the RP is taking a multi-faceted approach to mitigate the threat from SCC and consider that it is reasonable for the RP to suggest a safety case based on a claim of minimising the likelihood of occurrence supplement by an ISI programme to detect any problems before they become significant.

157. The following aspects have been noted for specific follow up in Step 3:

- Provision of a material selection justification taking into account UK ABWR specific water chemistry
- Optimised material choice for the Reactor Water Clean Up System
- Inclusion of the potential for chloride ingress, including protection measures and consequences, in the safety case

4.7 Design Summaries for Major Components

4.7.1 Assessment

158. The RP has provided design summaries, from a structural integrity perspective for the RPV, Main Steam Piping, Feedwater Piping and MSIVs, Refs. 10, 11, 12, and 13. This covers the major components in the nuclear island and gives an overview of the main design features, the functional requirements, the design requirements and diagrams of the main features of these components.

159. These are fairly high level documents, but are sufficient for me to gain a better understanding of the ABWR design in terms of the structural integrity of the major components and to identify whether there are design features that differ from approaches previously seen in Great Britain or appear complex that may require further consideration.

160. In general the design of the major components appears conventional. In terms of the RPV I noted that welded plates are used for the head and potentially for some of the main shell strakes. This differs from the approaches previously seen in Great Britain where forgings may be expected and this has been taken forward in RO-ABWR-0003 on RPV Design (discussed in section 4.3.2). The other aspect that I noted was the design of the Control Rod Drive (CRD) Penetrations appears complex. I have taken that forward in RO-ABWR-0002, discussed below.

161. Apart form these two aspects, there was nothing which I considered of particular note at this stage of GDA, but I may refer back to these documents (or similar) in subsequent stages as the safety case becomes more detailed and more developed.

4.7.2 Regulatory Observations related to the Design Summaries for Major Components

4.7.2.1 RO-ABWR-0002 – CRD Penetration Design

162. The detail design of the ABWR CRD Penetrations is complex. It consists of a nickel-base alloy CRD stub tube welded via a full penetration weld to the nickel-base alloy cladding of the low alloy bottom head; and a stainless steel CRD housing welded to the nickel-base alloy CRD stub tube welded via a partial penetration weld.

163. Due to the complexity of the design it will be necessary to demonstrate the initial and through-life integrity of the pressure boundary.

164. RO-ABWR-0002 is intended to provide guidance on ONR's expectations in this area, covering:

- loading mechanisms
- pressure vessel design code compliance
- inspection approaches at manufacture and through life
- material choice to minimise the potential for through-life degradation
- operational experience with this design of penetration

165. The RP has proposed a credible Resolution Plan to address the RO (Ref. 26) which will provide the necessary responses in later stages of GDA. My assessment of these responses will commence during Step 3 of GDA.

4.7.3 Items that Require Follow-up

166. I have not identified any aspects during my GDA Step 2 assessment of the "Design Summaries for Major Components" that require to be specifically noted for follow up during the Step 3 structural integrity assessment.

167. Further work will be needed on the ROs associated with this topic, and the documents may be referred to in subsequent stages of GDA, but this is part of normal assessment business.

4.7.4 Conclusions

168. Based on the outcome of my assessment of the "Design Summaries for Major Components" I have concluded that the design of the major components is largely conventional which raises no particular concern. Where there appear to be differences from the approaches previously seen in Great Britain or specific complexity I have raised ROs to assist the RP in meeting ONR's expectations.

4.8 Specific Characteristics of the ABWR Balance of Plant

4.8.1 Assessment

169. The RP has provided a design summary of the main components in the Balance of Plant (BOP) from a structural integrity perspective. BOP is considered as being downstream from the second MSIV through to the first check valve on the feed-water line. In common with other BWR designs the UK ABWR is a single coolant circuit design such that the primary coolant from the reactor is used directly in the steam

turbine outside of the primary containment. An integrity failure in the BOP may therefore lead to a loss of primary coolant. Thus the integrity of the BOP has specific significance in the design.

170. The description and classification indicates that the majority of the components are Safety Class 3 and will be designed to normal industrial standards. This is because the radiological consequences of a large break in the BOP is considered to be small and would not cause core damage. The only area that is considered to be Safety Class 2 is the main steam line from the second MSIV to the next valves down-stream, and would be designed to nuclear standards.

171. This classification decision would therefore imply that there are no higher integrity claims to be placed on the BOP. This is important in terms of the structural integrity assessment and I therefore sought to clarify the RPs position by raising RQ-ABWR-0164 (Ref. 8). The RQ asked whether there were any higher integrity claims thought likely for the pressure retaining components in the Balance of Plant. The response confirmed that there would be no higher integrity claims and the majority of the plant is Safety Class 3.

172. Thus, provided the decision to place the majority of the BOP components at Safety Class 3 can be sustained, there should be no significant integrity claims on the BOP components as they will be designed to normal industrial standards, and in particular there should be no higher integrity claims.

173. I will therefore liaise with the Fault Studies inspectors in later stages of GDA once the fault schedule has been fully developed to confirm that the BOP is at Safety Class 3. Provided this is the case the structural integrity assessment will only have a limited involvement in this area.

174. Designing the Safety Class 2 main steam line to nuclear standards would normally meet my expectations, however, the associated down stream valves are not classified to the same level and are only Safety Class 3. This is unusual and I will need to understand the RP's rationale on this point. This aspect is also referred to in Section 4.5.1 in the discussion on quality standards and design codes, and I understand that at this stage the RP's proposals are only an early indication of safety class. I will therefore undertake a further review during Step 3 once the RP's proposals have been finalised.

4.8.2 Items that Require Follow-up

175. I have identified one aspect during my GDA Step 2 assessment of "Specific Characteristics of the ABWR Balance of Plant" that requires to be specifically noted for follow up during the Step 3 structural integrity assessment. I will need to confirm that the BOP remains at Safety Class 3 in a later stage of GDA once the fault schedule has been fully developed to ensure that there are no significant integrity claims on the BOP components

176. Further work will be needed during GDA Step 3 to understand the basis for using Safety Class 2 main steam line pipework with Safety Class 3 valves downstream once the RP's proposals are finalised, but this is part of normal assessment business.

4.8.3 Conclusions

177. Based on the outcome of my assessment of the "Specific Characteristics of the ABWR Balance of Plant" I have concluded that there are no particular concerns from a structural integrity perspective provided the BOP remains at Safety Class 3 as the integrity claims will be low.

178. This will need to be confirmed and I have therefore identified an item for specific follow up in Step 3 once the fault schedule has been fully developed:

- Sufficiency of low integrity claims for the Balance of Plant safety case i.e. the reactor circuit downstream of the Main Steam Isolation Valves

4.9 Out of Scope Items

179. The RP has provided a document outlining the general approach that will be used for the Pre-Service Inspection (PSI) and In-Service Inspection (ISI) of the UK ABWR – "Summary of the PSI and ISI Plan for ABWR" (Ref. 14).

180. It gives a useful overview of the purpose of PSI and ISI, and that the approach will be based on the ASME Boiler and Pressure Vessel Code Section XI. It explains that the standard ISI programme will be supplemented by a programme specifically aimed at areas potentially susceptible to SCC or IASCC. However, whilst it provides a general commentary on the approaches, it provides no detail on what is being proposed.

181. This was discussed with the RP and no further detail could be provided as they were engaged in discussion with the prospective Licensee for the UK ABWR to decide what extent ISI would be included within the scope of GDA. These discussions would be concluded by the start of Step 3.

182. I have therefore left assessment of the PSI and ISI approach outside of the scope of my Step 2 assessment and will return to the subject in Step 3 once the RP has concluded their discussions. It should be noted that pervious GDAs have restricted themselves to consideration of the designs in terms of demonstrating accessibility to undertake ISI, and leaving the actual ISI programme to the Licensee. This could also occur in the case of the UK ABWR, but I would also need to have an understanding of the ISI programme aimed at areas potentially susceptible to SCC or IASCC in GDA as that will be an integral part of the justification related to potential degradation mechanisms.

183. It should be noted that leaving the assessment of the PSI and ISI approach outside the scope of the Step 2 assessment does not invalidate the conclusions from my GDA Step 2 assessment. This is because I consider the design of the major components to be of a generally conventional nature, and it should therefore be possible to demonstrate that a suitable PSI and ISI approach can be developed for the UK ABWR. I will capture the need to address this subject within my GDA Step 3 Assessment Plan.

184. In addition Section 3.2 notes five documents that were submitted during the later stages of Step 2 (Refs 30, 31, 32, 33 and 34). These five documents, although submitted during Step 2, were not intended to form part of my Step 2 assessment and they will be taken into account during the Step 3 assessment.

4.10 Comparison with Standards, Guidance and Relevant Good Practice

185. In Section 2.2 above I have listed the standards and criteria I have used during my GDA Step 2 assessment of the UK ABWR Structural Integrity to judge the adequacy of the preliminary safety cased. My overall conclusions in this regard can be summarised as follows:

- SAPs: The approach proposed by the RP on structural integrity appears consistent with ONR's expectations as identified the relevant SAPs. In particular the RP is proposing an approach to identify and justify the higher reliability components in line with the expectations of EMC.1 to EMC.3

- TAGs: The approach proposed by the RP is consistent with the TAG on the Integrity of Metal Components and Structures.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

186. Hitachi-GE has provided a PSR for the UK ABWR for assessment by ONR during Step 2 of GDA. The PSR together with its supporting references present the RP's preliminary safety case and associated claims in the area of Structural Integrity for the UK ABWR.

187. During Step 2 of GDA I have conducted an assessment of the parts of the PSR and its references that are relevant to the area of Structural Integrity against the expectations of the SAPs and TAGs. From the UK ABWR assessment done so far I conclude the following:

- the RP has adopted an approach to Structural Integrity classification that identifies the integrity claims needed to support the overall safety case
- the RP has adopted an approach to systematically identifying those components requiring a claim that the likelihood of gross failure is so low that it can be discounted
- the beyond design code compliance justification proposed by the RP using an avoidance of fracture demonstration for the highest reliability components appears consistent with ONR's expectations
- a multi-faceted approach is being taken to mitigate the threat from the stress corrosion cracking degradation mechanism
- the design summaries show that the main components of the reactor are generally of a conventional nature which gives confidence that their integrity claims will be justifiable
- I have not identified any important shortcomings, but I have raised four Regulatory Observations to aid the RP in meeting regulatory expectations during Step 3 and Step 4 of GDA:

    - RO-ABWR-0001 – Avoidance of Fracture – Margins based on the size of Crack-Like Defects
    - RO-ABWR-0002 – CRD Penetration Design
    - RO-ABWR-0003 – RPV Design (use of forgings and plate materials)
    - RO-ABWR-0004 – Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary

- I have found the RP to be receptive to ONR's approach and accepting of the need to provide beyond design code compliance justifications for the highest reliability components in line with ONR's expectations. The RP appears to have been well resourced, has consistently delivered good quality documentation to the agreed programme, and has made good use of UK contractors to provide specialist advice

188. Overall, I see no reason, on Structural Integrity grounds, why the UK ABWR should not proceed to Step 3 of the GDA process.

5.2 Recommendations

189. My recommendations are as follows.

- Recommendation 1: The UK ABWR should proceed to Step 3 of the GDA process.

- Recommendation 2: All the items identified in Step 2 as specifically noted for followed up should be included in ONR's GDA Step 3 Assessment Plan for the UK ABWR Structural Integrity. These are:

  - Sufficiency of low integrity claims for the Balance of Plant safety case ie the reactor circuit downstream of the Main Steam Isolation Valves
  - Provision of a material selection justification taking into account UK ABWR specific water chemistry
  - Optimised material choice for the Reactor Water Clean Up System
  - Inclusion of the potential for chloride ingress, including protection measures and consequences, in the safety case

- Recommendation 3: All the relevant out-of-scope items identified in sub-section 4.9 of this report should be included in ONR's GDA Step 3 Assessment Plan for the UK ABWR Structural Integrity.

6 REFERENCES

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2 *Safety Assessment Principles for Nuclear Facilities.* 2006 Edition Revision 1. HSE. January 2008. www.onr.gov.uk/SAPS/index.htm.

3 *Technical Assessment Guides.*

Integrity of Metal Components and Structures. NS-TST-GD-016 Revision 4. March 2013 www.onr.org.uk/operational/tech_asst_guides/index.htm.

4 *Not used*.

5 *Not used.*

6 *Generic Design Assessment of HGNE's Advanced Boiling Water Reactor (ABWR) – Step 2 Assessment Plan for Structural Integrity* ONR-GDA-AP-13-015 Revision 1**.** ONR December 2013. TRIM Ref 2013/412187

7 *Not used.*

8 *Hitach-GE UK ABWR – Schedule of Regulatory Queries raised during Step 2.* ONR TRIM Ref. 2014/271889

9 *UK ABWR GA91-9901-0005-00001 – XE-GD-0113 – Rev C* – Preliminary Safety Report on Structural Integrity – 1 April 2014. TRIM Ref. 2014/134314

10 *UK ABWR GA91-9201-0003-00035 – RE-GD-2010 – Rev 0* – Summary of the Design of the Reactor Pressure Vessel for UK ABWR – 17 March 2014. TRIM Ref. 2014/111877

11 *UK ABWR GA91-9201-0003-00036 – PD-GD-0010 – Rev 0* – Summary of the Design of Main Steam Piping for UK ABWR – 17 March 2014. TRIM Ref. 2014/111865

12 *UK ABWR GA91-9201-0003-00037 – PD-GD-0012 – Rev 0* – Summary of the Design of Feedwater Piping for UK ABWR – 17 March 2014. TRIM Ref. 2014/111868

13 *UK ABWR GA91-9201-0003-00038 – PVD-GD-0004 – Rev 0* – Summary of the Design of Main Steam Isolation Valve for UK ABWR – 17 March 2014. TRIM Ref. 2014/111873

14 *UK ABWR GA91-9201-0003-00039 – PD-GD-0009 – Rev 0* – Outline of the PSI and ISI Plan for UK ABWR – 17 March 2014. TRIM Ref. 2014/111095

15 *UK ABWR GA11-1001-0003-00001 – 1D-GD-0003 – Rev 0* – UK ABWR – Approach for the Avoidance of SCC – 17 March 2014. TRIM Ref. 2014/111859

16 *UK ABWR GA91-9201-0003-00054 – RD-GD-0001 – Rev 0* – Structural Integrity Classification Procedure – 7 April 2014. TRIM Ref. 2014/141719

17 *UK ABWR GA91-9201-0003-00055 – RD-GD-0002 – Rev 0 –* Weld Ranking Procedure – 7 April 2014. TRIM Ref. 2014/141736

18 *UK ABWR GA91-9201-0003-00056 – RD-GD-0003 – Rev 0 –* Defect Tolerance Assessment Plan – 7 April 2014. TRIM Ref. 2014/141769

19 *UK ABWR GA91-9201-0003-00058 – G-TY-53081 – Rev 0 –* Inspection Assessment Plan – 7 April 2014. TRIM Ref. 2014/141811

20 *UK ABWR GA91-9201-0003-00057 – G-TY-53082 – Rev 0 –* Inspection Qualification Strategy – 7 April 2014. TRIM Ref. 2014/141802

21 *UK ABWR GA91-9201-0003-00125 – SBE-GD-0019 –* Summary of the Design of BOP Components for UKABWR – 30 May 2014. TRIM Ref. 2014/221242

22 *UK ABWR GA10-9101-0100-05005 – RE-GD-2027 – Generic PCSR Sub-chapter 5.5 : Structural Integrity* – 28 May 2014. TRIM Ref. 2014/209664 (Advance copy of the PCSR)

23 The Demonstration of Incredibility of Failure in Structural Integrity Safety Cases. R Bullough, F M Burdekin, O J V Chapman, V R Green, D P G Lidbury, J N Swingler, R Wilson. International Journal of Pressure Vessels and Piping 78, pages 539-552, 2001.

24 *ONR-GDA-AR-14-002 – Revision 0 –* GDA Step 2 Assessment of the Internal Hazards of Hitachi GE's UK Advanced Boiling Water Reactor (UK ABWR). TRIM Ref. 2014/180666

25 *ONR Internal Report* – UK ABWR Material Degradation Mechanisms – 9 May 2014. TRIM Ref. 2014/180589

26 *Hitachi-GE UK ABWR – Schedule of Regulatory Observations raised during Step 2.* ONR TRIM Ref. 2014/271901

27 R6 – Assessment of the Integrity of Structures Containing Defects, Revision 4. EDF Energy Nuclear generation Ltd.

28 European Methodology for Qualification of Non-Destructive Testing. Third Issue. ENIQ Report No. 31 EUR 22906 EN. August 2007.

29 *ONR-GDA-AR-14-009 – Revision 0 –* GDA Step 2 Assessment of the Reactor Chemistry of Hitachi GE's UK Advanced Boiling Water Reactor (UK ABWR). TRIM Ref. 2014/180691

30 *UK ABWR GA91-9201-0003-00088 – JE-GD-0027 –* Structural Integrity Supporting Report – Load Combinations for Systems and Components – 30 May 2014. TRIM Ref. 2014/209586

31 *UK ABWR GA91-9201-0003-00111 – JE-GD-0028 –* Structural Integrity Supporting Report – Seismic Design for Systems and Components – 30 May 2014. TRIM Ref. 2014/209597

32 *UK ABWR GA91-9201-0003-00011 – RD-GD-0005 –* Structural Integrity Classification Report – 30 June 2014. TRIM Ref. 2014/249720

33 *UK ABWR GA91-9201-0003-00073 – RD-GD-0006* – Weld Ranking Application Report– 30 June 2014. TRIM Ref. 2014/246657

34 *UK ABWR GA91-9201-0003-00012 – RD-GD-0040 –Proposed Topic Report Structure for the component relating to Structural Integrity*– 30 June 2014. TRIM Ref. 2014/246695

Table 1

Relevant Safety Assessment Principles Considered During the Assessment

SAP No	Title	Description	Comment
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.	Considered in Sections: 4.1.1; 4.2.1; 4.3.1; 4.3.2; 4.4.1; 4.4.2; 4.5.1
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.	Considered in Sections: 4.1.1; 4.2.1; 4.3.1; 4.3.2; 4.4.1; 4.4.2; 4.5.1
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.	Considered in Sections: 4.1.1; 4.2.1; 4.3.1; 4.3.2; 4.4.1; 4.4.2; 4.5,1
EMC.4	Integrity of metal components and structures: general. Procedural control	Design, manufacture and installation activities should be subject to procedural control.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.5
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;

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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5
EMC.8	Integrity of metal components and structures: design. Requirements for examination	Geometry and access arrangements should have regard to the requirements for examination.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.5; 4.7.2
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3.1; 4.3.2; 4.5
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3.1; 4.3.2; 4.5
EMC.11	Integrity of metal components and structures: design. Failure modes	Failure modes should be gradual and predictable.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
EMC.13	Integrity of metal components and structures: manufacture and installation.	Materials employed in manufacture and installation should be shown to be suitable for the purpose of	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.5; 4.6

	Materials	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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5; 4.9
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;

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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Sections: 4.3; 4.4; 4.5;
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.	Considered in general terms in Section 4, and in particular in Section 4.6
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	Considered in general terms in Section 4, and in particular in Section 4.6

		and degradation processes on structures, systems and components that are important to safety.	
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.	Considered in general terms in Section 4, and in particular in Section 4.6
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.	Considered in general terms in Section 4, and in particular in Section 4.6
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.	Considered in Sections: 4.4.1; 4.5.1; 4.8.1
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.	Considered in Sections: 4.4.1; 4.5.1; 4.8.1
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.	Considered in Sections: 4.4.1; 4.5.1; 4.8.1