Office for Nuclear Regulation

New Reactors Division

Step 4 Assessment of Structural Integrity for the UK Advanced Boiling Water Reactor

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EXECUTIVE SUMMARY

Hitachi-GE Nuclear Energy Ltd is the designer and GDA Requesting Party for the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Hitachi-GE commenced Generic Design Assessment (GDA) in 2013 and completed Step 4 in 2017.

This assessment report is my Step 4 assessment of the Hitachi-GE UK ABWR reactor design in the area of Structural Integrity.

The scope of the Step 4 assessment is to review the safety, security and environmental aspects of the UK ABWR in greater detail, by examining the evidence, supporting the claims and arguments made in the safety documentation, building on the assessments already carried out for Step 3. In addition I have provided a judgement on the adequacy of the Structural Integrity information contained within the Pre-Construction Safety Report (PCSR) and supporting documentation.

My assessment conclusions are:

- I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for Structural Integrity.
- I consider that, from a Structural Integrity view-point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits being secured.

My judgement is based upon the following factors:

- The identification of the scope of the Structural Integrity boundary is adequate;
- The Structural Integrity classification process for components within the ABWR is adequate and applied consistently by Hitachi-GE;
- Suitable and sufficient examples have been provided for an adequate range of components covering all component types and safety classifications;
- An adequate safety case has been presented in detail for components of the highest reliability;
- ONR expectations in terms of demonstration of avoidance of fracture have been addressed directly and reconciled explicitly with the expected inspection capabilities;
- Materials choices for the ABWR are adequate, with due attention paid by Hitachi-GE to components of the highest reliability;
- The regions of the reinforced concrete containment vessel sampled are adequate for purpose and improvements have been made where appropriate;
- An adequate Structural Integrity case has been presented for the spent fuel storage canisters, acknowledging that these will not be deployed in the near term;
- Examples of lower-classification components have been provided and judged to be adequate;
- An adequate structure has been presented for the PCSR, which is capable of supporting licensing of the UK ABWR.

The following matters remain, which are for a future licensee to consider and take forward in their site-specific safety submissions. These matters do not undermine the generic safety submission but require licensee input/decision at a specific site.

There are a total of 21 Assessment findings, and three minor shortfalls. Generically, these fall into the area of defect tolerance, materials properties & product form, codes & standards, inspection, fabrication, fatigue and classification.

Overall, based on the samples undertaken, I am broadly satisfied that the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of

the GDA process present an adequate safety case for the generic UK ABWR design in the area of Structural Integrity. For this reason, the UK ABWR should be awarded a DAC at this present time.

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 IX	ASME Boiler and Pressure Vessel Code Section IX
ASME VIII	ASME Boiler and Pressure Vessel Code Section VIII
ASME XI	ASME Boiler and Pressure Vessel Code Section XI
BDL	Bottom Drain Line
BMS	Business Management System
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor (generically)
CRDM	Control Rod Drive Motor
CRD	Control Rod Drive
CUW	Clean Up Water
D/W	Drywell
DAC	Design Acceptance Confirmation
DEPSS	Drywell Equipment Pipe Support Structure
DMW	Dissimilar Metal Weld
DSM	Defect Size Margin
DTA	Defect Tolerance Assessment
DMW	Dissimilar Metal Weld
EA	Environment Agency
EAC	Environmentally Assisted Cracking
ELLDS	End of Life Limiting Defect Size
ENIQ	European Network for Inspection Qualification
EPDM	Ethylene Propylene Diene Monomer rubber
EQ	Equipment Qualification
FAC	Flow-Assisted Corrosion
FCG	Fatigue Crack Growth
FDW	Feedwater
FIV	Flow-Induced Vibration
FNC	Frazer Nash Consultancy
FSF	Fundamental Safety Function
FUF	Fatigue Usage Factor

GDA	Generic Design Assessment
GTAW	Gas-Tungsten Arc Welding
HAZ	Heat Affected Zone
HCU	Hydraulic Control Unit
HI	High Integrity
Hitachi-GE	Hitachi-GE Nuclear Energy Ltd
HLSF	High-Level Safety Function
HP	High Pressure
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICGT	In-Core Guide Tube
ICM	In-Core Monitoring
IVB	Independent Verification Body
IVC	Inspection Validation Centre
JEAC	Japan Electric Association Code
JSME	Japanese Society of Mechanical Engineers
JSW	The Japan Steel Works company
LFCG	Lifetime Fatigue Crack Growth
LOCA	Loss of Coolant Accident
LOSF	Lack of Sidewall Fusion
MDEP	Multi-national Design Evaluation Programme
MSIV	Main Steam Isolation Valve
MSP	Main Steam Pipework
NBA	Nickel-Base Alloy
NDE	Non Destructive Examination
NRW	Natural Resources Wales
ONR	Office for Nuclear Regulation
OECD-NEA	Organisation for Economic Co-operation and Development – Nuclear Energy Agency
PCSR	Pre-construction Safety Report
PCV	Primary Containment Vessel
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
QEDS	Qualified Examination Defect Size
R/B	Reactor Building
RCCV	Reinforced Concrete Containment Vessel
RCIC	Reactor Core Isolation Cooling System
RGP	Relevant Good Practice

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RHR	Residual Heat Removal
RI	Regulatory Issue
RIN	Reactor Internals
RIP	Reactor Internal Pump
RIS	Radiation Induced Segregation
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RSW	Reactor Building Service Water
RWCU	Reactor Water Clean Up
SAP(s)	Safety Assessment Principle(s)
SAW	Submerged Arc Welding
SCC	Stress Corrosion Cracking
SFAIRP	So Far As Is Reasonably Practicable
SHT	Solution Heat Treatment
SIF	Stress Intensity Factor
SLC	Standby Liquid Control
SLLDS	Start of Life Limiting Defect Size
SoDA	Statement of Design Acceptability
SMAW	Shielded Metal Arc Welding
SRV	Safety Relief Valve
SS	Stainless Steel
SSC	System, Structure (and) Component
SSER	Safety, Security and Environmental Report
SZB	Sizewell B Nuclear Power Station
TAG	Technical Assessment Guide
TAGSI	UK Technical Advisory Group on Structural Integrity
TJ	Technical Justification (of Inspection)
TOFD	Time Of Flight Diffraction
TSC	Technical Support Contractor
UK	United Kingdom
US NRC	United States (of America) Nuclear Regulatory Commission
VHI	Very High Integrity
WENRA	Western European Nuclear Regulators' Association

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Annexes

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1 INTRODUCTION

1.1 Background

1. Information on the GDA process is provided in a series of documents published on our website (http://www.onr.org.uk/new-reactors/index.htm). The expected outcome is a Design Acceptance Confirmation (DAC) for ONR and a Statement of Design Acceptability (SoDA) for the Environment Agency (EA) and Natural Resources Wales (NRW).

2. The GDA Step 3 summary report is published on our website (**http://www.onr.org.uk/new-reactors/uk-abwr/reports/step3/uk-abwr-step-3-summary-report.pdf**) (Reference 5). Further information on the GDA process in general is also available on our website (**http://www.onr.org.uk/new-reactors/index.htm**).

3. Hitachi-GE commenced GDA in 2013 and completed Step 4 in 2017.The Step 4 assessment is an in-depth assessment of the safety, security and environmental evidence. Through the review of information provided to ONR, the Step 4 process should confirm that Hitachi-GE:

   - Has properly justified the higher-level claims and arguments.
   - Has progressed the resolution of issues identified during Step 3.
   - Has provided sufficient detailed assessment to allow ONR to come to a judgment of whether a DAC can be issued.

4. During the Step 4 assessment I have undertaken a detailed assessment, on a sampling basis of the safety and security case evidence. The full range of items that might form part of the assessment is provided in ONR's GDA Guidance to Requesting Parties (http://www.onr.org.uk/new-reactors/ngn03.pdf). These include:

   - Consideration of issues identified in Step 3.
   - Judging the design against the Safety Assessment Principles (SAPs) and whether the proposed design reduces risks SFAIRP.
   - Reviewing details of the Hitachi-GE design controls, procurement and quality control arrangements to secure compliance with the design intent.
   - Establishing whether the system performance, safety classification, and reliability requirements are substantiated by the detailed engineering design.
   - Assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final as-built design.
   - Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

5. This is my report from the ONR's Step 4 assessment of the Hitachi-GE UK ABWR design in the area of Structural Integrity.

6. All of the regulatory observations (ROs) issued to Hitachi-GE as part of my assessment are also published on our website, together with the corresponding Hitachi-GE resolution plan.

7. This report provides the summary of the work done in Step 4, and preceding steps, of the GDA of the UK ABWR. This includes an overview of the structure and scope of the work performed by Hitachi-GE, and how this has been judged against UK expectations in terms of demonstration of a Structural Integrity Safety Case. This has been performed utilising the ONR Regulatory Observation (RO) process to address areas of specific interest and specific UK expectations, Regulatory Queries (RQs) where clarification has been required, and through assessment of the overall submissions made by Hitachi-GE.

8. Judgements have been made in line with the expectations laid out in the ONR SAPs. My interactions with Hitachi-GE have been focussed on ensuring that these expectations are met.

1.2 Scope

9. The scope of my assessment is detailed in assessment plan **(**ONR-GDA-AP-15-015 - Step 4 Assessment Plan - Structural Integrity - Gareth Hopkin - 03 November 2015. TRIM 2015/411182) (Reference 1).

10. The scope of my assessment covered all areas of the main pressure boundary, that is the reactor coolant circuit. I have also included assessment of other metallic containment, including aspects of the Primary Containment Vessel, the reinforced concrete Containment Vessel (RCCV), and the interim dry fuel storage containment vessel.

11. The scope of my assessment is appropriate for GDA because it captures the most salient points for the design which need to be confirmed before any licensing activities can take place. The nuclear safety significance has been key in informing the depth of my assessments; factors affecting safety will be addressed proportionate to the nuclear safety significance. I have also focussed my assessment to technical areas that are novel within the UK context, or where UK experience is low. Furthermore, sampling of lower safety classification documents, at an appropriate level ensures coverage of all areas across the plant.

12. Systems, structures and components within the scope of the Structural Integrity are, primarily, the pressure boundary components including the RPV through to the main turbine casing, including all supporting pipework. Moreover, the reactor internal components, not including the fuel and core, are included within the scope of my assessment.

13. Metallic pressure boundary components of the Reinforced Concrete Containment Vessel are within the scope of this assessment, but not the connections of these components with the surrounding concrete structures. Also included in my assessment is the interim spent fuel storage canister. Details of this system are included at a level appropriate for GDA.

14. Further detail on which sections were available in the sampling strategy section of this report, below.

1.3 Method

15. My assessment complies with internal guidance on the mechanics of assessment within ONR:

- Guidance on demonstration of ALARP (Reference 2).
- Guidance on production of reports (Reference 3).
- Peer review of legal and technical assurance (Reference 4).

2 ASSESSMENT STRATEGY

2.1 Standards and criteria

16. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Reference 32), internal TAGs (Reference 163), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites.

2.1.1 Safety Assessment Principles

17. The key SAPs applied within the assessment are included within annex 1

2.1.2 Technical Assessment Guides

18. The TAGs that have been used as part of this assessment are set out in annex 2

2.1.3 National and international standards and guidance

19. The international standards and guidance that have been used as part of this assessment are set out in annex 3

2.2 Use of Technical Support Contractors (TSCs)

20. It is usual in GDA for ONR to use TSCs, for example to provide additional capacity, to enable access to independent advice and experience, analysis techniques and models, and to enable ONR's inspectors to focus on regulatory decision making etc.

21. Table 1 sets out the broad areas where technical support was used. This support was required to provide extra resource in Step 4, where the quantity of documentation supplied by Hitachi-GE was very high. Moreover, I considered it appropriate to bring in support in specific areas where ONR expertise was not available. This brought benefits in that a full range of technical areas could be sampled, and ONR resource could be deployed to where it is most effective.

Work-package 1	Review of Compliance with Design Requirements of ASME III
Work-package 2	Review of Defect Tolerance Assessments, including Comparative Calculations
Work-package 3	Review of Selected Topic Reports
Work-package 4	Review of Pipewhip Methodology and Calculation
Work-package 5	Review of Selected Design Reports

Table 1: Work-package areas where TSC resource has been utilised

2.3 Integration with other assessment topics

22. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. The following cross-cutting issues have been considered within this assessment:

- Materials Selection Joint RO issued with Reactor Chemistry. This is an issue that covers the pressure boundary integrity, but also Reactor water chemistry, how it is used to provide assurance to pressure boundary integrity but also how it maintains appropriate worker dose mitigation.

- Bottom Drain Line RO authored by Reactor chemistry but having a key input and pertinence to Structural Integrity. The bottom drain line is a Class 1 component and hence of a higher nuclear safety significance. Moreover, the bottom drain line is an example component of a bottom head penetration.

- The Reinforced Concrete Containment Vessel (RCCV) has been assessed jointly with Civil Engineering. This structure provides containment in the event of a failure of the reactor coolant pressure boundary. This is a concrete structure with a metallic liner and metallic hatches. The interventions were led by Civil Engineering, with appropriate support from structural integrity.

- The issue of pipewhip has been led by Internal Hazards with appropriate support from structural integrity. Structural integrity interest lies with the reliance on pipewhip to inform the structural integrity classification of the component.

- Mechanical Engineering and structural integrity have worked together in the area of Equipment Qualification (EQ). EQ components have formed a route to demonstrating adequate safety and design within mechanical engineering, as well as informing the materials data available for defect tolerance analysis.

2.4 Sampling strategy

23. It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific, or generic, weaknesses in the safety case.

24. I have focussed on areas of highest nuclear safety risk and areas where an approach novel to the UK has been put forward by Hitachi-GE. I assessed which components were at the highest levels of nuclear safety risk through reference to the Hitachi-GE Structural Integrity classification process and the reviews performed by myself and the ONR technical support contractor. I assessed which areas of the design were novel by comparison with previous UK nuclear operating plant experience. The ABWR being a boiling water reactor, with a single circuit, large RPV and numerous bottom penetrations in that RPV, the level of novelty to the UK was larger than it would have been for a PWR design.

25. The components pertinent to the Structural Integrity discipline are those that provide the pressure boundary and metallic components within the pressure boundary. The pressure boundary of a generic light-water reactor includes the reactor pressure vessel and all the associated pipework and secondary vessels to which it is connected. For a Boiling Water Reactor (BWR), such as the UK ABWR, this includes components in the turbine building, including the main turbines and pipework associated therewith. Hence, I have sampled components along the main pressure boundary from the RPV to the turbines inclusive, with the level of sampling being proportionate to the nuclear safety function of the component in question.

26. In addition to the main pressure boundary, I have considered other pressure systems that might be called upon during service or accident conditions to also be within scope. This has included the metallic components of the main containment vessel, the Reinforced Concrete Containment Vessel (RCCV), excluding those metallic

components which do not have a pressure boundary function. Also included are the pressure boundary parts of the proposed interim spent fuel storage system.

27. Also included within the scope of the Structural Integrity assessment were the reactor internal components (RINs). Whilst these do not have a pressure boundary function, they are metallic components that maintain the geometry of the core.

28. The sampling strategy for this assessment was to ensure that all representative components covering the whole of the pressure boundary were covered, and that the nuclear safety significance of each submission was understood. This was achieved through reference to the Structural Integrity Classification documentation. I sampled all components having the highest safety significance, and hence classification: these were termed VHI (Very High Integrity) by the requesting party and consisted of the RPV, MS piping inside the RCCV and inboard MSIVs. The level of sampling of lower classification components decreased according to the classification of the component involved. This ran from Class 1 components, where in-depth analysis was performed on an extensive number of components, to Class 3 where a single representative component was sampled at a proportionate level. I sampled to ensure that, for each representative component assessed, the claims, arguments and evidence were laid out in sufficient detail so as to provide a suitable safety demonstration for the purposes of GDA.

29. Assessment of non-pressure boundary components, such as the Reactor Internals (RINs) that perform the function of maintaining the core geometry, was also made in proportion to their nuclear safety function and classification.

30. A sampling approach was taken to the RCCV components and credit taken where confidence could be demonstrated in Hitachi-GE's approach to design of these components.

31. The spent fuel interim storage was sampled in so far as it interacts with the civil structures. Sampling was performed sufficient to gain confidence that the outline solution proposed is capable of meeting reliability targets.

2.5 Out of scope items

32. All aspects of the pressure boundary are considered to be within scope, as is the notion of design for inspectability; this means that GDA is expected to provide assurance that the design facilitates pre-service and in-service inspections. This notwithstanding, items including in-service inspection and detailed record keeping are not included.

33. Connections of the metallic liner of the RCCV to the concrete structure are part of the Civil Engineering assessment (Reference 165) and, hence, excluded from the Structural Integrity assessment.

34. Detailed design of the spent fuel interim storage was excluded from the assessment, as was outline design in so far as it affected the performance of the canister outside of the reactor building, that is, whilst in interim storage off the nuclear island.

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3 REQUESTING PARTY'S SAFETY CASE

35. The Hitachi-GE safety case for Structural Integrity has followed a structure as documented in the Structural Integrity Topic Report Preparation plan (Reference 6). This gives an overview of how Hitachi-GE structured their submissions, in order for them to attempt to meet ONR's expectations. This document was delivered very late in Step 4 of GDA. This notwithstanding, the content of report had been presented informally at Level 4 meetings over the course of several years.

36. The top level document in the Structural Integrity Safety Case from Hitachi-GE is Chapter 8 of the PCSR. Revision C of the PCSR has been reviewed for this report. Revision C of chapter 8 of the PCSR has been developed significantly in comparison with the previous revisions. This sets out the high level scope of the Structural Integrity Safety Case, and sets the following objectives:

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

37. The PCSR also sets out, at a high level, the safety functional claims made, as well as the Structural Integrity classifications used within the Safety Case and how these link to nuclear safety significance.

38. The nuclear safety classifications used by Hitachi-GE in the Structural Integrity documentation are not synonymous with ASME class, but the mechanical requirements of the component. The safety classification, the levels of reliability inferred from this and how this links through to the Structural Integrity classification is discussed in the Structural Integrity classification report (Reference 7) with safety classification having been discussed within the Fault Studies discipline Step 4 report (Reference 166). Structural Integrity classifications, and their link to nuclear safety consequences, were presented in the PCSR thus:

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

Table 2: Structural Integrity Classification scheme used by Hitachi-GE.

39. VHI and HI classifications are abbreviations for Very High Integrity and High Integrity, respectively. These are described by Hitachi-GE as subcategories of safety class 1 plant, introduced to meet UK expectations.

40. A high-level overall ALARP justification has been provided as part of the PCSR. Hitachi-GE state that this is in line with chapter 28 of the PCSR, which treats ALARP generically across all discipline areas.

41. The safety functional claims, in terms of Structural Integrity, are presented in a “claims tree” format and linked back to the Fundamental Safety Functions (FSFs) and High Level Safety Functions (HLSFs) through chapter 5.4 of the PCSR. The only FSFs referenced are “Containment of radioactive materials”, and “Control of reactivity”. These are listed against the VHI & standard Class 1 components. The only two HLSFs listed are “Functions to form reactor coolant pressure boundary” and “Functions to maintain core geometry”. Class 2 and Class 3 components are not referenced in the claims tree.

42. The main pressure boundary components were classified by Hitachi-GE according to the nuclear safety significance of the component. This was reported in the Structural Integrity classification document (Reference 7). The classification document was produced initially within Step 2 of GDA, but was continually updated throughout the GDA process. This is a document underpinning the focus of attention from Hitachi-GE onto the areas of greatest nuclear safety significance.

43. Reference 6 describes the plan to provide Topic Reports of representative SSCs for Structural Integrity (SI) assessment. Hitachi-GE considered that it was necessary to prepare representative topic reports which cover all classes of components, from Very High Integrity (VHI), down to Class 3 components. Hitachi-GE did not consider components at classifications below standard Class 3.

44. Hitachi-GE presented information on only a selection of components, the total number increasing in relation to the safety function of the component. The coverage for components of the highest reliability, termed VHI by Hitachi-GE, was 100%, topic reports were produced for selected regions of Class 1, and single example reports were produced for Classes 2 and 3.

45. The components selected were chosen to be representative of the different component types making up the main pressure boundary of the UK ABWR. The components considered are shown in Table 3, along with a demonstration of the rational for the choice of example components.

Safety Class	Vessel	Heat Exchanger	Piping	Valve	Pump	Others
VHI	RPV	N/A	MS	MSIV	N/A	N/A
Standard Class 1	HCU†	RHR†	FDW	SRV†	RCIC/RSW†	RIN
Class 2	SLC Storage Tank	Covered by Class 1 Vessel	Covered by Class 1 Piping	Covered by Class 1 Valve	Covered by Class 1 Pump	N/A
Class 3	HP Turbine Casing	N/A	N/A	N/A	N/A	N/A

Table 3: Components considered by Hitachi-GE within the Structural Integrity assessment of the UK ABWR.

46. Each of the selected components were covered by a topic report or equivalent. These topic reports together are proposed to form the basis of Structural Integrity confidence in the claims put forward. The claims put forward for each of the selected components were based around the same structure. This structure was exemplified by Hitachi-GE via the table of contents for the RPV Topic Report. Certain components, marked with

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a † above, were reports produced jointly with the Mechanical Engineering discipline as Hitachi-GE judged that there was a significant overlap.

47. This table of contents is as follows:

48. This structure was also followed for lower classification components, but the level of demonstration was reduced commensurate with the nuclear safety significance of the documents. The topic reports, and equivalent documents, are intended to demonstrate the claims and present the arguments behind those claims.

49. The claims put forward by Hitachi-GE for highest reliability components are as per in section 6 of the table of contents above. Hitachi-GE stated that this has been based upon the TAGSI (UK Technical Advisory Group on Structural Integrity) 4-legged approach. The approach has been expanded to 5 nominally independent legs. For Class 1 components and lower, the original TAGSI 4-legged structure has been adopted.

50. The topic reports have a number of subordinate documents which Hitachi-GE have provided to contain the evidence to support the claims and arguments presented in the topic reports. Hitachi-GE presented a document structure as part of Reference 6. This is reproduced at Figure 3 below:

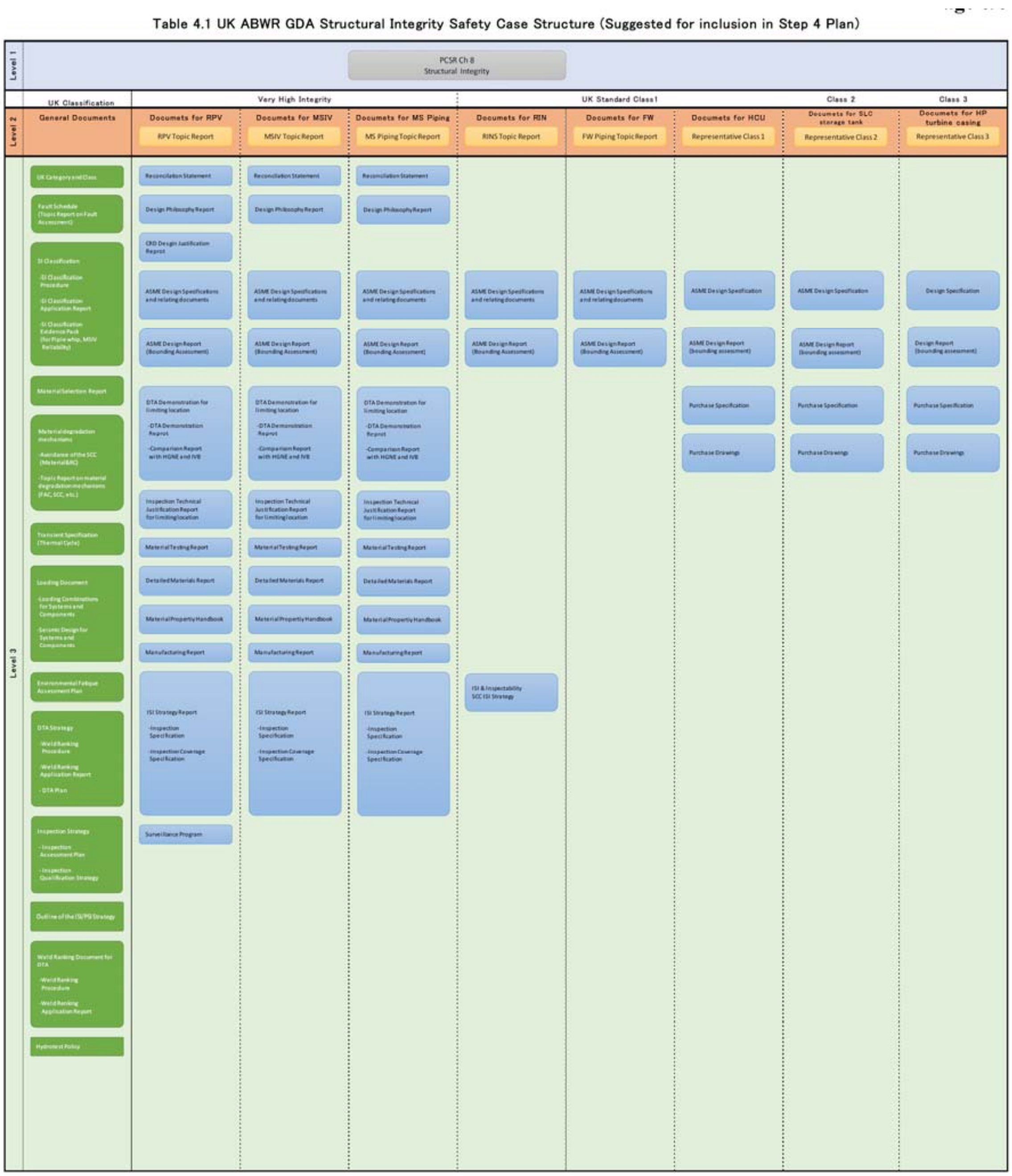

Figure 1. Safety Case Structure, reproduced from Reference 6

51. Details of specific items covered by Regulatory Observations (ROs) were covered as part of the supporting documentation for the topic reports.

4 ONR STEP 4 ASSESSMENT

52. This assessment has been carried out in accordance with ONR internal guidance on the "Purpose and Scope of Permissioning" (Reference 164).

4.1 Scope of Assessment Undertaken

53. The scope and sampling strategy are defined in section 1.2 and 2.4 respectively.

54. The scope of the assessment, in general, covers the PCSR and all of the documents subordinate thereto. Pressure boundary components, including primary, secondary and spent fuel storage pressure boundaries, have been considered, as well as the reactors internals. I have sought to get assurance that the Structural Integrity safety functions can be made within the Structural Integrity discipline, noting that items related to the procurement, or requiring information from the procurement phase, are appropriate to be left until the licensing phase.

55. I have made judgements based upon the adequacy of the documentation to demonstrate that it is suitable for GDA purposes. By this, I mean that suitable information has been presented to give confidence that the UK ABWR can be deployed within the UK and that a Structural Integrity safety case can be made, if appropriate supporting information and data is made available within the licensing phase.

4.2 Assessment

56. My assessment has mirrored the structure set out by Hitachi-GE in their safety submissions, as outlined in section 3, above. I judge that the methodology chosen by Hitachi-GE to ensure that all different classes and types of component are sampled within the Step 4 safety submissions is suitable for GDA purposes.

57. The structure proposed, in my opinion, provides claims which are then reiterated and expounded through Level 2 documentation and evidenced through documentation subordinate to these level 2 documents. I consider this to be a clear and well-structured approach to providing the assurances within GDA. I note, however, that further information will be needed to cover all aspects of the plant within site licensing. I consider that a complete and adequate safety case, covering all Structural Integrity components, should be provided by the licensee in the site licensing phase. This will be further developed as part of the discussions on RO-ABWR-0001.

58. In sentencing comments against documents, my technical support contractor (FNC) has provided an indication of the level of importance of each comment. These are:

- Category 1: Significant non-compliance, significant non-conservatism, significant safety issue, that would jeopardise the conclusions from any assessment.
- Category 2: Assessment routes or techniques not meeting good practice.
- Category 3: Major comments and points of clarification that could become Category 1 or 2 comments.
- Category 4: Minor comments and points of clarification.

59. I have reviewed each of these comments before submitting them to Hitachi-GE via the RQ process. To complete GDA, it was my expectation that all Category 1 and 2 comments need be fully addressed. Category 3 comments require a response within GDA and failure to close these comments may lead to Assessment Findings or, in

extremis, failure to close GDA. Category 4 comments do not prevent closure of GDA and are unlikely to lead to assessment findings, but should be noted and carried forward by the licensee as part of normal business.

4.2.1 Chapter 8 of the Hitachi-GE PCSR

60. The top level document for the Hitachi-GE Structural Integrity Safety Case is Chapter 8 of the PCSR (Reference 8). I have reviewed Revision C of the PCSR as part of my Step 4 assessment. The document is at a very high level and provides a signposting to subordinate documentation where the major part of the Safety Case resides. Links are made to the other relevant sections of the Safety Case through reference to other chapters in the PCSR.

61. The linkage to Safety Functional Claims within Chapter 8 of the PCSR, is done through reference to only two SFCs and only two HLSFs. This appears appropriate given the very high level of presentation in the PCSR. However, I note that the safety functional claims are only presented for Class 1 components and higher. It is not clear to me, nor is it explained within the PCSR, why the Class 2 and Class 3 components are excluded from consideration for SFCs. Given that the nuclear safety significance of these lower classification components is lower, I judge that this is acceptable for GDA purposes, but I consider it appropriate to raise the following Assessment Finding:

AF-ABWR-SI-01 The safety functions of all classes of component are not defined and collated coherently in the GDA submissions. To properly explain the safety functions of all classes of component, the licensee shall link the safety demonstration of components of lower Structural Integrity classification to the safety functions they perform explicitly.

62. Chapter 8 of the PCSR goes on to provide only a high level review of the key (Level 2) safety submissions. The Structural Integrity Classification document, as well as the representative topic reports are summarised in appendices to the chapter 8. This places the onus on the subordinate documentation. I have, therefore, focussed my review on the subordinate documentation, especially the topic reports for the representative components which are intended to demonstrate that the Safety Case can be made.

63. The PCSR states that the plant is to be designed to the “ASME B&PV” code, which I interpret to mean the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code. This code is updated on a regular basis and it is not specified within the PCSR which edition is being used, and with what addenda. Whilst editions and addenda are specified within the subordinate documentation, the absence of these specifics is a possible source of error. It is my expectation that the licensee should control the edition of the code used through overarching safety documentation, such as the PCSR. Before procurement, it is also my expectation that the licensee should review the differences between the code used and its most recent version, and assess whether it is ALARP to implement any changes made.

AF-ABWR-SI-02 In the safety documentation presented during GDA, the design code version is not controlled. The design code for the reactor plant forms the basis for good practice in design. The licensee shall institute controls over the edition of, and addenda to, the design code to be used.

64. I consider that Chapter 8 of the PCSR is adequate for GDA purposes. I recommend that, as part of normal business, the licensee should ensure the PCSR is restructured such that it is capable of demonstrating the Structural Integrity of the whole plant, and not just the areas sampled during GDA.

4.2.2 Structural Integrity Classification

65. The Structural Integrity classification of Systems, Structures and Components (SSCs) underpins the level of assessment done within the Structural Integrity discipline. From the outset of GDA Step 2 (Reference 160), it has been ONR's expectation that the SSCs pertinent to Structural Integrity should be classified, and the level of evidence produced by Hitachi-GE should be comparable directly to that classification.

66. Hitachi-GE have presented the classification of the Structural Integrity plant in the Structural Integrity Classification report, which has been revised through the course of GDA, in response to ONR comments, and ongoing assessment. The latest version is Revision 5, and was presented in January 2017. Hitachi-GE put forward five basic levels of classification for Structural Integrity components:

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

Table 4: Structural Integrity Classification scheme used by Hitachi-GE.

67. The terms VHI and HI are used to mean Very High Integrity and High Integrity respectively by Hitachi-GE.

68. The above classification scheme is in line with classification schemes used by others within the nuclear industry in the UK and consider that it is broadly acceptable. I have judged the VHI classification components against the ONR expectations, as set out in the SAPs, for a component of the highest reliability. For the purposes of this assessment, I have considered the ONR terminology of "component of the highest reliability" and Hitachi-GE's term, VHI, as directly analogous.

69. The use of a Structural Integrity classification intermediate between Standard Class 1 and VHI, denoted HI by Hitachi-GE, does align with ONR's SAPs. This notwithstanding, it is difficult to analyse what ONR's expectations would be for an HI component, and what level of additional evidence would be required above code compliance but, presumably, below a full VHI demonstration.

70. For the purposes of this GDA assessment, I have not formed a judgement on what is required within an HI Safety Case. This is appropriate because there have been no HI components proposed by Hitachi-GE during GDA. If HI components are proposed during licensing, it will be necessary for ONR and the licensee to interact to address expectations. This is unlikely to be a trivial task. ONR will monitor this as part of normal business, through the subsequent licensing and construction phases.

71. I have reviewed the Structural Integrity classification document a number of times over the course of the GDA, with the latest assessment having been made by my TSC. Following this review, an RQ was raised with comments against this in RQ-ABWR-1403 (Reference 16). Comments raised within that RQ have been addressed by Hitachi-GE with two exceptions. One of these issues was the matter of pipewhip and how it is applied to generate worst-case analyses. This is addressed in detail in the next section of this report.

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72. The second matter relates to the potential formation of missiles following the failure of pressure vessels. Specifically, it is not clear from the response to Comment 4 of RQ-ABWR-1403 that all pressure systems that could pose a missile risk to safe control of the plant had been identified. This could, potentially, affect the classification of the source vessel. It is not clear that UK relevant good practice has been followed in this case. This presents a risk to the Structural Integrity classification document, which presents an overarching risk to the Safety Case presented by Hitachi-GE. I consider it reasonable, therefore, to raise this as an assessment finding:

AF-ABWR-SI-03 During GDA, Hitachi-GE have not demonstrated that the secondary consequences of failure of lower classification plant, in terms of missile formation, is captured properly during the classification process. The licensee shall review the classification of all Structural Integrity components inside containment relating to the potential formation of missiles and their effect on adjacent systems structure and components.

73. Late in my assessment I was made aware that ONR's internal hazard specialist had questioned certain claims in the internal flooding case for the reactor building (R/B) under shutdown conditions (Reference 167). Notably, under the fault LOCA (mechanical) below Top of Active Fuel (TAF), protection against the consequences of flooding of the drywell and R/B under shutdown conditions was dependent on the closure of an access hatch to Division III via R/B room 216. Hitachi-GE claimed the risks were reduced ALARP because: the pressure boundary is at low pressure during outages; the access tunnels are normally closed unless maintenance activities are underway; and a flooding event would be identified by personnel. In addition, Hitachi-GE proposed using a slit break and leak from the BDL as a bounding case for their LOCA assessment rather than a postulated guillotine failure. Hitachi-GE also claimed a gross failure i.e. a guillotine failure could be discounted, but without recourse to the demanding evidence expected for a highest reliability claim under EMC.1 to EMC.3.

74. From a structural integrity perspective, Hitachi-GE's approach did not meet my expectations for discounting gross failure (SAP EMC. 1-3). In addition, the reliance on partial failure or limited leakage claim was inconsistent with the Hitachi-GE approach to SI classification, and meeting UK expectations. Overall, it was not clear whether Hitachi-GE's proposals were commensurate with reducing risks ALARP. In particular, meeting the UK expectation to provide engineered provision to avoid highest reliability claims wherever possible (SAP EMC.2 Paragraph 293). I discussed the position with ONR's internal hazards specialist who subsequently raised AF-ABWR-IH-15 for the licensee to consider and implement all reasonably practicable design options, including engineered measures, to ensure isolation of the flood path from the Lower D/W to the R/B in LOCA scenarios.

75. The difficulty in providing a case for flooding protection under the shutdown condition was unexpected. However, it does raise a concern with regard to the scope of the SI classification, which may have been predicated on the basis that the power operation condition will bound other plant states e.g. shutdown. I therefore raise an assessment finding for the future licensee to review other plant states and to confirm the SI classifications post GDA:

AF-ABWR-SI-04 – Late on in GDA, ONR became aware that the bounding consequences of failure, for certain components, may not be during power operation but during shutdown conditions. Because Hitachi-GE's Structural integrity classification process assumes that operation provides the bounding condition, the licensee shall review the structural integrity classification for all systems structures and components and confirm if the classification is bounding.

76. Based on the above, I am satisfied that the way forward identified is open to licensee choices and is therefore appropriate to address it via the assessment findings raised.

77. This notwithstanding, and noting that pipewhip considerations are covered separately below, I am broadly content that the Structural Integrity classification methodology and the individual classifications proposed are suitable for GDA, and consider that they provide a good basis for licensing. This has been confirmed through review of the Classification document by my TSC and in line with the assessments made to determine the levels of reliability required (Reference 166). Working with the Structural Integrity classification scheme proposed by Hitachi-GE I have addressed my assessment effort to components at the higher classifications. This is in line with ONR's principle of proportionality.

4.2.3 Pipewhip and Pressure Part Failure

78. This section of the Structural Integrity assessment has been worked on in conjunction with the ONR Internal Hazards (IH) discipline (Reference 167). These matters lie, strictly, within the IH discipline and that discipline has taken the lead role in this case. Pipewhip and pressure part failure are, however, of direct interest to Structural Integrity as they govern the Structural Integrity classification. As stated above, without a reliable classification process, it is not possible to ascertain the nuclear safety significance of the component. Hence, an in depth analysis in this area was also performed within Structural Integrity.

79. In Section 2 of Reference 9, Hitachi-GE have used the pipewhip methodology from the impact assessment procedure R3 (Reference 11). The R3 procedure (Reference 11), developed following extensive experimental work, is a widely used procedure in the UK nuclear industry to calculate impact assessments etc. following empirical equations in lieu of complicated numerical simulations. However, there are stated limitations with the R3 impact assessment methodology and it recommends bespoke numeral techniques are used in cases of significant departure from the stated limits.

80. In my opinion, for the energy based systems like pipewhip impacts, R3 (Reference 11) is recognised by ONR as a source of relevant good practice, so long as the stated limits are recognised and adhered to in the assessments.

81. In Section 3 of Reference 9, Hitachi-GE claims that the failure judgement criteria is based on various parameters like comparative bores or diameters of the impacting pipes (whipping and target), inclusion of valve on the whipping pipe etc. It also claims that if a whipping pipe impacts an SSC, the conservative assumption is, the latter would lose all its functions. I have reviewed those assumptions and in my opinion those are conservative and acceptable.

82. Following on from the R3 procedure, which has been used to calculate impacting energy", Hitachi-GE have developed an energy based criteria to arrive at the failure judgement of the impacting pipes. This has been based on experimental data presented in NUREG / CR 3231 (Reference 12); Figure 4 (Reference 9), which is equivalent to Figure 4.12 (Reference 12). This presents the comparison of a local deformation parameter and a global deflection parameter for 32 impacting pipes of different diameters and thicknesses (Schedule 40, Schedule 80 and Schedule 160). In NUREG / CR 3231 (Reference 12), the failure is judged by through-wall crack forming in the pipe following the impact, where both the impacting and target pipes were pressurised (~15.9MPa at ~288°C). In comparison, the design steam pressure for the UK ABWR is 8.62MPa and design steam temperature is 302°C. This means that the hoop / axial stresses for the NUREG samples are almost double those for the UK ABWR pipes of similar dimensions (diameter and thickness). Thus, in my opinion, following a NUREG type "impact", a similar UK ABWR pipe is less likely to fail due to lower stresses.

83. Following the NUREG test data in Figure 4.12 (Reference 12) (and Figure 4 of Reference 9), Hitachi-GE have developed a failure criteria based on the conservative (as per NUREG) boundary lines between the failure and non-failure regions of the test data (Page 72 of NUREG / CR 3231 Reference 12). That failure criteria considers the limiting values for the local deformation and global deflection from the NUREG results for the Schedule 80 pipes. This appears comparable with the main steam pipe geometry for the UK ABWR (Table 1, Reference 9, e.g., 700A Sch80), however, the NUREG also states that there is "*scatter in the impact performance*" and that "*while a boundary more complex may define the actual region, there is insufficient data to establish this boundary*". In my opinion, this is akin to a caution regarding direct application, to anyone potentially using the data from Figure 4.12 (Reference 12).

84. Hence, in Figure 4.13 the NUREG/CR3231 (Reference 12) presents the recommended damage diagram and identifies a limiting zone including a factor of √2 on the lower bound non-failure data from the Sch40 pipe tests. The revised recommended allowable local deformation / global deflection limiting parameters are ~0.2 / ~0.2 respectively. In comparison, this is significantly lower than those used as the failure criteria by Hitachi-GE in (Reference 9).

85. I agree that the NUREG's recommended damage box for Sch40 may be too conservative for the UK ABWR primary circuit related pipework inside the PCV (minimum Sch80). In my opinion, however, Hitachi-GE should have at least considered the use of a bounding box relevant for the Sch80 pipes (Figure 4.12 in Reference 12) and applied the factor of √2 to determine their failure criterion in Reference 9. I judge that such an approach would have been more demonstrably consistent with the NUREG, proportionate to the pipe size, and reasonable considering the conservatisms alluded in Section 3.3 of Reference 9.

86. I asked Hitachi-GE (Reference 13) to justify their having not explicitly following the NUREG recommended damage diagram in Reference 9, in so doing I noted that the NUREG has been claimed as a RGP in support of the pipewhip assessments. Hitachi-GE responded (Reference 13), stating that in Figure 4 (Reference 9), they have used a slightly modified version of the box from the NUREG Figure 4.12 (Reference 12) for the Sch80 pipes in lieu of the data for Sch40, which I agree, as the pipe sizes (UK ABWR MS line Sch80) are comparable.

87. Regarding the absence of the safety factor (√2), however, Hitachi-GE argued (Reference 13), based on claims that there are margins between the failed and non-failed NUREG data (Hitachi-GE's blue box in Figure 4 of Reference 9), that whilst NUREG data considered presence of through wall crack leading to leakage, Reference 9 considers guillotine failure of the pipe, which is arguably more pessimistic. I agree that guillotine failure could be arguably pessimistic over a through wall crack, however, in my opinion there was insufficient deterministic evidence to justify that the conservatisms stated in Section 3.3 of Reference 9 could be used to account for ignoring the NUREG recommended safety factor (√2). In addition, since the NUREG data comes with a perceived caution to the user of the data relating to highlighted uncertainties with the test data. Claims on integrity, based on raw data points, are therefore, in my opinion, not acceptable.

88. This situation occurred very late in the assessment process. In my opinion, there is not enough time available for any further analytical work needed to underpin Hitachi-GE's claimed and argued safety margin in Reference 9 and Reference 13.

89. The lack of demonstrable evidence as described above challenges the failure criteria (Section 3.3 of Reference 9) adopted by Hitachi-GE for all the UK ABWR pipewhip assessments. Therefore, I judge that the future licensee should review the failure criterion, revise/update all the UK ABWR related pipewhip assessments that have used that criterion, and how this affects Structural Integrity classification. This is

captured by assessment finding AF-ABWR-IH-15 within Internal Hazards (Reference 14)

90. Considering the significance and impact of the assessment finding detailed assessment of pipewhip work presented in Reference 9 has not been performed. However, for completeness of the report, I provide, below, a high level review of the assessment undertaken.

4.2.3.1 POSTULATED PIPE RUPTURE LOCATIONS

91. In Reference 9 and Reference 10, Hitachi-GE have assumed that the pipe preferentially fails at the terminal welds (e.g. before a bend or a valve etc.) over any other intermediate location. Furthermore, Hitachi-GE have considered two pipewhip movements: longest pipe length and greatest whipping angle.

92. In my opinion consideration of the longest length is pessimistic in the sense, that the whipping energy of the pipe is directly proportional to the length of the pipe (R3, Reference 11). In general, I judge this is pessimistic from an energy consideration, if the postulated pipe length is greater than the calculated hinge pipe length by R3. Conversely, consideration of the greatest whipping angle would ensure that any potential SSC may be hit and compromised by the whipping pipe, albeit of lesser energy, which otherwise would have been missed by the longest pipe due to possible space constraints. This mode would necessitate postulating failure at intermediate locations (weld or parent) in lieu of terminal welds only.

93. Whilst I agree that the longest length and/or greatest angle would most likely cover most of the pipe-to-pipe or pipe-to-SSC interactions, however, Hitachi-GE still needs to demonstrate that all such potential interactions are accounted for. Hence, in my opinion, Hitachi-GE's assumption of failure of the terminal welds for the pipe runs is not bounding for all possible cases, in the absence of detailed actual pipe layout, and needs to consider rupture at intermediate locations (weld or parent) of the pipe runs to capture interactions with potential SSCs, on a case by case basis. This is raised as an Assessment Finding (AF-ABWR-IH-15) within the Internal Hazards Assessment Report (Reference 14).

94. The issue with Hitachi-GE's choice of bounding location of failure in the pipe run (Reference 9 and 10) was raised in RQ-ABWR-1382 (Reference 15) and RQ-ABWR-1403 (Reference 16) and Hitachi-GE responded stating that "*consequences of failure at other locations can be largely bounded by assuming failure at weld locations*". As explained earlier, whilst this could potentially be true, however, in my opinion, the gap still exists to cover all such possibilities and needs to be assessed on a case-by-case basis following actual pipe layout. Hence, I judge that the assessment finding raised in (Reference 14) is valid, justified and aligned with the Structural Integrity position.

4.2.3.2 SMALL BORE HIGH ENERGY PIPING

95. I am generally content with the claims/arguments/evidence presented with the small bore piping systems (<50mm diameter) inside the PCV, in Section 5 of (Reference 9). This is primarily because the small bore high energy pipework would be bounded by those of larger diameters, because the smaller bore would result in lower forces in comparison. Additionally, for all the small bore pipes inside PCV, which are provided with restraints, Table 15 of (Reference 9) shows adequate design margins for safety. Hitachi-GE have also highlighted a postulated pipewhip scenario for such a representative pipe in Figure 49 and Figure 50 (Reference 9). I agree, considering the layout that even if such an event of rupture to happen, the effect would be localised inside the PCV and the consequence are deemed to be limited.

4.2.3.3 MODERATE ENERGY PIPES

96. Hitachi-GE defined (Reference 9) the medium energy systems as those with internal pressure <1.95MPa and temperature <95°C. In support of this assumption Hitachi-GE referred to NUREG-0800, IAEA Safety Standard and R3 (Reference 11). I am content with this sentencing as this conforms to the recommendation R3, which is used as the RGP for the pipewhip assessments for the UK ABWR design.

97. Hitachi-GE have claimed in Section 6 of Reference 9 that the moderate energy pipes would be bounded by large bore high energy pipes. Whilst I agree with that judgement in general, however, it still remains to be demonstrated that it is a bounding claim. This is because, Hitachi-GE haven't given any evidence to ensure that one or more medium energy large bore pipe/s would not bound small bore high energy pipe/s and the consequences thereof.

98. I agree with the screening logic chart presented in Figure 52 of Reference 9, which seems to be a reasonable sentencing methodology, however, as stated earlier, because of the lack of the bounding evidence, Hitachi-GE proposes detailed work post GDA for further evaluation. Future licensees should ensure that this work is followed up as part of normal business.

99. This issue was raised by ONR with Hitachi-GE in RQ-ABWR-0426 (Reference 17) & RQ-ABWR-1403 (Reference 16). I judge that Hitachi-GE's responses to be adequate in terms of closing out the RQs, but that the item of normal business detailed above should be progressed through licensing.

4.2.3.4 POSTULATED PIPE RUPTURE – DESIGN BASIS VS. BEYOND DESIGN BASIS

100. In Section 8 of Reference 9, for potential effects of pipe-to-pipe interaction inside the PCV, Hitachi-GE claims that for Pipe 1, the frequency of double ended guillotine failure (DEGB) is $< 10^{-5}$ per year, because all the pipes inside the PCV are ASME Section III. I agree that this is a reasonable estimate of failure probability and is acceptable for an ASME section III compliant vessel.

101. Hitachi-GE further claims that the probability of the consequential failure of the Pipe 2, being impacted by Pipe 1 is $< 10^{-7}$ per year. To substantiate that Hitachi-GE provided the evidence by undertaking further numerical analysis considering a worst case scenario where the postulated whipping pipe is assumed to rupture five target pipes (Reference 9). I accept these arguments on the basis that they are not front-line. This subject is further expounded in AF-ABWR-IH-15.

102. I have reviewed the pipe layout presented in Figure 56 of Reference 9 and in my opinion the scenario seems to be reasonably bounding. This is based upon a consideration of energies and considers the postulated failed pipe length.

103. In support of the claims and arguments, Hitachi-GE considered the following cases in Reference 9.

- **Postulated Pipe Failure with DEPSS**

104. Hitachi-GE have provided information on the internal hazard posed by a possible pipe impact on the Drywell Equipment Pipe Support Structure (DEPSS). This has the possibility of impacting the Structural Integrity classification.

105. In the postulated scenario, the pipe hits the DEPSS member and the analysis predicts significant plastic deformation of the member, but no failure. I have reviewed the methodology used in the numerical modelling and I agree that the plastic strains calculated at the DEPSS member following the postulated impact remain still below the allowable limits, which include the constraint effects due to multi-axial stresses.

106. In Figure 62 of Reference 9, Hitachi-GE claims that following the impact, the pipe would stop moving any further and that the predicted minimum clearance is 0.7m from the nearest pipe, which ensures no consequential impacts.

107. I do not fully agree that following the postulated impact, the whole length of the broken pipe would stop any further movement. I judge that, following the postulated impact a significant portion of the initial kinetic energy of the broken pipe would indeed be dissipated due to plastic deformation of the DEPSS member as well as the pipe itself. However, the pipe effectively available between the DEPSS impact location and the free end would still continue to move further due to inertia of motion, albeit with much less energy. This may result in consequential impact/s with neighbouring pipe/s. I therefore judge that Hitachi-GE's claimed minimum clearance of 0.7m (~2.3 feet) may potentially be too small to minimise the risk of "consequential" impact to ALARP.

108. I judge, therefore, that further work is necessary to substantiate the claim on minimum available clearance to minimise the risks of consequential impact to ALARP. This should be pursued by the future licensee as part of normal business, and through the resolution of the related AF within the Internal Hazards discipline (AF-ABWR-IH-15). This will be treated as part of normal business within Structural Integrity.

4.2.4 Reactor Internal Vibration Monitoring

109. The reactor internals (RINs) comprise several components inside the RPV; the RINs fulfil several functions, key ones include the restraint and supporting of the core, directing and controlling flow through the core, and the control of the location of fuel assemblies and control rods. In doing so, the RINs contribute to the fulfilment of the fundamental safety functions of controlling reactivity and the removal of heat from the reactor. Many of the RINs components therefore play significant role in reactor safety and they are identified by Hitachi-GE as UK Structural Integrity Standard Class 1.

110. In the Structural Integrity level 4 meeting in June 2015 (Reference 22), Hitachi-GE indicated that no through-life monitoring of the Reactor Internals (RINs) for vibration was proposed for the UK ABWR. This contrasts with the RPV lower shell/support region which includes monitoring for vibration induced from the RIPs (Reactor Internal Pumps). Hitachi-GE confirmed that for the RPV, recorded displacements (and hence inferred stress amplitudes) in combination with plant operating limits provide the basis for managing the risk of fatigue initiation.

111. I noted that the sources of potential vibration differ between the RIPs and the RINs: with vibration of the RPV related to a forced vibration input from operation of the RIPs; whereas for the RINs, flow induced vibration is the driver. FIV of reactor internal components is a known driver in light water reactor designs. I also noted that sources of good practice are available e.g. US NRC Regulatory Guide 1.20, Rev 3.

112. ONR's stated expectation was for Hitachi-GE to provide a demonstration of why their position on through-life monitoring of the RINs was ALARP. To form a view on whether Hitachi-GE proposals for managing the integrity of the RINs under potential vibration loading are ALARP, ONR raised the following questions and points of clarification in RQ ABWR-0639 (Reference 168):

1. Hitachi-GE to explain why there is provision for vibration monitoring through-life local to the RPV support region but not for the RINs.
2. Hitachi-GE to confirm whether their arrangements to monitor for FIV for the RINs comply with good practice e.g. US NRC Regulatory Guide 1.20, Rev 3.
3. Hitachi-GE to confirm whether the UK ABWR is a 'prototype' under RG 1.20 Rev 3 or equivalent guidance.
4. Hitachi-GE to confirm whether there are any design changes introduced for the UK ABWR compared with the reference plant used for FIV testing

113. Hitachi-GE presented a satisfactory position regarding the lack of provision of vibration monitoring equipment. This was related to the minimisation of penetrations; hence improving the Structural Integrity of the RPV. Additional monitoring equipment will require additional penetrations into the RPV of the ABWR, hence reducing the overall Structural Integrity.

114. Hitachi-GE also provided assurances and demonstration that the arrangements to monitor FIV was in line with US NRC Regulatory Guide 1.20, but did not expand their search for good practice and operational experience beyond this comparison. It would benefit any future licensee to expand upon this narrow interpretation of relevant good practice in relation to vibration monitoring. This can be pursued through the normal business of the licensee.

115. Hitachi-GE asserted that the UK ABWR will not fall into the category of "prototype" under the definitions given in US NRC Regulatory Guide 1.20. They did not, however, provide any formal discussion of this over and above the assertion that Kashiwazaki-Kariwa unit 6 (KK6) is considered the prototype plant and that it encountered "*no unusually high vibration levels".* I judge that this is an acceptable position for GDA and that data relating to this plant will not be available due to commercial confidentiality reasons. For licensing purposes, however, this will not be an acceptable position. Any future licensee should gain access to this information so as to inform their running of the UK ABWR and make appropriate judgements on the inspection regimes applied and confirm that these KK6 tests are commensurate with UK expectations. This can be pursued through the normal business of the licensee.

116. Hitachi-GE confirmed that the only design changes made for the RINs, compared with the reference plant of KK6, are a change in product form from plate to forged material, and an extension in the plant design life from 40 to 60 years. I concur with Hitachi-GE's judgement that this is unlikely to be a significant change and that further analysis is unnecessary.

117. In conclusion, I judge that the position regarding reactor internals vibration is adequate for Hitachi-GE to progress through Step 4 of GDA, but that matters remain that should be considered by the licensee in through normal business.

4.2.5 Specifics of Manufacture for the RPV Top Head

118. At the Structural Integrity Level 4 meeting in September 2015, Hitachi-GE presented details on how they plan to provide the ALARP justification for head manufacture. This was as part of the on-going discussions on RPV design, being managed through RO-ABWR-0003. Further detail on the design of the RPV head is provided in section 4.4.3 below. The proposal from Hitachi-GE was to utilise a multi-petal design of RPV top head, but the reasoning for the choice of the number of petals was not covered in detail.

119. The size of the petals drives the amount of welding that will be required in the fabrication of the head. ONR's expectation, as evinced by SAP EMC.9, is that the choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component. Therefore, I stated to Hitachi-GE that my expectation was for them to have taken all reasonably practicable steps to increase the petal size; thereby reducing the weld length.

120. During the Level 4 meeting in September, Hitachi-GE stated that it was not practicable to use fewer petals, because the machinery available at the chosen supplier could not deal with larger work-pieces. In response to this assertion, I consider that further

information is needed to produce an ALARP assessment for RO-ABWR-0003. I raised RQ-ABWR-0657 (Reference 23) to address this. The queries on that RQ were:

1. Are there alternative suppliers who could provide larger scale plate materials to suitable nuclear grades and quality expectations?
2. Is it practicable for the chosen RPV manufacturer to invest in larger scale production equipment?

121. This matter was treated outside of the main RO-ABWR-0003 discussions, but I considered that without a satisfactory conclusion to this RQ, RO-ABWR-0003 closure would be difficult.

122. Hitachi-GE responded to this RQ with Reference 24. This document stated that making a change in supplier runs the risk of a drop off in materials quality, and hence nuclear safety. This was related to the long history of joint working between Hitachi-GE and the RPV manufacturer and the known safety culture present at that supplier. I concur that this is a strong position and that nuclear safety may best be served by remaining with the original supplier.

123. I also concur with Hitachi-GE's argument that to change equipment, within the production facility proposed, also poses a threat to making a quality product, and hence a threat to nuclear safety. A full discussion of the level of this threat will not be possible, however, until the procurement phase commences within licencing.

124. I do note, however, that the facility to make larger plates of this type does exist within the wider global supply chain. I judge, therefore, that this constitutes a minor shortfall:

MS-ABWR-SI-01 To ensure that supplier selection is not unduly influencing nuclear safety, any future licensee should check to see if other credible suppliers are available, to manufacture the plate material for RPV dome petals, who could supply material to a similar or higher level of safety.

4.2.6 Position of the Shroud Support Leg Weld

125. From the design diagrams for the RPV, it became apparent that the position of the shroud support leg is coincident with the bottom head petal to dome circumferential weld. It is not immediately obvious that this meets ONR's expectations, specifically those in EMC.10: "*The positioning of welds should have regard to high-stress locations and adverse environments*"

126. In meetings in late January 2016, Hitachi-GE explained that there were production benefits, and associated safety benefits, to this arrangement. I raised RQ-ABWR-0789 (Reference 25), which asked for evidence around those claims. Hitachi-GE chose to respond to these claims using the RPV Design Philosophy Report (Reference 26) as a vehicle. Appendix B deals directly with these matters.

127. It is clear from Hitachi-GE's submissions that the level of stress at this location is, despite the discontinuity formed by the presence of the shroud support leg, suitably low with large margins against design limits, and that the fatigue usage factor (FUF) is reported to be 0.01. I am content, therefore, that the intent of EMC.10 is met. Moreover, I am content with Hitachi-GE's assertions that to change the location of the weld would require a significant effort, which is not proportionate to the potential safety benefit.

128. I judge the ALARP case put forward to retain the current geometry to be adequate.

4.2.7 Main Steam Isolation Valve Manufacture

129. The Main Steam Isolation Valves (MSIVs) on the UK ABWR have been designated as Very High Integrity (VHI) components by Hitachi-GE for Structural Integrity. This means that the pressure boundary function of the valves is VHI, the actuation of the valves is treated within the Mechanical Engineering discipline (Reference 169). There are also important Mechanical Engineering claims made on the MSIVs to limit the release of reactor coolant and radioactive material to the surroundings in the event of a MS pipe rupture by closing the MSIVs. This plant level safety function has been categorised as Category A by Hitachi-GE and the components to deliver it are designed to meet Safety Class 1 requirements.

130. Hitachi-GE have proposed obtaining mechanical properties data from a test block, cast alongside the main valve body. These data are an important input into the Structural Integrity Safety Case. The representative nature of this test block will be confirmed by sacrificing the equipment qualification (EQ) valve, once the EQ process is completed.

131. The expectations for pressure boundary integrity, as outlined in paragraph 220 of the Safety Assessment Principles (SAPs), is that:

"… properties should be obtained from fully representative samples of the material especially when the component or structure performs a principal role in ensuring nuclear safety."

132. I had been informed, via routine Level 4 meetings with Hitachi-GE, that the EQ valve was planned for procurement from a different supplier to the production valves. If this were the case, the representative nature of the test block would be confirmed for the EQ valve supplier only and not the production valve supplier. I raised Reference 27 to confirm this understanding.

133. Hitachi-GE responded that the supply chain and production route for the EQ MSIV will be representative of the supply chain and production route of the MSIVs for final plant. Hitachi-GE noted that they had already ordered the EQ MSIV and plans to use the same suppliers for final plant.

134. Hitachi-GE stated that they do not intend to change any of the production parameters or procurement routes between the EQ valve and the Production valves and will review the applicability of the data from the EQ valve test block prior to the use of the production valve data. I consider this to be an acceptable situation. If any production parameters or suppliers are changed between the fabrication of the EQ and production valves, my expectation is that the licensee will reconfirm the fully representative nature of the test blocks by alternative means. This is likely to be an onerous task. This will be monitored by ONR through normal business.

4.2.8 Reactor Internals Inspection

135. Hitachi-GE acknowledge that Stress Corrosion Cracking (SCC) or Irradiation Assisted Stress Corrosion Cracking (IASCC) of the reactor internals cannot be completely ruled out over the 60yr design life of the ABWR and consequently, in-service inspection will play an important role in underpinning the structural integrity of safety related components.

136. Reference 29 presented a limited strategy for the ISI of reactor internals; the responsibility for the full implementation of the ISI lies with the licensee. I note that while Hitachi-GE described the general requirements for the detection of SCC and IASCC, there is some specific detailed information that is presented regarding the inspection approach. The method for the detection of SCC/IASCC will be remote visual inspection using cameras and be implemented in accordance with the requirements of ASME XI (the strategy document refers to the Japanese standards but states that these are equivalent to ASME XI).

137. The ASME method for determining the sensitivity for the VT-1 inspection uses a standard resolution card (ASME XI Table IWA-2211-1). This sensitivity is not deemed to be sufficient for detecting SCC and consequently enhanced VT-1 (EVT-1) is recommended by the BWR Vessel Internals Program. To meet the requirements for EVT-1, the inspection system must be capable of detecting a wire with a diameter 0.013mm. The document describes the use of 'MVT-1' where the sensitivity requirement is to detect a wire with a diameter 0.025mm, twice that for the EVT-1.

138. An evaluation of the effectiveness of visual inspection indicates that the crack opening displacement at the surface is a key parameter for the detection performance; cracks with an opening displacement of 20μm can be detected with around 80% detection frequency under good conditions. While many mature SSC may have opening displacements greater than this, earlier onset of SCC may not. I judge that an insufficient case has been presented to demonstrate that the use of the techniques in Reference 29 will match the demands of the safety case. I have raised the following assessment finding to ensure this matter is taken forward beyond GDA.

AF-ABWR-SI-05 During GDA, Hitachi-GE have not demonstrated that the proposed inspection of reactor internals will meet the needs of the Safety Case. The licensee shall demonstrate that the inspection capability of the reactor internals matches the safety case requirements with regard to Stress Corrosion Cracking and Irradiation Enhanced Stress Corrosion Cracking. This must consider a demonstration of the capability of the in-service inspection taking into account the essential parameters of the defects and the inspection conditions.

4.2.9 Topic Reports

139. The topic reports are the next level of document below the PCSR. Noting that the PCSR acts as a signposting document with little technical detail contained within it, I considered it appropriate to perform an in-depth assessment of a large sample of these reports. The reports themselves present claims and arguments for each of the individual components discussed. All of the VHI components have been described in a topic report, with representative components being described for lower classes, with detail proportionate to the nuclear safety function of those components.

140. My expectations from these reports is that they should provide a structure by which the Safety Case can be framed and that evidence underlying the claims presented. This may be with reference to supporting documents which contain the evidence. UK expectations in terms of inspection and defect tolerance analysis (DTA) were treated separately, within the Hitachi-GE responses to RO-ABWR-0001, so it was not my expectation that the topic reports contain discussions on this, or other ROs.

141. I have sampled the topic reports provided by Hitachi-GE at a depth proportionate to their nuclear safety function. As discussed in Section 3, above, the key documents underpinning the claims made within Structural Integrity are the topic reports.

142. As part of my assessment I have used the services of a technical support contractor (Frazer Nash Consultancy) to perform detailed assessment of the final versions of some of these reports, with previous revisions being assessed internally to ONR. I have sampled the following topic reports:

- Reactor Pressure Vessel; (VHI)
- Inboard Main Steam Isolation Valves; (VHI)
- Main Steam Piping inside the RCCV; (VHI)

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- Reactor Internals; (ASME Class1)
- RPV Class 1 Components; (ASME Class 1)
- Feedwater Piping; (ASME Class 1)
- SLC Storage Tank; (ASME Class 2)
- HP Turbine Casing. (ASME Class 3)

143. These samples have provided me a view over all of the VHI components, which are the most important from a nuclear safety viewpoint. The VHI components are the Reactor Pressure Vessel (RPV), Main Steam (MS) Lines (sections within containment only) and the Main Steam Isolation Valves (MSIVs) (inboard of containment only).

144. I have also sampled a number of the class 1 component reports, specifically the Reactor Internals, RPV Class 1 components, and feedwater piping topic reports. A high level review was performed on the HCU topic report. This sampling is in line with the sampling strategy put forward in Section 2.4, above. The ASME Class 2 and Class 3 representative components were also sampled although the detail presented, and the assessment performed, was at a lower level compared with the higher classification components.

145. A series of RQs have been raised against these submissions, these are discussed below. Also included below are discussions of RQs pertinent to the topic report under discussion, where these have affected the Safety Case presented.

4.2.9.1 REACTOR PRESSURE VESSEL TOPIC REPORT

146. The Topic Report on RPV Structural Integrity is presented in Reference 31. The review performed by FNC is recorded as Reference 36 and included assessment of Revisions 1 and 2 of the Topic Report.

147. The components and welds of the RPV main pressure boundary have been assigned a classification of VHI, with the exception of the closure stud bolts, which have been individually assigned Standard Class 1 and for which a critical number of multiple failures has been classified VHI. Reference 31 covers only the VHI components of the RPV and other reactor components are addressed elsewhere.

148. In line with PCSR claims, and the Safety Case Structure Report, safety claims made in the Topic Report are as follows:

> Claim 1: Sound design promotes very high integrity.
>
> Claim 2: High quality of manufacture is specified to achieve very high integrity.
>
> Claim 3: Functional testing confirms integrity at start of life.
>
> Claim 4: Lifelong integrity is demonstrated by failure analysis.
>
> Claim 5: Inspection, monitoring and surveillance confirm very high integrity through life.

149. Arguments supporting each of these claims are made within the body of the Topic Report and the location of the evidence supporting these arguments is referenced. Arguments are made for the provenance of the design, assessment, manufacture, inspection and testing. Being a VHI system, my expectation is that the assessment requires a demonstration of defect tolerance by calculation and a programme of continual monitoring via In-Service Inspection (ISI).

150. TAGSI (UK Technical Advisory Group on the Structural Integrity of Nuclear Plant) recommends the provision of conceptually diverse arguments for defence in depth against the risk of failure, as presented in the form of a multi-legged safety case. Compliance with an established nuclear design code provides a foundation for substantiating each claim. Additional arguments and evidence is then presented for the conservative nature of the design loads, the choice of materials, management of degradation, defect tolerance and inspection regime. I judge that, although the TAGSI approach has not been followed exactly, the Hitachi-GE five claims can be mapped against the TAGSI four legs simply. Hence I judge that the claims and arguments, presented in Reference 31, generally follow a source of UK good practice; in accordance with that advocated by the TAGSI for demonstrating "Incredibility of Failure in SI safety cases".

151. The safety claims, arguments and evidence have also been assessed adequately against the expectations articulated in the SAPs (Reference 32). Any potential gaps have been addressed by ONR through the RQ process.

152. I judge that the report is comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence, as is available at the time.

153. The Topic Report also addresses a number of Regulatory Observations (ROs) that emerged from the GDA Step 2 review and subsequently. The ROs related to this document are:

- RO-ABWR-0001, Avoidance of Fracture;
- RO-ABWR-0003, RPV Design;
- RO-ABWR-0004, Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary.

154. The RP has developed and published resolution plans for each of these ROs and a brief narrative is provided in the Topic Report.

155. In total 30 comments were raised against the RPV Topic Report Revision 1 (Reference 31). There were 9 comments at Category 3 and 21 at Category 4. These comments were presented to Hitachi-GE as RQ-ABWR-1287 (Reference 33). Note that the comment numbering runs from 1 to 31, as one comment number (26) was not used.

156. Hitachi-GE have provided responses to the RQ comments in terms of a formal response document, dated 28th April 2017 (Reference 34) and an Up-issue of the Topic Report to Revision 2.

157. Having considered all of the information provided, I judge that 30 comments have been satisfactorily answered, including all those of Category 3 and above. Only one minor (Category 4) comment remains open (Comment 25), which relates to leakage detection systems and the absence of a leak before break (LLB) argument for VHI components. Noting that this is a category 4 comment, I do not consider this to be any impediment to GDA. Similar arguments are made for other VHI components. Further information has been requested on the Basis of Safety Cases on Reactor Coolant Pressure Boundary. I note that ONR does not generally accept LBB argument as frontline arguments in safety submissions. LBB arguments remain a useful vehicle for demonstration of defence in depth.

158. There has been a commitment to address certain comments during the Site Licensing phase. Specifically, these were:

- Materials identification, control of storage & issue (during construction).
- In-service repair and modifications.

159. Deferring these points until Site Licensing is considered an appropriate response at this time. The future licensee should progress these as part of normal business.

160. I judge that the claims and arguments presented in the RPV topic report are adequate and appropriate to provide a VHI Safety Case for the RPV, within the context of GDA. I look to the subordinate documents and the discussions presented under RO-ABWR-0001 to provide supporting evidence.

4.2.9.2 MAIN STEAM ISOLATION VALVE TOPIC REPORT

161. The Topic Report on MSIV Structural Integrity is presented in Reference 35. The review performed by my TSC is recorded as Reference 36 and included assessment of Revisions 1 and 2 of the Topic Report.

162. The valve bodies of the inboard MSIVs, within the Reinforced Concrete Containment Vessel (RCCV), have been assigned a classification of VHI. The Body Bonnet, Cover Flange, Bolting, Valve Stem and weld to the Drain Line have all been designated Standard Class 1 components. Reference 35 covers the VHI inboard valve body only and the welds to the Main Steam Piping are addressed in the MS Piping Topic Report (Reference 37). It is noted that the outboard MSIVs are designated Standard Class 1, but are designed and manufactured to the same standards as the inboard MSIV.

163. Similar to the RPV, the main safety claims made in the Topic Report are:

- Claim 1: Sound design promotes very high integrity.
- Claim 2: High quality of manufacture is specified to achieve very high integrity.
- Claim 3: Functional testing confirms very high integrity at start of life.
- Claim 4: Lifelong integrity is demonstrated by failure analysis.
- Claim 5: Inspection, monitoring and surveillance confirm very high integrity through life.

164. Arguments supporting each of these claims are made within the body of the Topic Report and the location of the evidence supporting these arguments is referenced. Arguments are made for the provenance of the design, assessment, manufacture, inspection and testing of the MSIV body. Being a VHI component, the assessment requires a demonstration of materials testing, defect tolerance by calculation and a programme of continual monitoring via ISI.

165. The structure of the Topic report, especially the claims and arguments, is very similar to the RPV topic report. This consistent approach is welcome and aids understanding for the reader. Similar to the RPV topic report, I judge that the claims and arguments, presented in Reference 35, generally follow a source of UK good practice; in accordance with that advocated by the TAGSI for demonstrating “Incredibility of Failure in SI safety cases”.

166. The safety claims, arguments and evidence have also been assessed against the expectations articulated in the SAPs (Reference 32) and any comment were raised and addressed via the RQ process.

167. On the whole the report is comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence.

168. The Topic Report also addresses a number of ROs that emerged from the GDA Step 2 review, these are:

- RO-ABWR-0001, Avoidance of Fracture
- RO-ABWR-0004, Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary.

169. RO-ABWR-0004 is primarily directed at the RPV, but the MSIV Topic Report concedes that the same principles are applicable to the MSIV.

170. In total 33 comments were raised against the MSIV Topic Report as part of (Reference 38). There were no Category 1 or 2 comments, 6 at category 3, 25 at category 4 and 2 additional uncategorised comments. Details of these comments can be found in the published RQ.

171. All category 3 comments have been closed via Hitachi-GE's formal response to RQ-ABWR-1288 (Reference 39). Two comments, relating to the control of materials during construction and in-service repair and modification, were not considered to have been fully addressed in the formal responses provided. However, the responses to the RPV Topic Report, discussed in Section 3, had sought to defer this issues until the Site Licensing phase, which was considered to be an appropriate response at this time. The two comments (Comments 30 and 33) have been closed at this time, on the expectation that these issues will be considered during the Site Licensing phase for all VHI and Standard Class 1 components contributing to nuclear safety.

172. Having considered all of the information provided, it has been agreed that all 33 comments have been satisfactorily answered. Hence, I judge that the claims and arguments presented in the MSIV topic report are adequate and appropriate to provide a VHI Safety Case for the MSIVs, within the context of GDA. I look to the subordinate documents and the discussions presented under RO-ABWR-0001 to provide supporting evidence.

4.2.9.3 MAIN STEAM PIPING TOPIC REPORT

173. The Topic Report on MS Piping Structural Integrity is presented at Reference 40. The review performed by my TSC is recorded as Reference 36 and included assessment of Revisions 1 and 2 of the Topic Report.

174. This Topic Report covers all inboard parts of the four Main Steam Piping lines, from the welds at the RPV main steam outlet nozzles to the welds at the outboard MSIVs, which are located immediately inside of the RCCV containment boundary. These sections of the Main Steam (MS) system have been assigned a classification of VHI. Piping downstream of the outboard MSIVs has been assigned a lower classification of Standard Class 1 and is not considered in this Topic Report. The Safety Relief Valves (SRVs) and discharge lines have been designated Standard Class 1 and are also not considered in this Topic Report.

175. The main safety claims made in the Topic Report are identical to the claims produced above for the other two VHI components:

- Claim 1: Sound design promotes very high integrity.
- Claim 2: High quality of manufacture is specified to achieve very high integrity.
- Claim 3: Functional testing confirms integrity at start of life.
- Claim 4: Lifelong integrity is demonstrated by failure analysis.

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

176. Arguments are made for the provenance of the design, assessment, manufacture, inspection and testing. Being a VHI system, the assessment requires a demonstration of defect tolerance by calculation and a programme of continual monitoring via ISI.

177. Similar to the two preceding VHI components, the structure follows a modified version of the standard TAGSI approach. I judge that this is acceptable.

178. The safety claims, arguments and evidence were assessed against the expectations articulated in the SAPs; any potential gaps addressed through the formal RQ process.

179. On the whole the report is comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence.

180. The Topic Report also addresses a number of ROs that emerged from the GDA Step 2 review:

- RO-ABWR-0001, Avoidance of Fracture.
- RO-ABWR-0004, Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary.

181. RO-ABWR-0004 is primarily directed at the RPV, but the MS Piping Topic Report concedes that the same principles are applicable to the piping. It may be considered more appropriate for the MS Piping Topic Report to refer to RO-ABWR-0035, which has a broader scope than RO-ABWR-0004.

182. In total, 30 comments were raised against the MS Piping Topic Report. Specifically there were 7 Category 3 comments and 23 category 4 comments. The details of these are recorded within RQ-ABWR-1289 (Reference 40). Hitachi-GE have provided responses to the RQ comments through the Formal response document (Reference 42), and via up-issue of the Topic Report (Reference 40) to Revision 2.

183. Comment 22 of RQ-ABWR-1289, relating to the control of materials during construction was claimed to be outside of the scope of the GDA. A second comment (Comment 25), relating to in-service repair and modification, was not considered to have been fully addressed in the formal responses provided. However, the responses to the RPV Topic Report, discussed above, had sought to defer these issues until the Site Licensing phase, which was considered to be an appropriate response at this time. The two comments (Comments 22 & 25) have therefore been closed for the purposes of GDA, on the expectation that these issues will be considered as part of normal business during the Site Licensing phase for all VHI and Standard Class 1 components contributing to nuclear safety. Two comments (Comments 6 & 12) have been closed through the Defect Tolerance Assessments (DTA) and the corresponding loads cases being discussed in the relevant DTA reports. The comment relating to leakage detection systems and the absence of any discussion of leak before break (Comment 18) has been closed on the basis that the identical issue is still open on RQ-ABWR-1287 and will be covered via that RQ.

184. Having considered all of the information provided, I judge that all comments have been answered satisfactorily for the purposes of GDA.

185. It is noted that there are several references to the qualification of pre-service inspections to European Network for Inspection Qualification (ENIQ) recommended practice, as outlined discussion of RO UK-ABWR-0001, Argument 4.4 and Argument 5.1). However, Argument 2.4 now refers to only ASME compliant inspection procedures. As will be discussed within the section of this document relating to RO-

ABWR-0001, it is ONR's expectation that the ENIQ methodology, or demonstrably equivalent methodology, be used for VHI components. ASME compliant inspections may not, in isolation, provide the appropriate level of confidence.

186. This notwithstanding, I judge that the MS Piping Topic Report is adequate for the purposes of GDA.

4.2.9.4 REACTOR INTERNALS TOPIC REPORT

187. The Topic Report on Reactor Internals (Reference 43) was reviewed by my TSC. This was included in my sample as the Reactor Internals (RINs) are the principal example of Structural Integrity components which do not form part of the pressure boundary. They are classified as ASME Class 1, and perform a significant safety function in supporting the core and reactor pressure vessel internal component. The TSC review report is presented at Reference 36.

188. The Reactor Internal components (RINs) are subdivided in two categories; core support structures, and other internal structures. Each category contains several individual sets of components; these are outlined below:

- Core Support Structures

  a) Core Shroud;

  b) Top Guide and Hardware;

  c) Core Plate and Hardware;

  d) Control Rod Guide Tube;

  e) Orificed Fuel Support;

  f) Peripheral Fuel Support;

  g) Control Rod Drive (CRD) Housing (within the RPV boundary);

  h) Shroud Support.

- Other Internal Structures that contribute to Emergency Core Cooling System (ECCS), Residual Heat Removal (RHR), and emergency shutdown include:

  a) Feedwater Sparger;

  b) Low Pressure Flooder Sparger;

  c) High Pressure Core Flooder Sparger and Coupling;

  d) Core Plate Differential Pressure Line;

  e) Start-up Range Neutron Monitor Drytube;

- Other Internal Structures that do not contribute to those functions include:

  f) In-Core Housing (within the RPV boundary)

  g) Head Vent and Spray Nozzle (within the RPV boundary)

  h) Reactor Internal Pump (RIP) Guide Rail

  i) Lower Guide Rod

j) Steam Dryer and Dryer Units;

k) Shroud Head;

l) Steam Separator;

m) Shroud Head Bolt;

n) In-Core Guide Tube (ICGT);

o) ICGT Stabilizer;

p) RIP Differential Pressure Line;

189. In general, the core support structures locate, position and support the fuel assemblies, control rods, shroud head assembly, neutron source, in-core instrumentation, differential pressure lines, spargers and internal piping. The other internal structures direct coolant flow, separate steam, and support instrumentation utilized for plant operation.

190. The RINs that have a function to support the core, functions related to ECCS or RHR, and emergency shutdown are classified as Standard Class 1. Other RINs are classified as Standard Class 2 & 3 or non-class.

191. The RINs Topic Report considers only the structural integrity aspects of the Standard Class 1 RINs identified above.

192. The main safety claims made in the Topic Report are:

- Claim 1: High Quality Is Achieved Through Good Design and Manufacture.
- Claim 2: Design Code Assessment Provides Assurance of Integrity.
- Claim 3: Functional Tests Confirm Integrity.
- Claim 4: In-Service Inspection & Monitoring Forewarns of Failure.

193. Arguments are made for the provenance of the design, assessment, manufacture, inspection and testing. Being Standard ASME Class 1 systems, the VHI requirement for a robust demonstration of defect tolerance has been relaxed by Hitachi-GE, with the focus being on design, manufacture, inspection and testing to an internationally recognised nuclear standard. I judge that this is appropriate provided they can be demonstrated to be in line with a source of good practice.

194. The claims and arguments, presented in the topic report, generally follow UK good practice and there is a degree of similarity to those presented for the VHI components discussed previously. The TAGSI standard structure for a Safety Case has been followed directly in this instance. I judge this to be appropriate.

195. The safety claims, arguments and evidence have also been assessed against the expectations articulated in the SAPs; potential gaps were raised with Hitachi-GE via the RQ process.

196. The report is, generally, comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence.

197. The Topic Report also references three ROs that emerged from the GDA Steps 2 and 3 reviews. These ROs contain a significant amount of the supporting evidence subordinate to the Topic Report:

- RO-ABWR-0019, UK ABWR Reactor Chemistry Safety Case: Strategy, Plan and Delivery.

- RO-ABWR-0022, Demonstration that the Primary Cooling System Operating Chemistry Reduces Risks SFAIRP.

- RO-ABWR-0035, Robust Justification for the Materials Selected for UK ABWR (Reference 17).

198. In total 29 comments were raised via RQ-ABWR-1504 (Reference 44) against the RINs Topic Report. There were 25 category 3 and 4 at category 4. All of these comments have been satisfactorily closed by Hitachi-GE through their responses to the RQ (Reference 45)

199. I judge that the topic report for the RINs is adequate for GDA purposes.

4.2.9.5 RPV CLASS 1 COMPONENTS TOPIC REPORT

200. The Topic Report on RPV Class 1 Components is presented at GA91-9201-0001-00271 (Reference 46) and has been reviewed on behalf of ONR by my TSC (Reference 36). The RPV class 1 components are at a lower classification than the body of the RPV and have been presented and Hitachi-GE's presentation, as well as ONR's assessment is, consequently, lower.

201. The RPV Class 1 Components are subdivided in three categories:

- 1. Nozzles & Penetrations:

  a) Control Rod Drive Penetrations, including Housing (205 off);

  b) In-core Penetration, including Housing (62 off);

  c) Water Level Instrumentation Nozzle (12 off);

  d) RIP Penetrations (10 off);

  e) Pump Deck Differential Pressure Line Instrumentation Nozzle (4 off);

  f) Core Plate Differential Pressure Line Instrumentation Nozzle (4 off);

  g) Seal Leak Detection Nozzle (1 off);

  h) Steam Outlet Nozzle Instrumentation Transition Pipe (8 off);

  i) Drain Nozzle (Bottom Drain Line) (1 off);

  j) Nozzle Safe Ends (5 types) and Thermal Sleeve (3 types);

  k) Steam Outlet Nozzle Extension (4 off);

  l) RIP Motor Casing (10 off).

- 2. Brackets & Pads:

  a) Stabilizer Bracket (8 off),

  b) Feedwater Sparger Bracket (6 sets),

  c) Dryer Hold Down Bracket (6 off),

d) Pad for In-Service Inspection (6 off),

e) Thermocouple Mountings,

f) Dryer Support Bracket (4 off),

g) Lifting Lug – Top Head (4 off),

h) Guide Rod (upper) Bracket (2 off),

i) Surveillance Basket Pads (2 sets),

j) Low Pressure Flooder Sparger Bracket (2 sets),

k) Guide Rod (lower) Bracket (2 off),

l) Refueling Bellows Support (1 off).

3. Other Components:

a) RPV Closure Stud Bolts, including nuts and washers,

b) Top Head O-Rings including Seat Surface,

c) RPV Support Skirt,

d) Anchor Bolts, including nuts and washers,

e) Stud Bolts for Head Vent & Spray Nozzle (N7), including nuts and washers.

202. This Topic Report considers the structural integrity of Standard Class 1 components of the RPV, these are distinct from the VHI parts of the RPV, in that the failure of these items is within the design basis. Similar to the other Class 1 components, the Safety Case is based upon the TAGSI four-legged structure. Similar to previous sections, I consider this to be appropriate.

203. Arguments below the four-legged safety claims are made for the provenance of the design, assessment, manufacture, inspection and testing. Being Standard Class 1 systems, the VHI requirement for a robust demonstration of defect tolerance is relaxed, with the focus being of design, manufacture, inspection and testing to an internationally recognised nuclear standard.

204. The safety claims, arguments and evidence have also been assessed against the expectations articulated in the SAPs (Reference 32) potential gaps against these expectations were captured in RQ-ABWR-1505 (Reference 47).

205. The report is, generally, comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence, as is available at the time.

206. The Topic Report Revision 0 makes no reference to the structural integrity issues that have been raised in previous steps of the GDA. My expectation is that the RPV Class 1 Topic Report should at least have discussed the Regulatory Observations relating to the Class 1 components of the RPV. Following the issue of revision 1 of the topic report Hitachi-GE have provided a suitable linkages to the ROs:

- RO-ABWR-0002, CRD Penetration Design.
- RO-ABWR-0034, Demonstrating the Inclusion of a “Bottom Drain Line” in the UK ABWR Design Achieves Inherent Safety and Reduces Risk SFAIRP.

207. All comments potential shortfalls against the SAPs were presented to Hitachi-GE via RQ-ABWR-1505. This included 15 category 3, 6 category 4, and 6 uncategorised comments.

208. Hitachi-GE have provided responses to the RQ comments through the formal response document (Reference 48) and revision of the topic report (Reference 46) to Revision 1.

209. Having considered all of the information provided, I judge that all comments have been satisfactorily answered. The comment relating to leakage detection systems and the absence of any discussion of leak before break (Comment 18) has been closed through reference to RQ-ABWR-1287 (Reference 33), where an extensively similar comment has been raised. This is covered above in section 4.2.9.1.

210. I recommend that the process for identification and control of materials (Comment 24) should be progressed by any future licensee during the Licensing phase through normal business. I judge that this topic report, and the claims and arguments contained therein, is satisfactory for the purposes of GDA.

4.2.9.6 FEEDWATER PIPING TOPIC REPORT

211. The Topic Report on FDW Piping Structural Integrity is presented in Reference 49. Revisions 1 and 2 of this report were reviewed by my TSC. The Feedwater (FDW) Piping system takes condensate from the three main condensers back to the RPV, through the RCCV penetrations and via two horizontal headers and six 12-inch riser lines that connect to nozzles on the RPV. Outboard of the RCCV penetrations, the condensate passes through several conditioning components including pumps, heaters, filters, air ejectors and demineralisers.

212. Working backwards from the RPV, the FDW Piping from the RPV nozzle to the second isolation valve outside the RCCV is designated Standard Class 1. The pipework upstream of the second isolation valve is designated Standard Class 3. Reference 49 considers the Standard Class 1 sections of the FDW Piping system.

213. The safety claims made on the FDW are in line with other Class 1 components. These align directly with the TAGSI four-legged structure. I consider that this meets UK good practice.

214. The safety claims, arguments and evidence have also been assessed against the expectations articulated in the SAPs; any potential gaps highlighted in RQ-ABWR-1509 (Reference 50).

215. I judge that the report is comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence, as is available at the time.

216. The Topic Report makes a passing reference to RO-ABWR-0035, Robust Justification for the Materials Selected for UK ABWR. RO-ABWR-0035 relates to a broad question of material selection, which has been addressed in the generation of a materials selection report. I have reviewed the topic report, at its final revision, and judge that it is consistent with RO-ABWR-0035.

217. In total 32 comments were raised against the FDW Piping Topic Report in RQ-ABWR-1509. There were, in total, 7 category 3 comments, 24 category 4 comments and 1 uncategorised comment.

218. Hitachi-GE provided responses to the RQ comments through the formal response document (Reference 51) and up-issue of the Topic Report to Revision 2 (Reference 49).

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219. Having considered all of the information provided, I judge that all comments have been satisfactorily answered. As per previous topic reports, I recommend that the identification and control of materials (Comment 28) should be addressed during the procurement and Licensing phase through normal business.

4.2.9.7 .SLC TOPIC REPORT (REPRESENTATIVE CLASS 2 COMPONENT)

220. The topic report on SLC (Standby Liquid Control System) Storage Tank Structural Integrity (GA91-9201-0001-00268) is presented in Reference 18. The SLC Storage Tank (noting that it is an atmospheric storage tank and not a pressure vessel) contains sodium pentaborate solution and is part of the SLC that injects a liquid neutron absorber (aqueous sodium pentaborate solution) into the core to shut down the reactor and maintain reactor shutdown, in the event that the control rods cannot be inserted during a SCRAM. The SLC is a secondary shutdown system with a function that is categorised as Safety Category A and has been assigned a safety classification of Class 2.

221. The main safety claims made in the Topic Report are:

- Claim 1: Sound design and design code assessment provides assurance of integrity.
- Claim 2: High quality manufacture is specified to ensure integrity is maintained throughout service.
- Claim 3: Functional testing confirms integrity at start of life.
- Claim 4: In service inspection and monitoring forewarns of failure.

222. The above claims align directly with TAGSI, and are unaltered compared with the claims made for Class 1 components. I judge that this level of detail will be capable of producing an adequate Safety Case. I further note that ONR's expectations is that effort should be directly proportionate to the nuclear safety risk involved. I consider that it would be prudent for any future licensee to ensure that effort is so directed.

223. To ensure that Arguments are made for the provenance of the design, assessment, manufacture, inspection and testing. Being a Standard Class 2 systems the focus is on design, manufacture, inspection and testing to an internationally recognised nuclear standard.

224. I judge that the claims and arguments, presented in Reference 18, generally follow UK good practice and there is a high degree of similarity to those presented for the Standard Class 1 components.

225. The safety claims, arguments and evidence have also been assessed against the expectations articulated in the SAPs; any potential gaps have been captured, via the RQ process in RQ-ABWR-1506 (Reference 52).

226. On the whole the report is comprehensive, well written and easy to follow. The document provides a well-structured set of claims, arguments and evidence, as is available at the time.

227. In total 20 comments were raised against the SLC Storage Tank Topic Report. These have been addressed in formal response document (Reference 19).Having considered all of the information provided, I judge that all comments have been satisfactorily answered.

4.2.9.8 HP TURBINE CASING TOPIC REPORT

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228. The UK ABWR steam turbine topic report is provided in Reference 20 and the HP steam turbine casing has been classified as a Class 3 item (Reference 7); I am content with this classification through my review of Reference 7. This report has been reviewed by an ONR internal SQEP resource. The casing provides the pressure boundary or containment for the reactor coolant and I looked at the claims, arguments and evidence provided by Hitachi-GE to justify the integrity and safety of the component design.

229. According to Reference 20, the design of the turbine casing is based on the Japanese Code JIS B 8201 and Hitachi-GE claimed that the design code is equivalent to ASME Section VIII. I questioned this through RQ-ABWR-1518 [Reference 21] and I am content with the response. Furthermore, I looked at the comparative study undertaken in Table 1 of Reference 20 and in my opinion the comparisons of the Japanese and American codes of the chosen parameters are reasonably similar and acceptable. I recommend that, during the procurement of the HP Turbine, a more complete comparison should be carried out as part of normal business.

230. In Reference 21, I raised a query on the use of the allowable stress of carbon steel for the casing. I am content that the minimum value has been used for the casing allowable stress considering the design temperature to calculate the minimum thickness requirement for the casing. I have reviewed the data presented in Table 2 of Reference 21 and am content that significant margins (>20%) exists in the casing thickness provided at various locations of the casing.

231. I also raised a query (Reference 21) on the potential brittle fracture issue with carbon steel for cold starts. I am content that low pressure during the cold-start would not result in high stresses at the turbine casing. Also, Hitachi-GE claimed that there is significant operational experience on “cold-starts” and there has been no reported incident. Whilst, there was no “evidence” to substantiate this claim, however, due to low pressure (hence low stress), in my opinion, this is reasonable.

232. However, in response to another query on Reference 21 on inspecting the bottom half of the casing on a regular basis, Hitachi-GE responded that it was difficult because of the rotor, which would prevent such intervention. It could only be possible during rotor maintenance or rotor exchange. I agree that this is a genuine concern**.** I judge this should be included in the overall inspection programme of the future licensee as part of normal business.

4.2.10 Topic Reports Summary and Conclusions

233. A broad range of topic reports has been sampled by ONR, using TSC and internal resource. I consider that these are suitable documents to provide the key inputs into the overall Safety Case. They are appropriately structured to provide the assurance that the safety of the plant follows the ONR SAPs. I have reviewed compliance of these SAPs using the RQ process and there are no outstanding matters.

234. As top level documents, much of the detail needed to meet ONR’s expectations for Structural Integrity are included in subordinate documents. Much of the detail of the Safety Case is included in responses to RO-ABWR-0001 and associated documents.

235. There are no assessment findings arising from this section.

4.2.11 Design Philosophy and Manufacturing Reports for VHI components

4.2.11.1 OVERVIEW OF DOCUMENTS REVIEWED

236. I sampled Hitachi-GE’s approach to the design of nuclear safety significant components through the sampling of two design philosophy reports, one covering a

component of the highest reliability and a second from a lower safety class of component.

237. The Design Philosophy Report should describe the technical basis for the component design and how it has been developed to enhance safety and reliability, hence the reports were reviewed in the context of whether they are capable of delivering the stated safety functions, whether they will be able to deliver the required level of reliability and if they meet, in broad terms, the requirements of the ONR Safety Assessment Principles (SAPs).

238. The first report reviewed is the RPV Design Philosophy Report, which is a VHI component. The second report reviewed is the Design Philosophy Report for Main Steam (MS) Piping class 1 piping (Reference 53).

239. Aspects of the RPV design have been covered through RO-ABWR-0003. The design philosophy and manufacturing reports for the RPV cover have also been covered in the write up of RO-ABWR-0003 later in this report. This section of my report covers generic aspects of these reports only.

240. Comments provided by my TSC (FNC), were categorised from Category 1 to 4 in a way similar to the topic reports discussed above.

4.2.11.2 CONCLUSIONS OF REVIEW

241. The documents reviewed all contained typographical errors, points of grammar, etc. These were not commented on unless they cause ambiguity, confusion, or lack of clarity. Hitachi-GE were advised to employ a proportionate approach in responding to comments, with most effort being focused on responding to the higher category (1-3) comments. In some cases, therefore, not all of the lowest Category 4 comments were responded to by Hitachi-GE. This is considered to be acceptable because Category 4 comments are only minor comments and points of clarification.

242. With regard to the RPV Design Philosophy Report (Reference 26), 0 Category 1 comments, 0 Category 2 comments, 5 Category 3 comments and 12 Category 4 comments have been raised. These were presented to Hitachi-GE at RQ-ABWR-1290. Hitachi-GE's responses were assessed and the amended version of the report reviewed (Revision 1). The responses to the queries were acceptable and have been appropriately incorporated into the report.

243. The review of the MS Piping Design Philosophy Report identified no significant (i.e. Category 1-3) comments. The 7 Category 4 comments generated were not formally issued to Hitachi-GE, but have been included in the TSC report (Reference 54) for information only, and to form an auditable record of the review.

244. For the Main Steam Piping Manufacturing Report (Reference 55), 0 Category 1 comments, 0 Category 2 comments, 7 Category 3 comments and 8 Category 4 comments have been raised in RQ-ABWR-1423 (Reference 56). Hitachi-GE's responses (via an amended version of the report) were reviewed. The responses to all of the Category 3 queries and 5 out of the 8 Category 4 queries were acceptable and have been appropriately incorporated into the report. Only three minor (Category 4) comments remain open, however, as stated above this is in keeping with advice to Hitachi-GE to prioritise higher category comments and does not represent a significant shortfall in the work.

245. From the sample of reports taken, I consider that the design and manufacturing reports provide a suitable overview of the manufacturability and inspectability of the components sampled. Notwithstanding that these documents are reviewed further under RO-ABWR-0003, I judge that they are suitable for GDA purposes.

4.2.12 ASME Design Calculations

246. The UK ABWR nuclear pressure systems are designed in accordance with the ASME III design code. This code compliance is the basis for demonstration that the components of the UK ABWR are capable of meeting UK safety expectations. I note that simple code compliance does not, in and of itself, demonstrate that the UK SAPs are met and the risks are maintained ALARP. Notably, UK expectations for components of the highest reliability, classified as VHI by Hitachi-GE, are that as part of an additional justification defect tolerance analysis should be performed in addition to code compliance calculations. This is covered in RO-ABWR-0001, and not in this section.

247. I performed this review of the ASME design by rule and design by analysis to ensure that code compliance could be demonstrated.

248. I considered it necessary, to replicate some of the design calculations to ensure that the code compliance can be demonstrated. I contracted my technical support contractor to perform reviews and comparative calculations of a sample of the ASME calculations performed. The TSC report is presented at Reference 57.

249. The documents reviewed, and their associated RQs, were:

   a. Stress Analysis and ASME III Assessments of the RPV Bottom Head (RQ-ABWR-1278) (Reference 58);

   b. Drywell Head Design Report; (RQ-ABWR-1360) (Reference 59)

   c. Stress Analysis Report of Control Rod Drive Housing Penetration for GDA; (RQ-ABWR-1381) (Reference 60)

   d. ASME Design Specification for MS and ASME Design Specification for FDW Piping (RQ-ABWR-1388) (Reference 61)

   e. Design Report for ASME Section III Class 1 Piping (RQ-ABWR-1507) (Reference 62)

   f. CRD Penetrations Design Justification Report (no comments raised)

   g. Equipment Hatch Design Report (no comments raised)

250. The components sampled include items that cover components VHI, Class 1 and elements of the metallic components from the RCCV. Components associated with the RCCV, specifically the equipment hatch design report and the drywell head design report, have been used to judge the adequacy of the RCCV metallic component design process.

251. The level of interpretation in the review of ASME calculations is low, the ASME code being of a prescriptive nature. Comments raised by the review were presented to Hitachi-GE via the formal RQ process. Notwithstanding the Assessment Findings raised below, these comments were generally of a clarification nature and the validity of the calculations presented was not, generally threatened. The details of the RQs raised are available through the FNC report.

252. All comments raised through the RQ process, including comments that were listed as “category 4” and therefore not required for completion of GDA, are considered closed

following receipt of the formal Hitachi-GE responses. The following points from the review of the Stress Analysis Report of the CRD Housing Penetration (Reference 3) and the Stress Analysis and ASME III Assessments of the RPV Bottom Head (Reference 1) should be noted:

253. From RQ-ABWR-1381 (Reference 60), comments 21 and 24 relate to the calculation of the range of stress intensity for fatigue assessments, where the directions of the principal stresses are assumed not to change between load cases. There are varying shear stresses present in the structure, so the assumption of constant principal stress directions was questioned. Hitachi-GE repeated the fatigue calculation accounting for varying principal stress directions, which showed for the bounding location the original method was conservative but that for other locations it was not. The assessment is accepted in this instance, as both methods produce Fatigue Usage Factors appropriately below 1. However, it is noted that the Hitachi-GE method has been shown to be non-conservative in some instances; hence fatigue usage factor calculation methodology changes should be considered where fatigue assessments show higher UFs.

AF-ABWR-SI-06 Because the fatigue usage calculation methodology used by Hitachi-GE is not demonstrably conservative for all cases, the licensee shall appropriately verify the fatigue usage factor calculations by other methodologies where margins to failure are low.

254. Hitachi-GE have provided a plan to address environmental fatigue, and an example of the application of this plan, at Reference 171. The purpose of that document is to describe the environmental fatigue evaluation procedure for UK ABWR components, and provide a sample evaluation as its demonstration. Hitachi-GE's methodology is based upon Regulatory Guide 1.207 (NUREG/CR-6909 Rev. 1)(Reference 172). In Reference 171, Hitachi-GE have performed representative analysis on the RPV bottom head and feedwater piping. This has shown an environmental fatigue usage factor (FUF_{en}) of, 0.189 and 0.268 respectively. I am content with the methodology proposed in so far as it has been applied in Reference 171. During the licensing phase I expect that this methodology will be applied to the plant more generally and the licensee will monitor developments in Regulatory Guide 1.207 and apply them appropriately.

255. Regarding RQ-ABWR-1278 Comment 13, where evidence was requested to support judgements made regarding axisymmetric modelling of asymmetric components, the argument provided was that the axisymmetric model results with additional stress concentration factors applied will be conservative of the real 3D stresses and that this will be investigated and proven in more detail in site licensing via the Shroud Support stress analysis report. The response was accepted on this basis; the licensee should seek to provide this extra detail as part of normal business during the licensing phase.

256. I judge that the design calculations performed by Hitachi-GE are acceptable for GDA purposes, that is, they demonstrate that there are no significant errors present that could threaten the Safety Case. The future licensee should, as part of the licensing and construction phases, confirm that the plant is compliant with the ASME code, as appropriate.

4.2.13 Spent Fuel Export – Structural Integrity considerations.

257. Hitachi-GE have proposed the use of a dry fuel storage system for the UK ABWR spent fuel management. The proposed method of export of this fuel is by first sealing the fuel into a canister in the region of the spent fuel pool, then exporting the filled canister with some form of over-pack protection, frequently referred to by Hitachi-GE as a cask.

258. The canister forms a metallic pressure boundary around the spent fuel within it; hence, it has been considered within the Structural Integrity GDA assessment.

259. Hitachi-GE do not propose to include the canister and cask detailed design within GDA. I judge that canister design is a developing area and that the first deployment of a canister of this sort may reasonably be expected to be at least a decade after first criticality. Moreover, through-life integrity of spent fuel canisters, outside of the nuclear island, is outside of GDA. I consider, therefore, Hitachi-GE's decision to exclude detailed design of canisters from GDA to be appropriate.

260. The spent fuel pool is located at the top of the reactor building, with the export route for sealed canisters at ground level. Hitachi-GE propose that the canisters/cask be lowered from the spent fuel pool area to ground level in a single stage utilising the reactor building crane. This leads to the potential for a drop from the crane onto the floor at ground level, this is termed the truck bay drop. Whether this decision is ALARP has not been considered by me within the Structural Integrity discipline, it has been considered within Mechanical Engineering and is discussed in that Step 4 report (Reference 169).

261. Hitachi-GE claim that the consequences of a drop event are mitigated by the use of an impact limiter. This impact limiter is integral to the floor below the main lifting area. A diagram from GA91-9201-0003-00436 (Reference 63) and is available through that reference. It shows the position of the impact limiter in relation to the potential truck bay drop. I judged that the drop onto this impact limiter could be considered as a bounding example of a cask/canister drop. For the purposes of GDA, I judged that if a concept design could be demonstrated to withstand this drop, then there would sufficient confidence that a solution would be possible at the time of commissioning.

262. From Hitachi-GE's safety submissions (GA91-9201-0003-00436 (Reference 63) and GA91-9201-0003-00917 (Reference 64)), the Safety Case presented was that an example canister could withstand a maximum impact loading of 60g without failing, and that the impact limiter would maintain loads to at or below this level with a reliability of 10^{-4} per demand. I note that the impact limiter is also presented at a concept level, rather than detailed design.

263. Both documents provided a justification through reference to supporting documents. There was no narrative discussion of how the information in the references informed the asserted reliability of the impact limiter or of the drop withstand of the canister. These claims formed an integral part of the spent fuel export Safety Case, without which it could not be demonstrated that the proposed option met the reliability limits. It was essential, therefore, that Hitachi-GE demonstrate the source of their confidence in these figures. I raised RQ-ABWR-0980 (Reference 65) to gain this confidence.

264. The queries within RQ-ABWR-0980 were

1. How do the references and supporting documents given in Attachment D to GA91-9201-0003-00436 support the claim of a minimum impact limiter reliability of 10^{-4}. This should include a discussion of the origin of the number claimed and a description of how this was arrived at.
2. How do the references and supporting documents given GA91-9201-0003-00917 support the claim of a maximum deceleration of 60g being acceptable? This should include a discussion of the origin of the number claimed and a description of how this was arrived at.

265. In their response to these queries, Hitachi-GE have given further reference to the operational experience with plastic foam impact limiters within the nuclear industry.

They also note that the design, manufacture, testing and maintenance will be performed according to internationally recognised standards. Hitachi-GE gave, in outline, the results of modelling and testing work that has been performed on materials of this type. Hitachi-GE further state *"The claim of 10^{-4} pfd is therefore not derived through any failure rate analyses, rather an engineering judgement based on the available guidance and reliability of high integrity passive systems"*. I consulted with the ONR Fault Studies inspector, who concurred that a reliability of 10^{-4} pfd is feasible for actuation of a passive system such as an impact limiter. Given the supporting evidence presented in GA91-9201-0003-00436 (Reference 63) on materials performance, I concur with this analysis. A robust demonstration that these reliabilities can be achieved in practice will, however, be necessary during the licensing phase. This will only be possible once the cask/canister and impact limiter technology has been selected.

266. The other underpinning assumption within Hitachi-GE's analysis was that an acceleration of 60g would not lead to failure of the canister pressure boundary. In Reference 64, Hitachi-GE provided a discussion on where this number had originated and how it related to the concept design of canister, which was being used as a concept design. Hitachi-GE provided further information that there are alternative canisters available that have been analysed as being capable of withstanding 75g without failing the pressure boundary. On this basis I am content that there is a concept design of canister that can withstand 60g.

267. I am, therefore, content that the case has been made that, in concept, a canister can be dropped onto an impact limiter from the maximum height of the truck bay and the pressure boundary will not be compromised, with a reliability of 10^{-4} pfd. This will require substantiation within the site licensing phase as part of normal business.

268. The above reliability contains the implicit assumption that the cask/canister will land upon the impact limiter and not onto any surrounding unyielding surfaces. To have confidence that the impact limiter is capable of performing its function, ONR needs a demonstration that, for credible drops, this will be the case. I raised RQ-ABWR-0945, which was intended to gain this confidence. The query in the RQ was:

1. Hitachi-GE should provide evidence that, for credible limiting drops, the canister will land on the impact limiter and not on the surrounding Civil structures. Limiting drops should include, but should not be confined to, the full height truck bay drop.

269. Hitachi-GE responded with GA91-9201-0003-01583 (Reference 67). This provided an overview of potential bounding drops and stated:

"It has been shown that for a representative range of credible drops into the hoist well the cask will land on the impact limiter. In the unlikely event of ledging faults, there is a potential for glancing blows against the surrounding civil structures but the magnitude of these is insufficient to challenge the integrity of the canister."

270. The submission by Hitachi-GE does not contain a full set of potential drop faults. I consider, however, that these are a suitable sample of bounding cases, fit for the purpose of confirming that a dropped cask/canister will land on the impact limiter. The possibility of glancing blows against surrounding civil structures will require substantiation within the site licensing phase.

271. The licensee will need to ensure that the spent fuel canister technology meets the specific requirements and constraints of the analyses performed in the GDA process. Hence, the licensee shall put in place a process to select a spent fuel interim storage technology and demonstrate that it is capable of withstanding the most limiting drops from the GDA assessment, balancing this against the need to demonstrate through-life integrity. This will be progressed by ONR though normal business. This has been

captured as a commitment by Hitachi-GE and will be passed formally to the licensee following GDA.

4.3 Regulatory Issues

272. Regulatory Issues (RIs) are matters that ONR judge to represent a 'significant safety shortfall' in the safety case or design and are the most serious regulatory concerns. RIs are required to be addressed before a DAC can be issued.

273. I have not raised any Regulatory Issues during my assessment of Structural Integrity of the UK ABWR.

4.4 Regulatory Observations

274. Regulatory Observations (ROs) are raised when ONR identifies a potential regulatory shortfall which requires action and new work by the RP for it to be resolved. Each RO can have several associated actions.

275. A summary of ROs related to Structural Integrity can be found in Annex 4

276. The four initial ROs raised within the Structural Integrity discipline were general in nature and related to shortfalls in the structure presented during step 2 of GDA:

277. RO-ABWR-0001 was intended to outline ONR's expectations in terms of safety case content for fracture assessment and inspection. This formed a substantive part of the requesting party's Safety Case, hence RO-ABWR-0001 should be considered as an integral part of the ONR assessment for the overall Safety Case and not as a discrete RO.

278. RO-ABWR-0002 relates to the Structural Integrity of the Control Rod drive mechanisms, which are mounted as bottom-penetrations of the RPV. The number and density of such penetrations is a novel design feature for a light-water reactor in the UK. The design and inspection of these features was, therefore, captured in this RO.

279. RO-ABWR-0003 relates to the design of the RPV. This captures the requirement for nozzle design to be ALARP, as well as the top head design of the RPV, which was proposed to be made from plate materials, and not forgings. It is not immediately obvious that this is aligned with the UK expectation that the number and length of welds, especially in components of the highest reliability, should be minimised.

280. RO-ABWR-0004 relates to the materials used in the main pressure boundary components of the highest reliability, and the demonstration that the choices made are ALARP.

281. The other Structural Integrity specific ROs were reactive to Hitachi-GE submissions. These addressed shortfalls in the materials presented during Steps 3 and 4 and not a shortfall in the structure.

282. RO-ABWR-0035 relates to the materials selection process for all of the plant, whether this selection process is robust and whether or not it can be shown to maintain the risk of a failure of the pressure boundary as ALARP, as well as whether the dose burden arising from the materials choices was ALARP. This RO was led by the Reactor Chemistry discipline, and was a joint RO between Reactor Chemistry and Structural Integrity.

283. RO-ABWR-0059 relates to the design and build of the Reinforced Concrete Containment Vessel (RCCV) head. The RCCV is a composite structure and managed jointly between Civil Engineering and Structural Integrity. The RCCV head is a metallic structure that would be flooded in the case of a severe accident.

284. I also considered aspects of RO-ABWR-0034, which relates to the Bottom Drain Line (BDL). Although this was not initiated by Structural Integrity, the resolution of this matter included significant input from the Structural Integrity assessors. Structural Integrity aspects are, therefore, considered below.

4.4.1 RO-ABWR-0001 – Avoidance of Fracture

4.4.1.1 BACKGROUND & SCOPE

285. RO-ABWR-0001 was raised at GDA Step 2 to provide guidance to Hitachi-GE in meeting the UK regulatory expectations relating to their avoidance of fracture demonstration. In UK civil nuclear practice an avoidance of fracture demonstration is one of several additional measures beyond compliance with an established nuclear design and construction code to discount gross failure for the highest reliability components.

286. ONR's expectation for highest reliability components is that the component or structure should be as defect free as possible and is demonstrated to be tolerant of defects (SAPs EMC.1). In particular the limiting defect size needs to be shown to be larger than the defect size that can be reliably detected by the applied examination techniques. This is provided through an avoidance of fracture demonstration, which involves a detailed fracture mechanics based defect tolerance assessment (DTA), using conservative material properties, to determine the limiting defect sizes at the start of life taking into account any potential for through-life crack growth. For the GDA the proposed non-destructive examinations for the components during manufacture then need to be capable of qualification to assure high reliability in the detection of such start of life defects with an adequate margin.

287. Hitachi-GE outlined their strategy for their avoidance of fracture demonstration in the RO-ABWR-0001 resolution plan (Reference 71). At GDA Step2, ONR concluded that Hitachi-GE's proposed approach provided a sound basis for an avoidance of fracture demonstration for the purposes of the GDA. Hitachi-GE committed to provide the arguments and evidence in sequenced responses in GDA Steps 3 and 4.

288. For the UK ABWR, Hitachi-GE designates highest reliability components as very high integrity (VHI) they include: the RPV; the section of the main steam pipework from the RPV to the MSIV (inboard of containment); and the inboard MSIV (Reference 7). In addition, for their avoidance of fracture demonstration, highest reliability includes SSCs designated by Hitachi-GE as high integrity (HI). However, for the GDA, Hitachi-GE confirmed that there were design provisions in the UK ABWR to avoid the need to invoke a HI claim (Section 4.4.2).

4.4.1.2 BASIS OF ASSESSMENT: PRINCIPAL RP SUBMISSIONS

289. Hitachi-GE provided extensive documented evidence for RO-ABWR-R0001. The principal submissions were identified in the Resolution Plan published on the ONR website:

- The materials property handbook (Reference 72 and Reference 73).
- The material testing strategy (Reference 74, 75 and 76).
- The structural integrity classification report (Reference 7),
- The weld ranking procedure (Reference 78)
- The weld ranking application report (Reference 79)
- The defect tolerance assessment plan (Reference 80)
- The inspection assessment plan report (Reference 81)
- The Inspection Qualification Strategy report (Reference 82)
- Design Philosophy Report for RPV (Reference 83)
- The reconciliation report (Reference 84)

4.4.1.3 ASSESSMENT OF RO-ABWR-0001

Standards & Criteria

290. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (Annex 1) internal TAGs (Annex2), relevant standards and RGP informed by past GDAs and existing practices adopted on UK nuclear licensed sites.

291. The SAP relevant to the assessment of RO-ABWR-0001 include EMC.1 to EMC.34 with EMC.1 to EMC.3 having specific relevance to highest reliability claims; EAD.1 to EAD.4 on Ageing and Degradation; ESC.1 to ECS.3 covering Safety Classification. Key SAP and TAG guidance that informed the assessment of RO-ABWR-0001 are discussed further below.

Scope of Assessment for RO-ABWR-0001

292. The scope of my assessment has included submissions by Hitachi-GE as evidence for their avoidance of fracture demonstration. The principal submissions provided by Hitachi-GE are listed in Section 3; several were updated in the course of my assessment. I review the evidence provided by Hitachi-GE in response to the RO-ABWR-0001 actions below.

293. I raised a series of questions in the course of my assessment and subsequently assessed responses by Hitachi-GE to my Regulatory Queries (RQ).

294. Through routine Level 4 engagement I discussed:

- the UK regulatory expectations, based on RGP;
- technical and safety aspects of each Hitachi-GE submission.

295. My assessment was conducted with TSC support; Frazer-Nash Consultancy Limited, provided independent expert advice on the Hitachi-GE defect tolerance assessment methods and their application, reported in Reference 85.

296. My focus for the assessment of the avoidance of fracture demonstration was to establish whether Hitachi-GE applied RGP and whether Hitachi-GE could demonstrate that they had an understanding of UK expectations and could satisfy them:

- in its work to substantiate the material properties for its fracture assessments, in particular, minimum fracture toughness values, along with its proposals for future materials properties testing to validate assumptions in its DTA;
- in its fracture assessments, and whether the outcome of analysis was acceptable and substantiated;
- in its proposals to demonstrate highly reliable NDT during manufacture that is assured through future inspection qualification and other measures such as repeat inspection; and
- to consider the strength of the overall fracture avoidance demonstration for VHI components by examination of the safety margins and the reconciliation of the material properties, DTA and the proposals for qualification of the manufacturing inspections.

297. For the GDA the expectation is that RPs' proposals provide the basis for highly reliable NDE during manufacture that is assured through inspection qualification and other measures e.g. repeat inspection. Hitachi-GE used preliminary inspection qualification technical justifications as a means to demonstrate an adequate level of confidence that objective based inspections can be developed and qualified. However, completion of

formal inspection qualification is a licensing matter and not within the scope of the GDA.

298. In addition, for the GDA, ISI defects were, in general, outside the scope of the fracture avoidance demonstration. An exception was nozzle crotch corner features in forgings. Notably, for these non-weld locations although the most challenging defect orientations for the DTA were more likely to be service-induced, Hitachi-GE chose to assess them within the GDA. This scope is fit for purpose for the GDA, however, my expectations are for a wider range of defect locations to be considered for in-service.

Assessment Structure

299. This section of the report is structured in four parts, which align with the Regulatory Observation Actions (ROAs) of RO-ABWR-0001:

- Material Properties
- Defect Tolerance Assessment
- Manufacturing Inspection (or examination)
- Overall Avoidance of Fracture Demonstration (or Reconciliation)

300. The term ‘examination’ refers to the non-destructive inspection system (techniques, procedures and operators) and for convenience is used interchangeably with ‘inspection’ unless otherwise stated. I discuss the above in turn below. For each topic I precis the relevant ROAs and outline:

- my assessment approach/sampling strategy;
- my RQs and the Hitachi-GE responses: and
- present summaries with my main conclusions.

301. The summaries for each topic inform my key regulatory judgments and outcomes relating to RO-ABWR-0001.

Materials Properties

302. Lower bound fracture toughness properties, other materials properties including strength data along with an allowance for through-life fatigue crack growth behaviour are required as input to the DTAs. In addition for the GDA, a strategy for undertaking fracture toughness testing on relevant material as part of the manufacturing process to underpin the minimum toughness values used in the DTAs is expected. These expectations were presented as Actions ROA-RO-ABWR-0001.A1.1 (Material Properties) and ROA-RO-ABWR-0001.A1.2 (Material Properties) respectively.

303. My sampling strategy was governed by my expectation that the material property reports should include a robust demonstration that the materials properties in the DTA are appropriate and that the finished product has suitable testing to show that the materials properties are known or the component can perform its safety function throughout.

304. In response to ROA-RO-ABWR-0001.A1.1, Hitachi-GE provided a compilation of material property data for use in the DTAs in materials property handbooks for the VHI components MSIV (Reference 72 and Reference 73). Hitachi-GE’s intention is that these are ‘live’ documents which will be repositories for all related materials data, including the as-built properties.

305. The material property handbooks are supplemented by detailed material reports for VHI and non-VHI components, which describe the basis of the material selections and the material specifications in accordance with ASME Section II. The material selections for the UK ABWR are discussed in Section 4.4.4 under RO-ABWR-0004

and section 4.4.6 for RO-ABWR-0035). The combination of the material property handbooks and detailed material property reports is intended to support the development of a comprehensive material data set and rationale for the materials choice in the structural integrity case for the UK ABWR.

306. The material property handbooks cover all those components classified by Hitachi-GE as VHI, and so include the RPV, MS Piping (inside the RCCV) and the MSIVs inboard of the RCCV. In addition, as Hitachi-GE now classify multiple RPV stud failure as VHI, I expect the inclusion of material property data for the RPV studs including strength and fracture toughness. The extant edition of the material property will therefore need updating to include relevant as-manufactured RPV stud data. The licensee will need to maintain and update their material property handbooks in support of future requirements; it is my expectation that ONR inspectors will pursue this during licensing through normal business. Given the high level of structural integrity claimed, I view this information as important to the development of the safety case and so raise as a minor shortfall to capture my expectation.

***MS-ABWR-SI-02 - The licensee should ensure that all relevant materials properties data to underpin the plant safety case, including that for multiple RPV stud failure, is included in an up-to-date materials properties handbook*.**

307. Similarly, multiple support failures in VHI components needs to be classified and that an appropriate structural integrity case is developed. I am not aware that Hitachi-GE have considered this point and do not consider this to be a significant threat to the safety case provided, therefore, I raise a minor shortfall for the future licensee to address.

MS-ABWR-SI-03 - The licensee shall ensure that consequences of multiple support failures for VHI components are classified and that an appropriate structural integrity case is developed.

308. Hitachi-GE performed fracture toughness tests to ASTM E-1820 (Reference 86) and carried out a statistical analysis to confirm the fracture toughness properties for the RPV beltline materials in the upper shelf region. The tests were performed on Japanese RPV steels which closely matched the chemical compositions, tensile properties and grain size of the RPV steels proposed for UK-ABWR (SA-508M Grade 3 Class 1(forging) and SA-533M Type B Class 1 (plate). The fracture toughness K_{Jc} was estimated from measured J_{Ic} toughness values and also included weld and HAZ materials. In undertaking fracture toughness testing within GDA, Hitachi-GE have mitigated the risks for the future licensee and aimed to achieve ‘best practice’ in the provision of evidence to cover the material testing aspects of RO-ABWR -0001.

309. For the forging and plate materials the $\mu - 2\sigma$ (95% confidence) lower bound value of 264 MPa√m was obtained. However, for the weld metal a minimum $\mu - 2\sigma$ value of 215MPa√m was estimated. The value of 215MPa√m represents a confidence interval of $\mu - 1.75\sigma$ (92% confidence) for the weld material. This notwithstanding, Hitachi-GE proposed using a fracture toughness value of 220MPa√m for the RPV weld metal in their DTAs.

310. I understand that an upper shelf fracture toughness of 220 MPa√m is consistent with general practice, for example in the United States, the value is referenced back to ASME XI Flaw Evaluation Procedures, (Reference 87). However, this value is higher than the values quoted in more recent European Nuclear Pressure Vessel design codes, for example the Annex ZG of the French RCC-M code, quotes an upper shelf toughness for low alloy steel weld of 170MPa√m above 200°C (Reference 88).

311. The use of an upper shelf fracture toughness of 220 MPa√m for the RPV weld metal without the support of adequate and sufficient data would not meet my expectations if

used in the final DTA for product data (in line with ONR's SAPs). However, consistent with the approach adopted in previous GDAs, I was content for Hitachi-GE to use a value of 220MPa√m for their RPV DTAs on the basis that the future licensee will either underpin the lower bound upper shelf value by undertaking additional fracture toughness testing on representative weld metal or justify the use of a lower value in the RPV DTAs. This represents a risk for Hitachi-GE, during manufacture, if reconciliation with the DTAs, places undue demands on qualified inspection. I review the position further, taking cognisance of the conservatism in the DTAs and claimed DSMs, as part of the reconciliation later in this report, but raise assessment finding AF-ABWR-SI-07 below.

312. Similarly, for the MSP and MSIV the material properties proposed for the DTA were presented in a specific material property handbook (Reference 73). Hitachi-GE proposed lower bound upper shelf fracture toughness values of 215MPa√m for the MSIV. The fracture toughness values were obtained from a combination of testing and recognised sources, and I was generally satisfied with the values that have been assumed for the purposes of the GDA. Notably, the 215MPa√m was obtained from specific tests of the valve parent and weld metals (Reference 173). I accept these values for the GDA, but my view is that the achievement of a value of 215MPa√m is challenging for casting material. As with the RPV weld metal I view this as a commercial risk for Hitachi-GE. I consider the position further under the reconciliation part of this report below, but raise assessment finding AF-ABWR-SI-07 below.

313. It is also noteworthy that all materials selected for VHI components are expected to remain on the upper shelf of the fracture toughness curve through-out the plant operating temperature range. I view this as a significant safety benefit which accords with meeting SAP EMC.23. An important aspect covered in the material property handbook includes the allowance for through-life degradation. This includes, for example, the effects of thermal aging, irradiation embrittlement, mechanical property changes and fatigue crack growth behaviour. I reviewed the basis of Hitachi-GE's justification of their allowance for thermal ageing, mechanical property degradation and fatigue crack growth. Hitachi-GE provided adequate evidence to justify their approaches. In particular, Hitachi-GE confirmed that no VHI components are expected to be subject to a stress corrosion component for fatigue crack growth. Indeed, Hitachi-GE clarified that their FCG calculations were sufficiently conservative (Reference 89). This response is considered sufficient for GDA, but I recommend that, as part of normal business, the licensee should assure themselves that the FCG calculations are suitably conservative.

314. For the fracture avoidance demonstration the focus on my assessment was Hitachi-GEs proposals to account for the effects of irradiation embrittlement at the RPV beltline. During Structural Integrity Level 4 meetings, Hitachi-GE indicated that they intend to use the Japanese national code JEAC 4201 to assess irradiation embrittlement of pressure vessel steels. This code is only available in Japanese. It was, therefore not possible for ONR to directly assess whether this meets expectations or not. I note that Hitachi-GE provided supporting information on this standard, but not a full translation. Whilst ONR does not mandate the use of any specific code, my expectation is that I should be able to compare the work done directly with examples of international relevant good practice.

315. The UK ABWR is designed against the ASME code; I judge that the irradiation embrittlement curves from the US standard NUREG/CR 6551 are suitable for a comparison to be made. Hitachi-GE provided this comparison in response to RQ-ABWR-1384.

316. In (Reference 90), I judge that Hitachi-GE have a good understanding of both the US and Japanese standard practices. From the data presented, I judge that the JEAC 4201 standard is significantly more conservative than the corresponding NUREG

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standard. Whilst ONR will always require fully representative samples for irradiation embrittlement management, as per EAD.3 of the ONR SAPs, I judge that the Hitachi-GE's position is satisfactory for GDA.

317. I note that the fracture toughness energies predicted differ significantly between the JSME and ASME predictions. As part of normal business during the licensing phase, the licensee should confirm whether the irradiation embrittlement figures are reasonable and, hence, whether the expectations being put on the inspection performance are also reasonable.

318. The estimated upper shelf fracture toughness values of K_{JC} for forging and weld metal were 244 and 195MPa√m respectively. For expediency, Hitachi-GE used a fracture toughness value of 190MPa√m in their DTA. I view this as a reasonable estimate subject to confirmation of the start of life properties and achievement of the material composition. I also note that for the UK ABWR the expected reduction in the upper shelf fracture toughness through-life is low circa 15% assuming an initial upper shelf fracture toughness of 220MPa√m

319. I cover Hitachi- GE RPV surveillance programme and their material archiving strategy under RO-ABWR-0004 (Section 4.4.4).

320. The material property handbooks are adequate to support the DTAs for the purposes of GDA, but post GDA there is a need for the Licensee to produce a comprehensive material data set for use during the design and assessment process and also to support through life operation. This will need to cover all relevant data including the basic design data and the confirmatory batch and weld specific test data from the complementary additional toughness testing programme. I take this forward as assessment finding **AF-ABWR-SI-07:**

321. AF-ABWR-SI-07 – Because some example toughness values produced by Hitachi-GE during GDA are below the values used within the Defect tolerance analysis, the assumptions used within the analyses must be justified within licensing. The Licensee shall, therefore, underpin the lower-bound upper-shelf materials toughness value for all Very High Integrity components and materials using representative testing, or justify the use of a lower value in their Defect Tolerance Analyses.

More generally, materials data, across all classifications of component underpins the Structural Integrity safety cases. Confidence is needed, therefore, across a wide range of materials data, for all classes of component. The Licensee shall produce a comprehensive and appropriately justified material data set for the VHI, Class 1 and Class 2 components for use during the design and assessment process and also to support through life operation. This will need to cover all relevant data including the basic design data and the confirmatory batch and weld specific test data from the additional fracture toughness testing programme. It will need to be clearly presented such that the initial pedigree of the data can be traced following the literature trail with comparison to other international data sets where possible.

322. Hitachi-GE recognised the need to perform additional material testing for VHI components and submitted their fracture toughness testing strategy in response to ROA-RO-ABWR-0001.A1.2 (Material Properties) (Reference 91). I view confidence in the appropriateness of fracture toughness testing and derived minimum fracture toughness values as important to the fracture avoidance demonstration. However, following my assessment of Reference 91 and discussions with Hitachi-GE, three key points emerged relating to their fracture toughness testing proposals: the sampling directions; the use of a stretch zone method; and the need for fully representative

testing to establish the fracture toughness of the MSIV bodies. I discuss these points next.

323. As part of an avoidance of fracture demonstration for a component of the Highest Reliability's Safety Case, it is ONR's expectation that materials properties are characterised. This includes the fracture toughness, which is a vital input into the DTA. Hitachi-GE chose to sample the T-L direction (detailed below), which is neither the direction of highest stress, nor the lowest expected materials properties. I did not find Hitachi-GE's reasoning why data from specimens in this orientation convincing. This cast into question the validity of the DTA performed by Hitachi-GE. An illustration of the different test directions is provided in Figure 1 below.

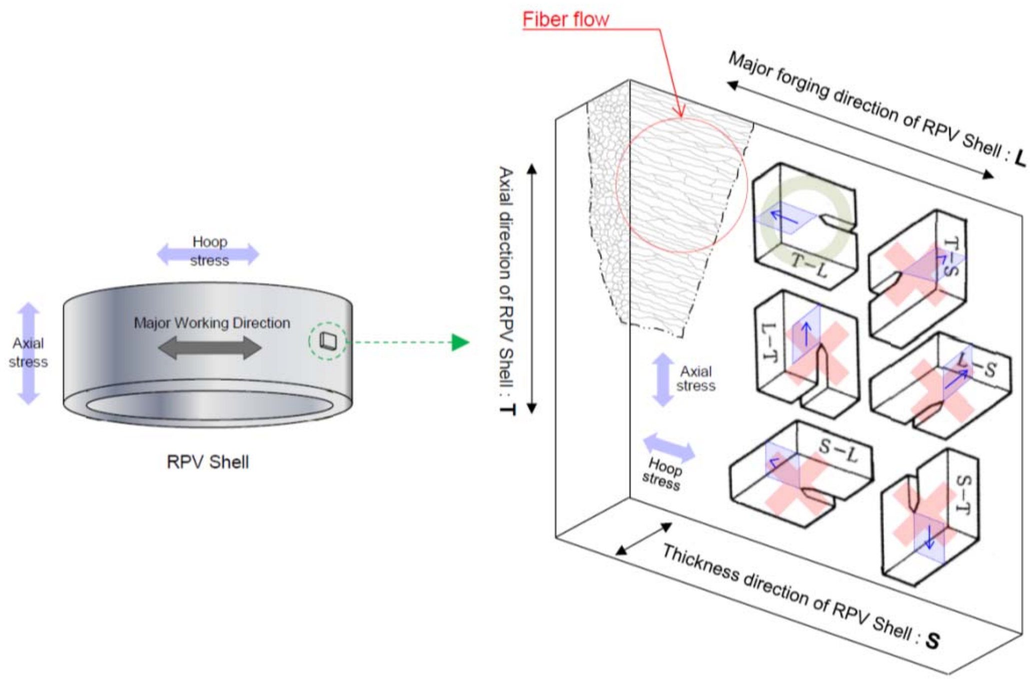

Figure 2: Definitions of sampling directions for fracture toughness testing of cylindrical components, (Reference 152), taken from GA91-9201-0003-01951 - RD-GD-0047 - Rev 0 - Response to RQ on the Direction of Fracture Toughness Test Specimens for RPV Materials.

324. During Level 4 interactions, Hitachi-GE presented information to show that there is no significant variation in fracture toughness expected with specimen orientation for the RPV shell forging tested and is inferred to be applicable to all similar forgings. The level of scatter in fracture toughness data is high. Hence, if there is no significant difference between fracture toughness properties with orientation for the materials in question, I judge that Hitachi-GE's proposed test direction will be adequate. I raised RQ-ABWR-1246 to gather this assurance.

325. Hitachi-GE presented data to show that the Charpy impact toughness does not vary significantly with orientation. Charpy impact data is not analogous to fracture toughness testing, but Charpy energy and fracture toughness do vary directly with one another. Hence, I judge that the Hitachi-GE proposals for fracture toughness testing, in so far as testing direction is concerned, is adequate to support the DTA.

326. Hitachi-GE also presented proposals for the fracture toughness testing of the MS Piping and MSIV (Reference 92). The MSP and MSIV component thicknesses were insufficient to achieve a valid test under ASTM E1820 (Reference 86). Hitachi-GE therefore proposed the use of J_Q in place of $J_{IC.}$ This was a fracture toughness

determination based on measurement of the stretch zone width. This methodology infers plasticity, and hence fracture toughness, from an examination of the plastic stretch zone at the crack tip of a fracture toughness specimen.

327. I was unfamiliar with this approach and I asked Hitachi-GE to provide further evidence to understand its basis and justify its application in a UK safety case (Reference 93). I also sought expert advice from TWI (Reference 94). Subsequently, Hitachi-GE confirmed that they will not be using stretch zone width to derive fracture toughness and removed it from their submissions.

328. In discussions with Hitachi-GE a further point emerged relating to their fracture toughness testing strategy for the MSIV castings. Notably, Hitachi-GE's procurement arrangements were such that a different supplier may be used to that chosen for the pre-production qualification of the MSIV manufacture, which included fracture toughness testing. This proposal did not meet my expectations with regard to the assurance that the subsequent material testing would be fully representative for a highest reliability component (SAP EAD.03). I cover this point in more detail in Section 4.2.7; it will suffice to say that Hitachi-GE subsequently resolved the matter by committing to undertake equipment qualification, including fracture toughness testing, on production MSIV castings.

329. Overall, following resolution of my RQs and discussions I was satisfied with Hitachi-GE's fracture toughness testing strategy for VHI components.

Summary: RO-ABWR-0001 Material Properties

330. For closure of RO-A BWR-0001 my expectation is that the RP provides adequate and sufficient evidence that the materials properties in the DTA are appropriate and that the finished product has suitable testing to show that the materials properties are known or bounded for the regions of greatest safety interest.

331. I sampled Hitachi-GE's minimum toughness values and conclude these are suitable for their DTA. My conclusion is based on:

- The results of fracture toughness testing
- The derivation of appropriate lower toughness bounds along with commitments if needed from Hitachi-GE to justify the values used in their DTAs.
- A conservative allowance for the effects of irradiation embrittlement.

332. I also conclude that Hitachi-GE's fracture toughness testing strategy for VHI components is adequate. My conclusion is based on:

- Identification of the specimen orientation most likely to provide the limiting toughness
- A commitment to undertake representative testing for all highest reliability components.
- The implementation of recognised RGP.

333. Overall, I am satisfied that Actions ROA-RO-ABWR-0001.A1.1 (Material Properties) and ROA-RO-ABWR-0001.A1.2 (Material Properties) can be closed and that the Hitachi-GE fracture toughness values and testing strategy are adequate for purpose of the GDA.

334. There are some specific points where further work is needed by the future licensee. To ensure that these points are satisfactorily resolved I have raised assessment findings under the materials section.

Defect Tolerance Assessment

335. The Hitachi-GE approach for their DTAs was based on the R6 Defect Assessment Procedure (Reference 95). 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 was satisfied with the choice of this procedure as the basis for the fracture mechanics assessment.

336. The R6 procedure is based on a Failure Assessment Diagram (FAD), which illustrates proximity to failure, and indicates the predicted failure mode (Figure 2). The vertical axis of the FAD (Kr) represents the ratio of applied stress intensity factor to the fracture toughness of the material. This provides a measure of the proximity to failure by fracture. The horizontal axis (Lr) represents the ratio of the applied load to the load required to cause plastic collapse of the section containing the postulated flaw. This provides a measure of the proximity to failure by loss of material strength. The interaction between the two failure modes is represented by the failure assessment line, which is established from the tensile properties. The proximity to failure for given defect size is represented by a locus of assessment points through to a limiting defect size (Figure 2).

337. Hitachi-GE designate the limiting through-wall defects as the End of Life Limiting Defect Size (ELLDS); the crack size that can be detected and sized with high confidence as the Qualified Examination Defect Size (QEDS); the through life fatigue crack growth as the Lifetime Fatigue Crack Growth (LFCG), and the margin between the ELLDS and the QEDS plus LFCG is termed the Defect Size Margin (DSM). The Hitachi-GE approach accords with the expectations in RO-A BWR-0001, provided the ELLDS divided by the sum of the QEDS and Lifetime Fatigue Crack Growth (LFCG) exceed the target DSM of 2.0, (Reference 80):

DSM = ELLDS/(QEDS + LFCG)

338. In practice, Hitachi-GE developed their NDT technical justifications in parallel with their DTAs. The start of life limiting defect size (SLLDS) i.e. the through-wall extent of a defect to give a defect size margin (DSM) of 2.0 at the end of life was calculated to provide a target for the development of their NDT technical justifications. I note that any additional margin between the QEDS and SLLDS provides a DSM > 2.0. However, to ensure that reasonable demands were placed on the future qualification of the NDT systems I confirmed with Hitachi-GE that the achievement of a DSM of 2.0 was the priority (Reference 96).

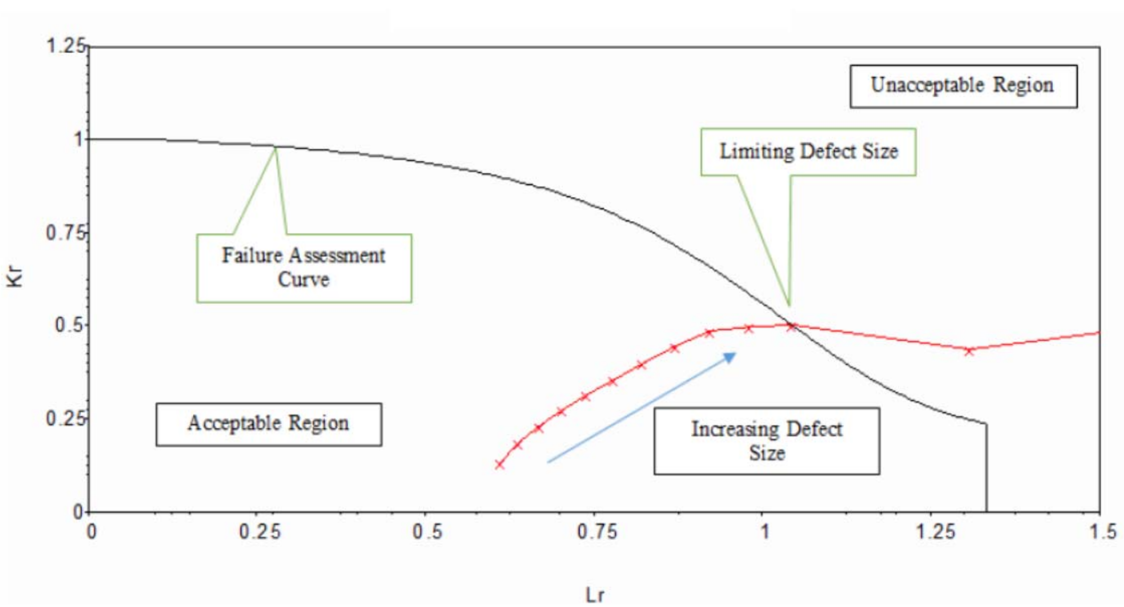

Figure 3: Example of R6 Failure Assessment Diagram

339. Hitachi-GE applied the software based implementation of the R6 procedure, R-Code developed by EDF Energy to undertake the limiting defect size calculations (Reference 97). I note that this code is used widely for analysing defect tolerance within the nuclear community, but that it is ONR's expectations that any software used is properly validated and verified. I note that Hitachi-GE used a UK contractor to aid them with their analyses. Where appropriate the stress distributions were taken from existing stress analyses. Residual stresses values and distributions were based on Section II.7.6.1 of R6 or measurements. I am content that these sources of data are suitable for GDA purposes, but these assumptions should be checked during the licensing phase as part of normal business. Recognised stress intensity factor solutions and plastic collapse solutions from R-Code were then used to undertake the limiting defect size calculation. All the DTAs were based on this standard approach.

340. To calculate the SLLDS, either a reverse fatigue crack growth calculation can be performed using the End of Life Limiting Defect Size (ELLDS) as the starting point, or alternatively the DSM for a range of start of life defect sizes can be evaluated (forward fatigue crack growth). Hitachi-GE used the forward fatigue crack growth approach. In principle, both reverse and forward approaches provide an adequate means for calculating FCG the purposes of a fracture avoidance demonstration, provided there is appropriate conservatism in the assumptions and that the results are underpinned, where appropriate, by sensitivity studies, for example transient ordering and sequence. This should be confirmed by the licensee as part of normal business.

341. Hitachi-GE used Paris Law FCG equations taking cognisance of the environments given in the American Society of Mechanical Engineers (ASME) Code Section XI (Reference 98). The transients were applied in sequence based on the total number of transients specified for the 60-year design life.

342. At GDA Step 2, ONR was generally content with the Hitachi-GE DTA proposals and the use of R-Code to implement the R6 procedure for calculating the limiting defects defect sizes for a sample of the VHI welds. However, at Step 4, specific aspects of the Hitachi-GE DTA proposals including those for FCG were raised in RQs and are discussed below.

DTA Assessment Strategy

343. There are two actions under RO-A BWR-0001 relating to DTA: Action ROA-RO-ABWR-0001.A2.1 (Fracture Assessment) covers the identification of the limiting locations for each component requiring a highest reliability claim; Action ROA-RO-ABWR-0001.A2.2 (Fracture Assessment) includes establishing the limiting defect sizes for the limiting locations using elastic-plastic fracture mechanics methods with bounding transients and estimates of FCG through life.

344. My assessment approach for DTA considered Hitachi-GE's weld ranking procedure and its application to identify a set of limiting weld locations for the GDA. In addition, my view of Hitachi-GE's proficiency in using the elastic-plastic R6 procedure was informed through reviews of their generic methodologies and the application of R6 to specific VHI welds. I also considered the effectiveness of Hitachi-GE's verification arrangements and their proposals for reconciliation.

DTA Methodologies and Verification

345. Hitachi-GE's verification process was detailed in Reference 80. At GDA Step 3 their process involved using the RPV fabricator as the main contractor to perform the analysis with input from AMEC Foster Wheeler as an advisor. The use of a UK contactor was necessary because the R6 methodology is novel in Japan. A separate department within AMEC Foster Wheeler (Independent Verification Body (IVB)) was

commissioned by Hitachi-GE to perform an independent analysis, which could involve full independent verification of the DTA results.

346. Hitachi-GE also presented proposals for reconciliation between the DTA and NDE for the GDA and future site licensing (Reference 80). For Step 4 the Hitachi-GE proposals supplemented the procedure verification via separate DTAs at Step 3 with comparison discussions, and expert panels with independent representation prior to reconciliation between the Hitachi-GE DTA and NDE teams.

347. I noted that Hitachi-GE's independence protocol included full IVB autonomy in the calculation assumptions, choice of SIF and plastic collapse solutions. Indeed, for several VHI locations IVB used their own FE models to calculate stresses e.g. the RPV shell weld, MS piping and MSIV whereas for other DTAs, Hitachi-GE provided some input information.

348. In principle, Hitachi-GE proposals for DTA verification during GDA were logical, robust and included a significant level of independent challenge.

349. I sampled Hitachi-GE's verification arrangements for their initial DTA of the RPV shell weld local to the support skirt through RQ-ABWR- 0357 (Reference 99 and Reference 100). For this location the IVB undertook a separate independent DTA. The results of Hitachi GE and IVB analyses diverged by a factor of approaching 2, with the IVB giving the most conservative answer (Reference 101).

350. Subsequent re-analysis showed that the reason for the discrepancy was a difference in assumptions employed by the analysts. For example the RPV manufacturer's analysis was a simplified analysis of the shell, not considering the presence of the support skirt. The IVB analysis was a more sophisticated model which included the presence of the support skirt.

351. Hitachi-GE presented good evidence to show that the IVB DTA was excessively conservative and that the combination of materials properties and stressing was not reasonable. When the same input assumptions used by IVB were used by Hitachi-GE, the answers produced were consistent. I was content that the answers achieved by Hitachi-GE remained demonstrably conservative for the given input assumptions.

352. I judged that this was a good example of the effectiveness of the Hitachi-GE verification process for the GDA. I gained significant confidence in the robustness of the process employed by Hitachi-GE and the independence of their verification process for the DTA during GDA. In view of this high level of confidence I reserved the commissioning of ONR comparative DTA work until GDA Step 4. I commend Hitachi-GE for the rigour of their DTA verification applied at GDA Step 3, and their use of UK support at GDA Step 4, which has contributed to the development of their proficiency in applying the R6 procedure.

353. I noted the Hitachi-GE verification and validation proposal for their R6 calculations at Step 4, used expert panels as a means to validate the Hitachi-GE approaches in lieu of the IVB. I questioned the veracity of these arrangements. Hitachi-GE explained that the IVB was engaged at Step 3 to undertake independent comparisons which validated the Hitachi-GE methods and improved their proficiency in applying the R6 procedure. Whereas, at Step 4, members of the IVB provided independent challenge of the assumptions to ensure a balanced QEDS i.e. a conservative result but with achievable demands placed on the future qualification of the NDT system (Reference 102).

354. I observed that Hitachi-GE improved their understanding of the R6 procedure through-out the GDA. This was noticeable in the understanding demonstrated by Hitachi-GE when working out the reasons for the differences between their results and those of

the IVB, and later when challenged by the expert panels. Hitachi-GE have undoubtedly benefited from access to UK R6 expertise.

355. However, post GDA, I am uncertain as to whether Hitachi-GE's verification and validation of its R6 fracture mechanics calculations would meet UK expectations for highest reliability components e.g. the need to use alternative methods to verify the reliability of the R-Code software. I am also unclear whether the Hitachi-GE arrangements include provision to invoke UK expertise as in GDA. I acknowledged that Hitachi-GE have made significant progress in its understanding and application of the R6 defect assessment procedure through the GDA Step 4, but raise AF-ABWR-XW to support licensing:

AF-ABWR-SI-08 Software that is the intellectual property of a third party has been used to perform defect tolerance analyses. Confidence in this software and the methodologies it embodies is critical to the safety case; because of this, the licensee shall ensure that robust verification and validation arrangements, incorporating adequate independent review and methods, are developed and implemented to underpin the fracture assessments for very high integrity and high integrity components.

<u>DTA Plan</u>

356. The DTA plan describes the methodology for undertaking the fracture mechanics assessment (Reference 80). It includes Hitachi-GE's basic approach used in the fracture mechanics assessments of the selected regions of the highest reliability components to establish the limiting defect sizes.

357. Hitachi-GE's DTA plan includes details of the important parameters needed for the fracture 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. Hitachi-GE also affirm their 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 defect size taking account of through-life crack growth. This is consistent with the approach established in previous GDAs.

358. At GDA Step 2 my review of Hitachi-GE's DTA plan did not identify any particular areas of concern, and I was satisfied that the proposed approaches would lead to a conservative fracture assessment.

359. Hitachi-GE subsequently revised their DTA Plan following the issue of RO-ABWR-0001 (Reference 103). The main changes included further explanation of the derivation of SLLDS and QEDS, a revised description of the DSM, deletion of the ρ factor (replaced with the R6 'v' factor; accounts for primary and secondary stress interactions effects on the applied Stress Intensity Factor (SIF)) and updates to reflect the fracture assessment status.

360. In addition, to describing their approach to reconciliation for the GDA and licencing, the Hitachi-GE's DTA plan included generic proposals relating to the application of the R6 procedure. I asked Frazer-Nash to undertake a more detailed review of the Hitachi-GE's DTA Plan and their application of R6. I raised RQ-ABWR-1279 to progress my assessment (Reference 104).

361. All questions in my RQ were satisfactorily addressed by Hitachi-GE, apart from a question regarding fatigue cycle counting in the fatigue crack growth (FCG) calculations. Briefly, Hitachi-GE applied a 'cycle by cycle' counting method whereas RGP e.g. BS 7910 recommends a 'rainflow' cycle counting method, whereby the maximum and minimums across all the different transients are combined together to

maximise the stress range. Hitachi-GE provided a comparison for the RPV shell to petal weld between the rainflow and cycle to cycle methods that showed little difference in the FCG calculated. This study suggested that the 'cycle by cycle' method predicts larger FCG than the 'rainflow' method, which is counterintuitive.

362. I was therefore not convinced that the FCG cycle counting approach adopted by Hitachi-GE was demonstrably conservative. However, as the predicted FCG for all VHI welds is low < 5mm, I recognise that the non-conservatism may be small. In addition, the DSMs claimed by Hitachi-GE were all at least 2, and in many cases significantly higher, which was in general , indicative of reasonable defect tolerance for all VHI components (Reference 84).

363. In practice the operating profile (and hence accumulated FCG) will be determined by decisions made by the future licensee. I raise the following assessment finding for the future licensee to gather further evidence to address this point **AF-ABWR-SI-09:**

AF-ABWR-SI-09 – The fatigue damage calculation methodology, used by Hitachi-GE, is not demonstrably conservative. Because of this, the licensee shall demonstrate that either the Fatigue Crack Growth cycle counting approach is demonstrably conservative, or that sufficient conservatisms exist elsewhere in the Fatigue Crack Growth and Defect Tolerance Analysis calculations to compensate for any under-estimation of Fatigue Crack Growth.

364. The DTAs were undertaken for a sample of VHI welds and nozzle crotch corners which Hitachi-GE considers to be representative of the most onerous locations. As noted above there is some subjectivity in this selection process, but I am content that these are a generally representative and sufficient set of limiting locations for the purposes of providing a demonstration for GDA.

365. However, Hitachi-GE recognise that further work will be required post GDA to extend the scope of their DTA work to a wider range of locations in VHI components (welds and forgings) during the licensing phase in order to confirm that the limiting locations have indeed been considered. I take this forward as an assessment finding **AF-ABWR-SI-10:**

AF-ABWR-SI-10 – Hitachi-GE have provided example locations for defect tolerance analysis within GDA but not yet justified the whole plant sufficiently. Therefore, the Licensee shall undertake Defect Tolerance Analysis on a wider range of weld locations on the Very High Integrity components in order to demonstrate that the limiting locations have been assessed. The Licensee shall also undertake Defect Tolerance Analysis on the vulnerable areas of the parent forgings in order to demonstrate that the limiting locations have been assessed.

366. <u>DTA for Specific VHI Locations</u>

367. This review has been informed by reviews performed by my TSC. The transients used within the analysis have not been the subject of Structural Integrity review, but have been treated within the Fault Studies discipline (Reference 166). My sampling of the specific DTAs was informed by Hitachi-GE's structural integrity classification (Reference 7), and focused on the VHI components. To inform a view on the veracity of Hitachi-GE's DTA capability, my sampling also considered Hitachi-GE's weld ranking procedure and its application (Reference 78 and Reference 79 respectively).

<u>Weld Ranking</u>

368. For closure of RO-ABWR-0001 the expectation is that the RP demonstrates an understanding of UK expectations by developing their DTAs and Technical Justifications for inspection (TJs) and applying them to a bounding set of assessment

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locations. The weld ranking procedure describes Hitachi-GE's methodology for identifying the limiting regions of the highest reliability components for detailed assessment during GDA. The identification of the limiting regions follows a semi-qualitative approach 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.

369. Although the procedure is termed a weld ranking procedure, it includes 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.

370. At GDA Step 2, ONR was content with Hitachi-GE overall approach outlined in their weld ranking procedure (Reference 78). I recognise there is inevitably an element of subjectivity in the ranking process, but am satisfied the process should provide a suitable approach to identifying the limiting regions.

371. However, the application of the weld ranking procedure to the higher integrity components was reported late in Step 2 and so consideration of the limiting regions identified by the procedure and whether they provide sufficient coverage of the higher integrity components was addressed at GDA Step 3. I concluded that the Hitachi-GE approach was adequate to prioritise their assessment work (Reference 105).

372. The Weld Ranking Application Report describes the outputs from the application of the weld ranking procedure to identify the limiting areas on the highest reliability components (Reference 79). At GDA Step 2, Hitachi-GE identified 7 high priority locations in the RPV and 1 location in the MS Piping for detailed evaluation (see Figures 3 and 4):

373. Reactor Pressure Vessel

- Longitudinal weld in top head (2, Figure 3)
- Circumferential weld of flange-top head side (3, Figure 3)
- Shell 4 to bottom head circumferential weld (7, Figure 3)
- Circumferential weld of bottom head (9, Figure 3)
- Main steam outlet nozzle (N3) to shell weld (10, Figure 3)

374. Main Steam System Piping

- Main Steam Seamless pipe to MSIV (W104, W305, W505, W704,Figure 4)

[redacted]

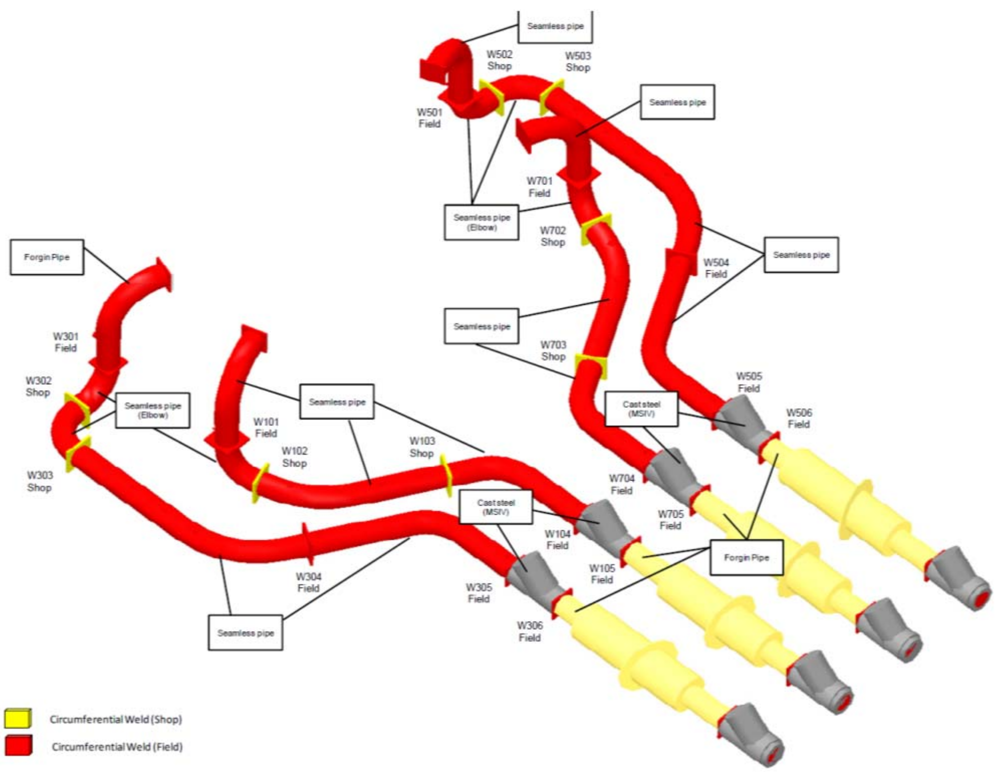

Figure 5: MS Piping Weld Locations Schematic (Reference 79)

375. The above listing includes a variety of VHI weld types; shell to shell; nozzle to shell, weld repairs along with nozzle crotch corner features. I was satisfied that Hitachi-GE had identified a challenging set of locations with representative weld types and geometrical features to progress the GDA. In my view Hitachi-GE's weld ranking process provides an adequate basis for the development of their DTA Plan and for addressing ROA-RO-ABWR-0001.A2.1.

376. Following ONR questioning (Reference 106), Hitachi-GE established that postulated multiple RPV stud failures i.e. > 8 adjacent studs would have unacceptable consequences for nuclear safety (Reference 7). I consider that a vessel closure provision for a highest reliability component that can accommodate in excess of 8 adjacent stud failures is indicative of adequate redundancy and the absence of a 'cliff-edge' effect from load shedding.

377. In view of this redundancy, I also expect an adequate level of defect tolerance for multiple stud failures. Hitachi-GE intend to use NUREG 1.65 (Reference 107) for their defect tolerance demonstration post the GDA. I am content with this approach, but also expect the depth of the defect tolerance demonstration to be informed by the likelihood of initiating a defect i.e. the fatigue usage factor and the ISI/replacement policy. These are licensing matters post the GDA. I raise the following assessment finding to capture my expectation:

AF-ABWR-SI-11 – The Reactor Pressure Vessel closure studs are standard ASME class 1 components when considered individually, but a component of the highest reliability, when considering multiple (common-cause) stud failure. Because Hitachi-GE have not considered common cause failure explicitly within GDA, it is required within licensing. The licensee shall provide a suitable defect tolerance demonstration for multiple stud failures with the depth of the demonstration informed by appropriate good practice, the fatigue usage factor, and the inspection/replacement policy.

378. ONR commissioned its TSC to perform DTA review. The depth of these reviews took cognoscente of the rigour applied in the Hitachi-GE DTA verification process and independent expert scrutiny. Review of the Technical Justifications for Inspection were performed by internal ONR resource and took cognisance of Hitachi-GE's weld rankings. My sampling choice therefore considered both the DTA and NDE scoring. I cover the inspection aspects later, it will suffice to say that low combined scores were candidate assessment locations for ONR's comparative DTA work and review of the inspection TJs. My comparative work and the reasoning behind my selections are detailed below:

- RPV Shell Weld; potential for the reduction in fracture toughness due to irradiation embrittlement ((parent forging & Weld 7 Figure 2, RQ 1508 (Reference 108)
- RPV Bottom Head Welds; low combined scores Welds 8 and 9 Figure 2, (Reference 109)
- RPV Bottom Head Welds (Welds 8 and 9 Figure 2, (Reference 110)
- Main Steam Piping to inboard MSIV weld; low DTA and combined scores Weld W305, (Reference 111)
- MSIV Body repair; challenge to demonstrate defect tolerance under the combined effects of cast material, random defect orientations and residual stress. (Reference 112),

379. I discuss my regulatory queries along with Hitachi-GE responses for each VHI location next.

RPV Shell Weld

380. The focus of my initial assessment of the RPV Shell weld (parent forging & Weld 7 Figure 2) was to gain assurance in the veracity of Hitachi-GE's DTA verification and arrangements. I raised RQ-ABWR- 0357 to progress my assessment. In addition, my regulatory queries covered the identification of the limiting DTA location in the RPV lower shell and support region, DTA for non-weld regions and the choice of defect aspect ratio (Reference 113).

381. In their response Hitachi-GE explained that the limiting DTA location/defect orientation took account of the major working direction during forging; the stress distributions at the support skirt juncture with the RPV shell weld; and a trial ELLDS calculations. I was satisfied with Hitachi-GE's reasoning and noted that the initial DTA for the RPV shell weld and support skirt juncture indicated ELLDS values of 50 mm and 60 mm respectively. I was satisfied that RPV shell weld was the limiting location for the progression of the GDA.

382. I was aware that areas of stress concentration and cycling could initiate cracking in non-weld locations and asked Hitachi-GE to explain whether non-welded regions would feature in the GDA. Hitachi-GE identified nozzle crotch corners had caused problems historically and so committed to including nozzle crotch corners in their DTAs for the GDA. Hitachi-GE also clarified that the choice of defect aspect ratio for inspection of the plant during manufacture and build; a 6:1 surface breaking defects were assumed in the DTA with sensitivity studies covering 2:1 and 10:1 defect depth to length ratios accorded with their DTA Plan (Reference 114). Overall, Hitachi-GE provided an adequate response to my regulatory queries raised in RQ-ABWR- 0357.

383. The initial DTA for the RPV shell at GDA Step 3 indicated limiting ELLDS and SLLDS values for a 6:1 inner surface radial-axial defect of 80 and 39mm respectively under a hydrotest condition prior to service (transient C28). Fatigue crack growth (FCG) was low < 5mm showing that the defect size margin would not be significantly eroded through-life and hence that the initial QEDS value of circa 39 mm was dominated by the limiting ELLDS value of circa 80 mm under the hydrotest (Reference 115).

384. Post GDA Step 3, Hitachi-GE revised their DTAs for the RPV, MS Piping and MSIV. The revision of the DTAs took account of several developments; Hitachi-GE's comparative work, internal expert panel discussions and RQs raised by ONR. For the RPV shell weld DTA, the main changes included the inclusion of membrane and bending stresses in the estimation of plastic collapse, reduction in upper shelf fracture toughness, revision of the seismic loadings. In my opinion the most important of the above change related to the plastic collapse solution and the use of a reduced fracture toughness of $190MPam^{1/2}$. The ELLDS and SLLDS values for a 6:1 inner radial-axial defect under the limiting LOCA (transient C22) were subsequently reduced to 48.8 mm and 22 mm respectively (Reference 116).

385. I examined the change in the limiting transient for the ELLDS. Hitachi-GE explained that the limiting hydrotest condition (transient C28) is completed in the construction phase and so did not contribute to in-service FCG. I am also aware that the qualified manufacturing inspection will be supplemented with qualified pre-service inspections prior to service, which will screen out any structural significant defects with high reliability.

386. I was satisfied with Hitachi-GE's explanation for the change in the limiting ELLDS transient for the RPV shell weld. However, reconciliation would require confidence for the GDA in the reliable detection of the lower SLLDS defect i.e. a QEDS value of up to 22 mm. I raised RQ-UK-ABWR-1516 to progress resolution of this point and I discuss further under my consideration of the manufacturing inspection proposals, below.

387. With the support of my TSC I also undertook a review of Hitachi-GEs updated DTA for the RPV shell weld. My main queries related to the FCG laws and residual stress values. Hitachi-GE provided satisfactory responses to close-out the RQ.

388. In addition, my TSC undertook sample comparative ELLDS calculations for postulated internal axial defects which were within 1% of Hitachi-GE. For internal circumferential defect the comparative calculations were also within 4%. (Table 5). The results of the ONR commissioned comparative work for the RPV Shell weld therefore closely matched the results from Hitachi-GE.

389. The comparative calculations completed for the RPV shell weld was part of a wider and more detailed review of the DTAs, which included the RPV bottom head welds, MS Piping and the MSIV. I discuss these in turn next.

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Table 5. Summary of Comparative DTA and FCG Calculations [redacted]

* These defects reach the calculation limit of the SIF solution (85% of the wall thickness of 25.95mm) before becoming limiting and so both calculations give the same result by default.

† Assessment thickness of 47.32mm assumed.

RPV Bottom Head Weld

390. I reviewed two revisions of the DTA for the RPV bottom Head welds and issued RQ-ABWR-1282 and RQ-ABWR-1421 respectively (Reference 117 and Reference 118). My key questions concerned residual stress values, sensitivity to bending stress categorisation, modelling of the cladding, accounting for crack face pressure and the FCG calculations.

391. Hitachi- GE provided satisfactory responses to the majority of my questions. I progressed my understanding and confidence in Hitachi-GE's capability in using the R6 procedure via comparative DTA work.

392. Using the input data from the Hitachi-GE DTA, I commissioned comparative ELLDS and FCG calculations (both in-air and in-water environments) for a sample of defects. The comparative R6 calculations successfully replicated a sample of the ELLDS calculations. I was therefore broadly content with this aspect.

393. However, replication of the FCG calculations for external defects (air environment) was initially only partly successful. For example, the SLLDS for an outside circumferential defect at the shell to petal weld is quoted by Hitachi-GE to grow from 40.3mm (a) x 120.9mm (c) to 42.7mm x 122.1mm. In contrast my independent calculations indicated growth at the deepest (a) and surface (c) of 80% of those presented by Hitachi-GE. I investigated the reasons for the large discrepancy with Hitachi-GE. In addition, Frazer-Nash improved their method for extracting the stress ranges from the Hitachi-GE reports. Following refinement the FCGs were very close to those of Hitachi-GE (Table 5). I consider the above comparative work provided an adequate check of the Hitachi-GE air environment FCG calculations.

394. Similarly, replication of the FCG calculations for internal defects (water environment) was initially only partially successful. For example, the SLLDS for an inside axial defect at the petal to dome weld was quoted by Hitachi-GE to grow from 70.7mm (a) x 212.1mm (c) to 75.2mm x 218.0mm. Whereas the ONR commissioned independent calculations gave a grown defect size of 77.3mm x 218.5mm from the same initial starting defect description i.e. growth at the deepest (a) point was 48% larger than presented by Hitachi-GE. I investigated the reasons for the large discrepancy with Hitachi-GE. Frazer-Nash also revised the comparative FCG calculations to include more accurate stress ranges and obtained FCGs very close to those of Hitachi-GE. I view this as providing an adequate check of the Hitachi-GE water environment FCG calculations. Overall, Hitachi-GE provided adequate responses to close RQ-ABWR-1421.

Main Steam Piping

395. My assessment of the DTA for the main steam piping included reviews of the Hitachi-GE's DTA and the IVB comparison reports along with comparative ELLDS calculations.

396. In reviewing Hitachi-GE's DTA for the Main Steam Piping (MSP) I raised several questions through RQ-ABWR-1280. The key points related to the use of the up to date revision of R6 and BS 7910, residual stress values, the effectiveness of PWHT, FCG laws, stress analyses and plastic collapse sensitivity studies. Hitachi-GE responded and provided adequate responses to close RQ-ABWR-1280.

397. I also undertook a high level review of the independent comparison work completed by the IVB, (Reference 119). I made the following observations:

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- Hitachi-GE's method of stress calculation (ASME design equations) over-estimated the stresses compared to the simple 2D axisymmetric Finite Element Analysis (FEA) used by the IVB.
- FCG was small and similar results were obtained by Hitachi-GE and the IVB.
- Initially, significant differences were found between the SLLDS and ELLDS values[1]. These were attributed mainly due to over-conservatism by the IVB in terms of choice of plastic collapse solutions and welding residual stresses in the ELLDS calculations. In my opinion, Hitachi-GE and the IVB provided sound explanations to reconcile the differences in their results.
- Once these differences were reconciled, Hitachi-GE and the IVB obtained similar DTA results when using the same inputs and assumptions.

398. As part of my review, I commissioned comparative DTA calculations for the MSP. Using the same inputs as specified by Hitachi-GE and R-Code Version 5.1 (as used by Hitachi-GE) the ELLDS results obtained are presented above in Table 5.

399. The results show very good agreement. With regard to FCG using the same inputs and methods, Hitachi-GE and the IVB obtained almost identical results (Figures 12 and 13 of Reference 119). As the FCG was small < 3mm, I did not commission additional comparative FCG calculations.

MSIV Body

400. The scope of my assessment of the DTA for the MSIV body included reviews of the Hitachi-GE's DTA and the IVB comparison reports along with comparative ELLDS calculations.

401. In my review of Hitachi-GE's DTA report I raised several questions the most significant covered the corrosion allowance, the FE model boundary conditions, the location of stress classification lines, the bolt-up cycles in the FCG calculations and the choice of plastic collapse solution. I progressed my assessment through RQ-ABWR-1281 and received satisfactory responses from Hitachi-GE (Reference 120).

402. My view is that there are considerable conservatisms in the DTA calculations for the MSIV, for example treating the membrane stress from bolting-up as primary, treating through-wall bending stresses as primary and the use of local limit load (collapse) solutions. Nonetheless two points are worth noting:

- As with all the DTAs the results are contingent on the fracture toughness value being achieved in practice for the MSIV casting and repair welds a value of 215MPa√m needs to be underpinned by fully representative testing.
- The proposed surface and UT NDE inspections will be qualified for 100% of the MSIV body, because repair welds could be required anywhere. However, the NDE will only be deployed in the areas of actual repair welds. It was not clear whether the NDE inspections should also be deployed to look for defects in the parent castings.

403. I consider these points further under the reconciliation section and the manufacturing inspections section respectively.

404. An independent comparison to the Hitachi-GE DTA of the MSIV was performed by AMECFW as the IVB and is reported in Reference 121. This included a completely independent thermal and mechanical FEA. I undertook a high level review and make the following observations:

- There were significant differences in stress categorisations, stress paths, flaw geometries, bounding transient selection and fatigue cycles used between Hitachi-GE and the IVB. In general, I consider the assumptions made by Hitachi-GE are more appropriate than those made by the IVB, who were in some cases overly conservative. Despite this, because of conservatisms in their calculations, Hitachi-GE consistently calculated smaller (or similar) SLLDSs and ELLDSs than the IVB.

- The smallest SLLDS and ELLDS calculated by Hitachi-GE and the IVB are similar. Given the significant differences in the calculations, this was most likely a coincidence, but provides further confidence in the independent verification process.

405. Overall, in my opinion, Hitachi-GE and IVB provided sound explanations to reconcile the differences in the DTAs.

406. I also commissioned Frazer-Nash to undertake comparative ELLDS calculations for the MSIV. The comparative ELLDS results obtained are presented in Table 5, above.

Summary: RO-ABWR-0001 DTA

407. For closure of RO-A BWR-0001 my expectation is that the RP demonstrates an understanding of UK expectations by developing their DTAs and TJs and applying them to a bounding set of assessment locations.

408. My assessment approach for DTA considered Hitachi-GE's weld ranking procedure and its application to identify a set of limiting weld locations for the GDA. In addition, my view of Hitachi-GE's proficiency in using the elastic-plastic R6 procedure was informed through reviews of their generic methodologies and the application of R6 to specific VHI welds. I also considered the effectiveness of Hitachi-GE's verification arrangements and their proposals for reconciliation.

409. Hitachi-GE developed a weld ranking procedure to identify the limiting regions of the highest reliability components. This took account of the expected limiting defect sizes and the difficulty in detecting such defects to establish those areas which are likely to be limiting in an avoidance of fracture demonstration. The process also included areas of the parent materials subject to high stresses or inspection difficulties (above). I am satisfied the process and its application provided a suitable basis for identifying the limiting regions and to prioritise Hitachi-GE's assessment work for the purposes of the GDA. I therefore consider Action ROA-R0-ABWR-000.A2.1 (Fracture Assessment) closed.

410. Hitachi-GE's proposals for DTA verification during GDA were in principle logical, robust and included significant independent challenge. I gained further confidence from sampling the application of the verification process. I commend Hitachi-GE for the rigour of their DTA verification applied at GDA Step 3, and their use of UK support at GDA Step 4, which has contributed to the development of their proficiency in applying the R6 procedure.

411. Hitachi-GE provided adequate responses to my regulatory queries relating to generic methodologies in their DTA plan. All questions in my RQ were satisfactorily addressed by Hitachi-GE, apart from a question regarding fatigue cycle counting in the fatigue crack growth (FCG) calculations. I was not convinced that the FCG cycle counting approach adopted by Hitachi-GE was demonstrably conservative. Resolution of this question is dependent on the operating profile, which is a matter for the future licensee. I anticipate this point can be readily addressed during licensing on account of the low predicted FCG and the expected DSM margins from the DTAs, but have raised an assessment finding for the future licensee to address.

412. I am satisfied that Hitachi-GE have provided adequate evidence to establish the limiting defect sizes for limiting locations using the R6 fracture mechanics methods

with bounding transients and estimates of fatigue crack growth through life. My conclusion is based on the following:

- In general, there is good agreement between the Hitachi-GE DTA results and those of the Independent Verification Body (IVB) AMEC Foster Wheeler (AMECFW), once assumptions and conservatisms had been rationalised.
- Comparative End of Life Limiting Defect Size (ELLDS) calculations by Frazer-Nash (using the Hitachi-GE inputs) for the MSP, MSIV, RPV bottom head weld and RPV shell weld which closely match the Hitachi-GE results.
- Comparative FCG calculations by Frazer-Nash (using the Hitachi-GE inputs) for the RPV bottom head weld in both air and water environments which closely match the Hitachi-GE results.

413. Overall I am satisfied that Action ROA-R0-ABWR-000.A2.1 (Fracture Assessment) can be closed and that the Hitachi-GE DTA results provide a suitable basis for reconciliation with NDT for the purposes of the GDA.

414. There are some specific points where further work is needed by the future licensee. To ensure that these points are satisfactorily resolved I have raised assessment findings UK-ABWR-SI-07, UK-ABWR-SI-08, UK-ABWR-SI-09, UK-ABWR-SI-10 and UK-ABWR-SI-11.

RO-ABWR-0001: Manufacturing Inspections

415. The expectation for GDA is for Hitachi-GE to demonstrate that high reliability non-destructive examination (NDE[‡]) can be performed during the manufacture of VHI and HI components and that this reliability can be demonstrated through a formal process of inspection qualification. In practice, this entails applying NDE methods that are based upon sound physical principles and that are widely used in industry.

416. The approach to the assessment is twofold:

- Establish that the design of VHI and HI components enables high reliability NDE to be applied.
- Review the effectiveness of Hitachi-GE's proposed NDE techniques based upon the evidence presented in their 'GDA technical justifications'.

417. Hitachi-GE have stated (Reference 122) that the manufacturing inspections for the VHI and HI components will be qualified in accordance with the ENIQ methodology (Reference 123).

418. The main requirements of the ENIQ methodology are:-

- The qualification of the NDE procedure and the NDE personnel are separated.
- NDE procedures are qualified through a combination of written technical justification and practical demonstration. The technical justification brings together the physical basis of the NDE techniques with experimental and theoretical evidence to support the ability of the NDE to meet the pre-defined inspection objectives.
- The qualification body (a body of experts acting independently of the inspection organisation and licensee) assess the technical justification and the results from

[‡] *In some of the other GDA documents and references, the synonymous term 'Non-Destructive Testing (NDT) may be used.*

the practical trials. If the qualification body judges that the outcome is satisfactory, it will issue a qualification certificate or equivalent statement.

- NDE personnel are qualified to apply a specific inspection procedure; personnel qualification includes blind trials to demonstrate competence in applying the procedure.

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

420. In my opinion, Hitachi-GE's strategy for qualifying the end of manufacture NDE of VHI components is a well-developed document that describes the principle activities for inspection qualification along with the roles and responsibilities of the relevant organisations.

4.4.1.4 HITACHI-GE APPROACH FOR QUALIFIED INSPECTIONS DURING GDA

421. Reference 125 presents Hitachi-GE's approach to providing evidence that manufacturing NDE is able to deliver the required capability. The main element of this approach was to present the evidence for the NDE capability in the form of 'GDA technical justifications'. These documents were shortened versions of the technical justifications that would eventually be needed for a full qualification and included a summary of: the inspection objectives; an overview of the proposed inspection techniques and available evidence to support the capability of the NDE.

422. Hitachi-GE commissioned an independent assessment of each of the GDA technical justifications from an '*ENIQ-based GDA qualification body*', the purpose of this independent assessment being to assess whether (Reference 125):

- *the evidence that is presented is correct and relevant to the proposed NDE techniques.*
- *the evidence supports the conclusion that the proposed NDE system will provide a reliable inspection at end of manufacture.*
- *the reliability of the NDE system can be demonstrated through qualification.*

423. Hitachi-GE appointed AMEC Foster Wheeler's Inspection Validation Centre (IVC) to undertake the role of the GDA qualification body. This assessment process comprised:

- A review of the GDA technical justification with IVC comments recorded on a comments checksheet.

- Hitachi-GE's response that included a description of how they planned to address each comment. IVC then stated whether the proposal for addressing the comments was acceptable. In several cases there was a mutual acknowledgement that the issue could only be resolved during the full qualification.

- IVC verified that the comments had been addressed as agreed (or otherwise).

424. An important input to the inspection plans is the QEDS derived from a R6 fracture mechanics calculation (above). The initial QEDS was calculated prior to the production of each GDA technical justification but these values were subsequently revised at a stage when the technical justifications had already been produced. In most cases Hitachi-GE confirmed that the initial value of the QEDS was acceptable, being either correct or conservative. However, in one case, the RPV shell welds, the through-wall dimension of the QEDS was revised down from 30mm to 20mm. Hitachi-GE responded to this change by producing a new technical justification.

425. The GDA technical justifications aimed to demonstrate with a reasonable level of confidence that the end of manufacture NDE was capable of:

- detecting with high reliability all planar defects at or greater than the QEDS.
- characterising defects with high reliability as either volumetric or planar.
- measuring the size and position of defects.

426. The potential defects of concern that could arise during manufacture were derived from an expert panel review; this also included an expert judgement of the main features of these defects that could influence the inspection performance. I have seen outline details of the expert panel review and I am content with the process applied.

4.4.1.5 TECHNICAL JUSTIFICATIONS: GENERIC FEATURES

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

428. The technical justifications propose the deployment of ultrasonic probes using a motorised scanner combined with digital data collection and display. This approach provides good control of the probe scanning, the ability to analyse the data off-line and provides a permanent record of the inspection that can be viewed at a later stage.

429. Hitachi-GE produced the GDA technical justification in the format of an ENIQ style technical justification described in Reference 122. These technical justifications presented the inspection objectives and the proposed NDE methods along with evidence to support the claim that the NDE techniques are likely to be capable of meeting the inspection objectives. While it is not a GDA requirement to use the ENIQ format for such a purpose, I welcome this approach since it demonstrates Hitachi-GE's understanding of the ENIQ methodology for inspection qualification and was helpful in seeing how the information could be extended to form part of a full qualification. As such the documents that have been produced during GDA are expected to be a good basis for producing full technical justifications in the future.

430. A critical part of the inspection in most cases is defect characterisation which assesses whether a defect is planar or volumetric. This is an important stage of the inspection since the accept/reject criteria defined by ASME III NB 5331 specify that all defects assessed as planar by NDE are deemed unacceptable, thereby leading to repair or rejection of the item. With regard to the avoidance of fracture claim for VHI components, it is essential that the NDE has high reliability for correctly characterising all planar defects that are at or larger than the QEDS. The relative importance of the role of characterisation means that the measurement of defect size has a significantly minor role in determining whether defects are acceptable or rejectable. Hitachi-GE have selected the methods defined in EN ISO 23279 for defect characterisation which evaluates the nature of a defect from a comparison of the signal amplitudes from different ultrasonic beams along with echodynamic information (the variation of the ultrasonic response observed as the probe is moved over the defect).

431. The physical basis of this method for characterisation seems sensible and is based on sound physical principles. The flow chart that is applied to establish the nature of the defect defaults to a planar defect at each stage and in this respect the method appears to be conservative. On the one hand, the apparent conservative nature of this evaluation method appears beneficial, but I note that weld repairs are often detrimental for structural integrity. Consequently, an overly conservative approach in defect

characterisation that may result in unnecessary repairs is undesirable. I note that, while the application of industrial standards may provide good technical solutions, the high reliability that is demanded for an inspection of a VHI component remains to be established; furthermore, the reliability of the approach defined in EN ISO 23279 will vary with the inspection situation and will need to be established during the full qualification exercise.

432. In some cases, the GDA technical justifications specified an accept/reject criteria based upon the defect size leaving the possibility that the component could be deemed acceptable yet still containing planar defects. I explained to Hitachi-GE that this did not meet my expectation of confirming that components were as defect free as possible. I also pointed out that this approach was also inconsistent with the acceptance criteria defined in ASME III NB 5331. I used the technical justification for the RPV petal to bottom dome weld as an example and issued an RQ (Reference 138) that included a request for Hitachi-GE to clarify its intentions with respect to accept/reject criteria. Hitachi-GE confirmed that all planar defects, regardless of dimensions, are rejectable and while the response to the RQ was specific to the petal to bottom dome weld it was also applicable to all cases. To ensure that this taken forward I have raised the following assessment finding:

AF-ABWR-SI-12 It is ONR's expectation that planar defects in major components should be rejected. Because Hitachi-GE have not provided sufficient confidence that this will be done, the licensee shall set appropriate inspection objectives for the end of manufacture Non-Destructive Testing to specify that all defects characterised as planar will be rejected.

433. The evidence for the likely inspection performance was derived largely from theoretical analysis and by extrapolating the results from previous trials and site inspections. In general I judged this to have been sufficient and satisfactory and enabled me to make the appropriate judgements regarding the end of manufacturing NDE of VHI components.

434. I noted that there was a gradual improvement in the quality of the technical justification as Hitachi-GE developed its understanding of manufacturing NDE qualification and in particular, of the ENIQ methodology. Initially, I felt it necessary to provide detailed comments to help Hitachi-GE understand my expectations for GDA. At later stages, I was able to make appropriate judgements without having to assess the fine detail of the documents. The UK context in inspection qualification is complex. Hitachi-GE have met UK expectations, but with significant input from ONR. The licensee should, therefore, utilise specialist knowledge of the UK context to support licensing as part of normal business.

4.4.1.6 ASSESSMENT APPROACH

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

- inspection objectives could be met using well established NDE techniques that were based upon sound physical principles.
- the component was, as far as reasonably practicable, designed to facilitate the end-of manufacture NDE.
- there was a reasonable prospect that an inspection could be developed that would be successful in a full qualification exercise. In practice, this means that the ultrasonic techniques provide several means for providing good margins for detection and characterisation.

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

437. My approach to the assessment reported here has been to combine a thorough assessment of a sample of inspection plans supplemented by a targeted review of others.

438. This approach has been informed by Hitachi-GE's arrangements for the production of technical justifications which has included the independent review by the IVC. In each case, I have used the IVC review as a means to developing confidence in the effectiveness of the techniques. To this end, I have looked at how comments raised by the IVC have been addressed by Hitachi-GE.

439. In each case, I have considered the design of the component to establish whether, as far as is it is reasonable, it facilitates inspection. In practice, for ultrasonic inspection, this entails ensuring that there are no severe scanning restrictions that might limit the overall reliability. There must also be no physical constraints that could limit the ability to provide a suitable surface for ultrasonic inspection. In addition to the geometrical configuration, it is important that components are suitably transparent to ultrasound.

440. I recognise that referring to techniques described in NDE codes and standards provides evidence of their maturity but this does not, by itself, provide the level of assurance required for the inspection of a VHI component. Ultimately, the effectiveness of the techniques will need to be confirmed through the rigours of a full qualification exercise as part of normal business during the licensing phase.

4.4.1.7 END OF MANUFACTURE INSPECTION OF ABWR RPV SHELL WELD (SHELL 3 TO SHELL 4)

441. In Reference 126, Hitachi-GE included circumferential RPV shell welds in the sample for demonstrating the effectiveness of the end of manufacturing inspection within GDA. Although the RPV shell 4 to bottom dome weld is more limiting from an NDE perspective due to access restrictions, Hitachi-GE considered it prudent to include the shell 3 to shell 4 weld as it was located in the beltline region of the RPV. I am content that this is appropriate.

442. The main features of the weld that have an influence on the inspection performance are:

The weld fusion faces are approximately perpendicular to the surface of the RPV. Consequently, lack of sidewall fusion defects will also be aligned to the inner and outer surface of the vessel.
The wall of the RPV is relatively thick (approximately 180mm).
The inner surface of the RPV is clad with austenitic stainless steel.
There is generally good access to deploy ultrasonic probes from both sides of the weld.

443. Cracks can have tilts and skew of up to 5° with respect to the principal plane. LOSF can have tilts and skews of up to 5° and 2° respectively.

444. Hitachi-GE produced an initial version of the technical justification (Reference 129) that described the proposed NDE techniques for the high reliability detection and rejection of planar defects greater than the QEDS. This version was based upon a QEDS of 30mm x 180mm (for circumferential defects) and was judged to be conservative against a SLLDS of 39mm. Subsequently, the DTA was recalculated to consider additional factors which produced a SLLDS of 22mm. In response the QEDS was revised to 20mm x 120mm (for circumferential defects); the length of transverse/axial

defects was restricted to 110mm, the width of the inspection volume. The technical justification was subsequently, revised to provide confidence in the capability for the detection and rejection of the smaller defects.

445. I am content with the selection of manufacturing defects of concern proposed by Hitachi-GE in their safety submissions. These include:

- lack of side wall fusion (LOSF) which are located at the weld fusion faces and are principally aligned with the fusion face. LOSF are defined as being smooth on the fusion face side of the defect and can be rough or smooth on the weld side.
- circumferential cracks which can occur in the weld or the heat affected zone, can be rough or smooth and are principally aligned parallel to the weld centreline (the welding direction).
- transverse cracks which can occur in the weld or the heat affected zone, can be rough or smooth and are principally aligned perpendicular to the welding direction. The major axis of transverse cracks is parallel to the inner/outer surface.

446. The revised technical justification (Reference 130) included significant changes to the ultrasonic techniques when compared with the previous version. I also noted an improvement in the general layout and technical content of the revised version, a result of the evolution of these documents that has occurred through the GDA (the initial version was produced early in GDA).

447. The proposed ultrasonic inspection consists of applying a combination of direct pulse echo, tandem and corner trap techniques for detecting the defects of concern. The tandem inspection is designed specifically to detect embedded defects that are aligned close to the radial direction. For detection the probes are scanned using automated techniques from only the external surface; the option of scanning from the internal surface is considered for defect evaluation (characterisation, sizing and location). I welcome the application of an automated inspection over manual deployment as it offers more control over the scanning pattern and provides a permanent record of the data that can be interpreted off-line.

448. In undertaking an independent review, IVC reviewed revision DR1 of the technical justification and then agreed with Hitachi-GE's responses to the IVC comments. It appears that IVC comments have been addressed in moving to revision 1 of the technical justification but there is no explicit statement by IVC that they have checked all of the comments have been adequately addressed.

449. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on two considerations: first the design of the weld enables relative good access for ultrasonic inspection; secondly the inspection techniques are based upon sound physical principles and use well understood and established techniques.

4.4.1.8 END OF MANUFACTURE INSPECTION OF ABWR RPV SHELL 4 TO PETAL WELD

450. The RPV shell 4 to bottom dome weld was included in the sample of welds for GDA as the change of section in the petal region restricts the access for scanning ultrasonic probes from the lower side of the weld. In this respect this weld represents a more challenging situation when compared with the shell weld in the vicinity of the belt line region.

451. The technical justification (Reference 136) states the QEDS as having a through-wall extent of 30mm which is significantly smaller than the SLLDS of 40.3mm.

452. In undertaking an independent review, IVC reviewed revision DR1 of the technical justification and then reviewed Hitachi-GE's responses to the IVC comments. All but two of the responses were judged by IVC to be acceptable. While there is no explicit statement by IVC that they have checked all of the comments have been adequately addressed, it appears that the IVC comments have been addressed in moving to revision 0 of the technical justification (there is a statement to this effect in the technical justification).

453. The techniques described in the technical justification are essentially the same as those applied for the RPV shell 3 to shell 4 manufacturing inspection (see section 4.4.1.7). The only difference between the two inspections is that the change of section in the vicinity of the petal restricts the scanning from petal side of the weld.

454. While the technical justification does not include scans from the inner surface, such scans can be performed and would provide a valuable means of enhancing the overall inspection capability and compensate for the change of section. Indeed, I note that the technical justification specifies that scans from the inner surface will be performed to aid sizing if deemed necessary thereby confirming that such scans are possible.

455. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects at or larger than the QEDS). My judgement is essentially based on two considerations: first the design of the weld enables relatively good access for ultrasonic inspection; secondly the inspection techniques are based upon sound physical principles and use well understood and established techniques.

4.4.1.9 END OF MANUFACTURE INSPECTION OF ABWR RPV TOP HEAD TO FLANGE WELD

456. The RPV top head to flange weld joins the top head petal (fabricated from SA-533M Type B Class 1 plate) to the forged flange (fabricated from SA-508M Grade 3 Class 1). The end of manufacturing inspection is required to detect and reject planar defects that are larger than the QEDS of 9.5mm x 57mm. It appears strange that the through-wall size is given with such precision (and not given as 10mm) as the smallest value for the SLLDS was calculated as 10.4mm.

457. The main features of the weld that have an influence on the inspection performance are:

- The weld fusion faces are approximately perpendicular to the surface of the RPV top head/flange.
- The weld is relatively thick (nominal thickness 99mm) and there is a gradual taper on the flange side of the weld.
- The component has a mainly spherical geometry (although the outer is not strictly spherical due to the taper on the external surface).
- For specific manufacturing defects given in the technical justification, defects are principally aligned with the radial direction (within 5° of the radial direction). Defects that have their principle directions in the welding direction and transverse to the welding direction are considered. Defects are considered to be smooth or rough.

458. The GDA technical justification for the ABWR RPV top head to flange weld specifies the following techniques for the detection and evaluation of defects of concern:

- Tandem techniques for detecting embedded defects.
- Mode-conversion tandem techniques for detecting embedded defects.

- Pulse echo techniques for detecting near surface defects through either specular reflection or through the corner trap mechanism.

459. These techniques are applied from both sides of the weld on the external surface but some scans from the flange side may be restricted due to the radiussed surface at the base of the flange. Optional scans are also defined from the inner surface to confirm and to evaluate defects located at the mid-wall and inner region of the weld.

460. IVC reviewed the initial revision of the technical justification. Hitachi-GE proposed a resolution for each comment and revised the technical justification accordingly. IVC reviewed the updated version (revision 0) to assess whether the comments had been adequately addressed. IVC concluded that while some of the comments had not been fully addressed, it was sufficient for the purposes of GDA that these could be addressed by a future revision of the technical justification at the full qualification stage.

461. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on two considerations: first the design of the weld enables relative good access for ultrasonic inspection; secondly the inspection techniques are based upon sound physical principles and use well understood and established techniques.

4.4.1.10 END OF MANUFACTURE INSPECTION OF ABWR RPV TOP HEAD LONGITUDINAL WELD

462. The RPV top head longitudinal weld joins the top head petals together, which are, in turn, attached to the top head dome; all components fabricated from SA-533M Type B Class 1 plate. The end of manufacturing inspection is required to detect and reject planar defects that are at or larger than the QEDS of 9.5mm x 57mm. It appears strange that the through-wall size is given with such precision (and not given as 10mm) as the smallest value for the SLLDS was calculated as 10.4mm. This suggests an unrealistic level of precision is being considered by Hitachi-GE.

463. The main features of the weld that have an influence on the inspection performance are:

- The weld fusion faces are approximately perpendicular to the surface of the RPV top head petal/dome.
- The weld is relatively thick (nominal thickness 99mm).
- The dome and petal have a spherical geometry.
- For specific manufacturing defects given in the technical justification, defects are principally aligned with the radial direction (within 5° of the radial direction). Defects that have their principle directions in the welding direction and transverse to the welding direction are considered. Defects are considered to be smooth or rough.

464. The GDA technical justification for the ABWR top head longitudinal weld specifies the following techniques for the detection and evaluation of defects of concern:

Tandem techniques for detecting embedded defects.
Mode-conversion tandem techniques for detecting embedded defects.
Pulse echo techniques for detecting near surface defects through either specular reflection or through the corner trap mechanism.

465. These techniques are applied from both sides of the weld from the external surface. Optional scans are also defined from the inner surface to confirm and to evaluate defects located at the mid-wall and inner region of the weld.

466. IVC reviewed the initial revision of the technical justification. Hitachi-GE proposed a resolution for each comment and revised the technical justification accordingly. IVC reviewed the updated version (revision 0) to assess whether the comments had been adequately addressed. IVC concluded that while some of the comments had not been fully addressed, it was sufficient for the purposes of GDA that these could be addressed by a future revision of the technical justification at the full qualification stage.

467. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on two considerations: first the design of the weld enables relative good access for ultrasonic inspection; secondly the inspection techniques are based upon sound physical principles and use well understood and established techniques.

4.4.1.11 END OF MANUFACTURE INSPECTION OF ABWR RPV BOTTOM DOME TO PETAL WELD

468. The bottom dome/bottom petal of the RPV has a torispherical geometry with both pieces forged from ASME II SA-508M Grade 3 Class 1 low alloy steel. The dome and the petal are joined by a single circumferential weld that is located between the penetrations for the reactor internal pumps (RIPs) and the penetrations for the control rod drive mechanisms and the in-core monitoring instrumentation. The centre line of the weld forms a cylinder with the RPV axis at the centre; this means that the weld centreline is at an angle of approximately 30° to the inner/outer surface of the dome/petal. The weld fusion faces are approximately parallel to the weld centreline. The inner surface of the dome/petal is clad with nickel-base alloy.

469. The position of the control rod drive (CRD) penetrations and in-core monitoring penetrations severely restricts the scanning of ultrasonic probes on the dome side of the weld. Additionally, the location of the ten RIP penetrations severely restricts scanning of ultrasonic probes from the petal side of the weld at the locations of the RIPs.

470. I have reviewed, and am content with, the defects of concern proposed by Hitachi-GE. These include:

- lack of side wall fusion (LOSF) which are located at the weld fusion faces and are principally aligned with the fusion face. LOSF are defined as being smooth on the fusion face side of the defect and can be rough or smooth on the weld side.
- circumferential cracks which can occur in the weld or the heat affected zone, can be rough or smooth and are principally aligned parallel to the weld centreline (the welding direction);
- transverse cracks which can occur in the weld or the heat affected zone, can be rough or smooth and are principally aligned perpendicular to the welding direction. The major axis of transverse cracks is parallel to the inner/outer surface.

471. Cracks can have tilts and skew of up to 5° with respect to the principal plane. LOSF can have tilts and skews of up to 5° and 2° respectively.

472. The QEDS determined from the DTA was 45mm x 270mm for circumferential defects and 45mm x 76mm for transverse cracks. The smaller length for transverse cracks was to restrict manufacturing defects to the weld and heat affected zone region.

473. The technical justification (Reference 137) described a process for evaluating defects that meant that a defect characterised as being planar would only be rejected/repaired if it was measured as being larger than the QEDS. By applying this process, the presence of a relatively large planar defect, albeit smaller than the QEDS, would be deemed acceptable. This potential outcome did not meet with my expectations based upon the ONR SAPS that components should be as defect free as possible. Nor did this potential outcome meet the accept/reject criteria given in ASME III NB 5331 which states that all planar defects are unacceptable. Furthermore, this would place an increasing burden on the sizing capability. I raised this point under regulatory query RQ-ABWR-1474 (Reference 138) and in response Hitachi-GE confirmed that the accept/reject criteria should be amended to ensure that any defect characterised as planar would be deemed rejectable (Reference 139). I am satisfied that the response to the regulatory query was sufficient on this matter and that a revision to the technical justification was not necessary. The response on this matter also resolved some additional comments in RQ-ABWR-1474 relating to defect sizing.

474. The inspection techniques defined in the technical justification for transverse defects comprise a combination of tandem and conventional pulse echo techniques deployed from the external surface of the weld. In general these scans can be applied unhindered but the technical justification notes that there are some positions where the presence of the CRD penetrations limits the effectiveness of the tandem inspection near to the external surface. However, this limitation is compensated for by the use of high angle pulse echo inspections.

475. The proposed inspection for circumferential defects deploys ultrasonic probes scanned on the external surface from the petal side of the weld; the presence of the CRD and ICM penetrations severely restricts the deployment of probes from the dome side. The circumference of the weld can be separated into two regions with regard to the inspection techniques and capability, the regions of the weld where the position of the RIP penetrations restricts coverage and those regions between the RIPs where there is no such restriction.

476. For the regions between the RIP locations, scans from the petal side using 45° and 60° probes are used for detecting circumferential defects. The orientation of the weld with respect to the scanning surface (approximately 30°) means that a 60° probe gives close to normal incidence on circumferential planar defects and will therefore be expected to give a high amplitude signal. Away from these regions, the scanning restrictions imposed by the RIPs limit the coverage of the 60° probe to the outer regions of the inspection volume. To overcome these scanning restrictions, Hitachi-GE proposed the use of two primary detection techniques, a ‘straddle’ technique and back-wall obscuration.

477. The straddle technique applies separate transmit and receive probes that are positioned circumferentially either side of the RIP. The beam angle, the position of the transmit and receive probes and the skew of the probe (the rotation of the probe about the surface normal) are calculated such that the specular reflection of the beam from the face of a defect is captured by the receive probe.

478. The back-wall obscuration technique for the detection of circumferential planar defects relies on the plane of the defect being misaligned by approximately 30° to the dome/petal surface normal. A large misaligned defect will present a significant obstacle to a 0° ultrasonic beam deployed from the outer surface thereby reducing the amplitude of the signal reflected from the inner surface (the back-wall). The technical justification claims that a planar defect as large as the QEDS will result in a large

reduction of the back-wall echo such that the back-wall obscuration technique provides a reliable technique for defect detection.

479. IVC provided comments on Rev D2 of the technical justification to which Hitachi-GE provided a response on the comment sheet as to how each of the comments would be addressed. IVC then assessed whether the response had adequately addressed the comments. Hitachi-GE subsequently, issued revision 0 but there is no evidence that IVC reviewed this version of the technical justification to confirm that the comments had been addressed as proposed. I have noted that there are some open points where it appears that IVC did not accept Hitachi-GE's response and it is not clear as to whether these have been adequately addressed in the issued version of the technical justification.

480. In my opinion the back-wall obscuration technique does not meet the test for highly reliable detection on two grounds:

- It is not a widely used detection method for manufacturing defects.
- It is not founded on sound physical principles.

481. While the technique may provide useful information in evaluating signals found using other probes, the back-wall obscuration technique should not be considered as a primary detection method. I raised this matter with Hitachi-GE in regulatory query RQ-ABWR-1474 (Reference 138). In response, Hitachi-GE stated the back-wall obscuration technique will not be considered as a primary detection method but, as a secondary method, it is effective in confirming any signals that are observed using the straddle technique.

482. I concluded that there are significant portions of the weld where, due to scanning restrictions, the detection of circumferential defects relies on the deployment of a single, novel and complex technique. As such, I judged that there was a significant risk that the inspection as it stood would not meet the demands of a full qualification exercise. Consequently, I raised the two points on this matter in regulatory query RQ-ABWR-1474 (Reference 137), these are discussed in the following paragraphs.

483. Although the justification of the position of the weld is made in the RPV design philosophy report (Reference 140) I asked, in view of my assessment of the inspection issues, whether the design represented an ALARP position or whether changes could be made to improve the inspectability. Related to this, I asked whether the manufacturing route provided any earlier stages where the inspectability would be improved. Alternatively, I asked whether the specified manufacturing route could be modified to allow better access for inspection at an earlier stage. Hitachi-GE responded by stating that any modification to the weld position or to the manufacturing route would depart from that used on previous ABWRs and thereby presents a risk to quality.

484. The distance between the CRD penetrations and the weld in the vicinity of the RIPS was unclear and I therefore asked Hitachi-GE to provide this detail and to explore the possibility of performing scans from the inner surface. Hitachi-GE responded that, in the most limiting circumferential locations, the distance between the edge of the CRD penetration and the weld centreline was not sufficient to allow probes to be scanned from the dome side of the weld.

485. Hitachi-GE subsequently added a supplement to their response which included:

- an outline description of additional scans that could potentially be performed and would improve the overall capability for the detection and characterisation of defects. These options included the deployment of phased array probes from the

bores of the CRD and RIP penetrations and scans from the inner surface that utilised multiple reflections from the defect and external surface.
- results from practical trials performed on a test piece that represented the salient features of the component and contained deliberately implanted representative defects.

486. This discussion was raised late in the GDA process and hence Hitachi-GE's supplement was produced too late in GDA for Hitachi-GE to fully develop the proposals for the additional scans. Nonetheless, in my opinion these proposals have made a significant contribution to understanding how the inspection could potentially be improved through further development. Also the results gathered from the practical trials and presented in the supplement arrived too late for me to undertake a thorough assessment. I do note however that the results appear to provide good general agreement with the calculated signal amplitudes presented in the technical justification.

487. To ensure that this is taken forward I have raised the following assessment finding:

AF-ABWR-SI-13 The petal to bottom dome weld area of the reactor pressure vessel is an area of physically restricted access and the initial proposals for inspection of this area are not demonstrably adequate to support the Safety Case. Because of this, the licensee shall ensure that the inspection capability in this area is adequate to support the safety case.

488. Overall, I am satisfied that the information presented by Hitachi-GE in the final response to RQ-ABWR-1474, provides sufficient mitigation of the risk that I identify above – that the inspection may not meet its required objectives.

4.4.1.12 END OF MANUFACTURE INSPECTION OF ABWR MSL NOZZLE CROTCH CORNER

489. The crotch-corner of the main steam nozzle is a highly stressed region and as such specific assurance is required that there are no defects at the end of manufacture that could grow in-service and threaten structural integrity. Hitachi-GE identified that the only defects that should be considered are those that are surface breaking and are principally aligned in a nozzle radial plane; both smooth and rough defects are considered.

490. The QEDS is defined as 10mm x 20mm, which is significantly smaller than the SLLDS of 18.7mm x 37.4mm. There is therefore potential to increase the QEDS to increase the margins of the inspection for the detection and characterisation of defects.

491. Revision 0 of the technical justification (Reference 134) was produced relatively early in the GDA process. I provided detailed comments to Hitachi-GE on this revision of the technical justification to aid the understanding of what was expected for GDA. An important comment at this stage related to characterisation where the technical justification did not describe the process for evaluating the defects as either planar or volumetric. Revision 0 of the technical justification had been independently reviewed by IVC prior to my review (Reference 135); the conclusion of this review was that most of the comments raised on the prior version had been addressed and the only remnant comments were those that would be dealt with during a full qualification.

492. The inspection consists of pulse echo techniques deployed from the inner surface using angle compression wave probes for detection, characterisation and sizing with time of flight diffraction (TOFD) used to supplement through-wall sizing. The surface of the cladding is ground to provide a suitable surface for scanning probes.

493. The technical justification presents a thorough analysis of the effect of the component geometry on the mis-orientation of the ultrasonic beam with respect to the defect.

494. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects at or larger than the QEDS). My judgement is essentially based on two considerations: first the inspection techniques are based upon sound physical principles and use well understood and established techniques; Hitachi-GE have satisfactorily considered (within the context of GDA) the impact of the geometry on the inspection performance.

4.4.1.13 END OF MANUFACTURE INSPECTION OF ABWR RPV MAIN STEAM NOZZLE TO VESSEL WELD

495. The main steam nozzle is joined to the RPV via a set through nozzle. Both items are fabricated from SA 508 low alloy steel which have low ultrasonic attenuation and the inner surface is clad with a 4mm layer of stainless steel. The main features of the weld that affect ultrasonic inspection are:

- The weld fusion faces are aligned parallel to the nozzle axis over the majority of the thickness – at some circumferential locations there are significant regions of where the fusion faces are at 12.5° to the nozzle axial direction.
- The weld is relatively thick (nominal thickness approximately 180mm including the cladding).
- The weld has a ‘saddle’ geometry.
- The change of section on the nozzle imposes significant scanning restrictions.
- For specific manufacturing defects given in the technical justification, defects are principally aligned with the radial direction (within 5° of the radial direction). Defects that have their principle directions in the welding direction and transverse to the welding direction are considered. Defects are considered to be smooth or rough.

496. The defects of concern are lack of sidewall fusion and cracks with the latter occurring in the longitudinal (welding direction) and transverse directions. The principle inspection objective is to detect and reject planar defects that are larger than the QEDS of 15mm x 90mm (longitudinal defects) and 15mm x 78mm (transverse defects).

497. Revision 0 of the technical justification (Reference 132) was produced relatively early in the GDA process. I provided detailed comments to Hitachi-GE on this revision of the technical justification to aid the understanding of what was expected for GDA. The most important comment at this stage related to the characterisation where the technical justification did not describe the process for evaluating the defects as either planar or volumetric. In response, revision 1 was produced (Reference 153) for which I produced a further round of comments.

498. The final version, revision 2 (Reference 133) was subject to the independent review by the IVC. From the comments sheet provided to ONR (Reference 150) I note that IVC raised a relatively large number of comments, which were addressed to the satisfaction of IVC. IVC observed some confusion over the numbering of the different versions but I do not consider these to be material to my assessment in the context of GDA.

499. The proposed inspection techniques described in the technical justification comprise a combination of direct pulse echo, tandem and corner trap techniques for detecting the defects of concern. The tandem inspection is designed specifically to detect embedded defects that are aligned close to the vessel radial direction. For detection, the probes are scanned using automated techniques from the external surface of the vessel and from the nozzle bore; the option of scanning from the internal surface is considered for defect evaluation (characterisation, sizing and location).

500. The technical justification has addressed the geometric effects of the saddle geometry on the inspectability and includes a consideration of the impact on the alignment of the transmit and receive probes for the tandem inspection.

501. The technical justification presents some parametric studies on the effect of cladding on the performance of those techniques where the ultrasonic beam propagates through the cladding. These studies are also referred to in later technical justifications where similar inspection situations exist. The studies used a test piece that had a 12mm layer of strip-cladding on the far surface (relative to the inspection surface) and measured the reduction in signal amplitude due to the propagation of the ultrasound through the cladding. The main conclusion from these studies is that the cladding had little effect for instances where the beam passed through the uniform region of the cladding (in between the strip overlap regions). In contrast, losses of approximately 12dB were seen when the ultrasound passed through the strip overlap regions. The technical justification argues that the actual thickness of cladding in the ABWR RPV is small, compared with the 12mm on the test pieces and therefore the loss of signal will be proportionally lower. However, there are additional factors that have not been considered in the technical justification. Firstly, the grain structure at the strip overlap regions is less homogeneous leading to a greater disruption of the ultrasonic beam and secondly the inner surface is less smooth in the vicinity of the overlap such that the reflected signal is reduced. While the strip overlap region is small compared with the strip width, the impact on those techniques that rely on a reflected signal from the inner surface of the RPV could be significant. The impact could potentially be reduced by removing the surface undulations at the clad strip overlap through machining or grinding the inner surface and the merits of this should be investigated further.

502. I have raised the following assessment finding to ensure that this matter is explored further along with other aspects of a design for inspectability approach.

AF-ABWR-SI-14 Because it is not apparent that the inspection conditions for very high integrity components reduce risks so far as is reasonably practicable, the licensee shall review all inspection conditions for all very high integrity welds to see where improvements could be made. This should include, but not be confined to, further investigations on the effect of cladding surface finish, the surface condition of the inspection surface and the locations of obstacles.

503. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on two considerations: first the inspection techniques are based upon sound physical principles and use well understood and established techniques; Hitachi-GE have satisfactorily considered (within the context of GDA) the impact of the geometry on the inspection performance.

4.4.1.14 END OF MANUFACTURE INSPECTION OF ABWR RPV MAIN STEAM PIPE TO MSIV WELD

504. The main steam line between the main steam nozzle and the inboard main steam isolation valve is classified as VHI and Hitachi-GE have included it in their examples within GDA to support the avoidance of fracture case. In selecting the location on the main steam line for more detailed assessment Hitachi-GE separated the ranking for inspectability and defect tolerance. The DTA was undertaken on the most limiting weld locations from a defect tolerance perspective and the derived QEDS at this location was then applied throughout the whole main steam line. The most limiting region from an NDE perspective was then selected using this QEDS for considerations within GDA (Reference 126).

505. The site chosen as the most challenging for the manufacturing NDE was the main steam line to MSIV; the main reasons for this choice being that the ultrasonic attenuation is expected to be significantly higher on the casting (MSIV) side of the weld and the change of section on the MSIV body restricts scanning.

506. The weld under consideration joins the forged carbon steel main steam line to the cast carbon steel MSIV body and has a nominal thickness of approximately 34mm. In my assessment of revision 0 of the technical justification (Reference 127) I noted that there was not a clear link between the requirements of the NDE and the objectives of the inspection in relation to the structural integrity case. I therefore raised RQ-ABWR-1154 (Reference 143) seeking Hitachi-GE to describe how the NDE performance requirements would ensure that planar defects of structural concern (those larger than the QEDS) would be detected and rejected. I also noted that the technical justification stated:

- the tilt of circumferential cracks to be within 5° of the radial circumferential plane and while this may be relevant for cracks located in the weld, I did not believe that it represented the case for cracks located in the HAZ.
- that the face of cracks would always be rough whereas I considered that the defect description should encompass smooth cases.

507. I therefore raised regulatory query RQ-ABWR-1155 requesting Hitachi-GE to provide evidence for these defect descriptions and to clarify its position on this matter. In response, Hitachi-GE explained that the description as given in the technical justification represented the worst case. The technical justification was subsequently revised (Reference 128) such that cracks located in the HAZ were aligned principally along the fusion face and to include the potential for smooth cracks.

508. The QEDS is defined as 4mm x 24mm, which is significantly smaller than the SLLDS of 7.8mm x 46.8mm. It appears therefore that the margins for detection and characterisation of defects could be increased by increasing the QEDS to be closer to the SLLDS.

509. IVC provided comments on revision 1 of the technical justification to which Hitachi-GE provided a response. IVC then assessed the adequacy of the response and provided its conclusions on the comments sheet dated 5/01/2017. Most of the responses have been accepted to the satisfaction of the IVC and for those that have not, it is my opinion that they do not affect the overall conclusion for the inspection. The agreed changes appear to have been carried forward to the final technical justification (Reference 128).

510. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on two considerations: first the design of the weld enables sufficiently good access for ultrasonic inspection; secondly the inspection techniques are based upon sound physical principles and use well understood and established techniques.

4.4.1.15 END OF MANUFACTURE INSPECTION OF ABWR MSIV REPAIR WELD

511. Reference 141, which was called a capability statement, rather than a technical justification, stated that the inspection of repair welds in the MSIV valve body would not be subject to a future full qualification. Hitachi-GE argued that, since ASME XI did not require qualification of such repair welds, a qualification of the end of manufacture inspection using the principles of ENIQ, was not necessary. This approach was inconsistent with my expectation that, since manufacturing defects could potentially exist in the repair welds (of a VHI component) of a size that would be of structural

concern, the capability of an inspection to detect and reject such defects should be confirmed through a process of rigorous qualification. This qualification should include both inspection procedures and personnel as defined in Hitachi-GE's strategy for inspection qualification of VHI components (Reference 122)

512. Following discussions on this matter, Hitachi-GE stated that they would separate repair welds in to major and minor categories, with only the inspections of major repair welds being subject to full inspection qualification.

513. I sought confirmation of Hitachi-GE's revised approach for this inspection through regulatory query RQ-ABWR-1497 (Reference 142) which included the following points:

- Hitachi-GE needed to confirm the position on qualifying inspections used to ensure the absence of defects larger than the QEDS.
- The approach used to categorise weld repairs as major and minor should be defined.
- The overall approach for ensuring that repair welds were as defect free as possible should be described.

514. In response to RQ-ABWR-1497 (Reference 142) Hitachi-GE confirmed that all repair welds that were classified as major would be subjected to qualified ultrasonic inspections. The threshold for major repair welds was any weld that was deeper than the QEDS (14mm) which is lower than the SLLDS (14.5mm). Since a welding manufacturing defect cannot be deeper than that of the repair, this threshold means that no minor weld repair could contain a defect that could be of structural concern.

515. In practice, Hitachi-GE will apply the same NDE techniques for major and minor repair welds, the only difference is that the procedures and personnel for minor repair welds will not be subject to full qualification.

516. While Reference 141 is titled as being a capability statement, it is written in the style of an ENIQ technical justification and follows closely the recommended contents described in Reference 124. As I had received a satisfactory response from Hitachi-GE to RQ-ABWR-1497 regarding the revised status of the inspection and because I considered the capability statement to provide sufficient content for the purposes of GDA, I did not seek a revision to the document. From hereon I refer the capability statement as a technical justification to reflect the revised status of the inspection.

517. The MSIV body is a single piece casting made from SA-216 Gr WCB carbon steel with any repair welds being made from carbon steel. Although some areas may be more susceptible to casting defects than others, in principle a repair weld can exist anywhere in the casting. Consequently, the relative ease of inspection will depend upon the location of the repair and there are some areas where scanning ultrasonic probes will be severely restricted.

518. The technical justification states that, in general, defects will be removed by excavating the surface that is closest to the defect and consequently the maximum depth of the weld repair will be half of the section thickness. There are some regions of the casting where this is not possible due to access restrictions for excavating and welding. The shape of the weld repair, when viewed on to the surface, will be rectangular and will be oriented such that the sides of the rectangle will be along the principal directions of the casting (axial and circumferential). The angle of the excavation fusion faces will depend upon the depth; deep excavations will have a fusion face that is at 60° to the surface whereas the fusion faces for shallow excavations will be at 20°. I note that the specified shape and orientation of weld repairs will make it easier to control the ultrasonic inspection.

519. The specified defects of concern include only lack of sidewall fusion defects, and while these may be considered to be the most likely type of planar defect of concern, there exists the possibility that cracks could occur as is considered by Hitachi-GE for other VHI welds. In any case, the inspection will need the capability to discriminate between volumetric and planar defects for all locations within the repair. I believe that the ultrasonic techniques described in the technical justification could readily be adopted for the detection and rejection of crack-like defects at any location in the weld (or HAZ).

520. The technical justification presents a thorough analysis of both the general case for weld repairs and also for the locations that are considered particularly difficult due to access considerations. Specifically, the technical justification considers those issues that have detrimental effects on the inspection performance and then selects regions of the MSIV body where these issues are most prominent. From these regions Hitachi-GE defined the 'crotch region' of the casting to be most limiting and concentrated its analysis on this region.

521. The IVC reviewed revision D1 of the technical justification and then assessed the adequacy of Hitachi-GE's responses as presented on the comments sheet (dated 22/05/2017). While there is no explicit statement by IVC that they have checked all of the comments have been adequately addressed, it appears that the IVC comments have been addressed in moving to revision 0 (dated 30/05/2017) of the technical justification (there is a statement to this effect in the technical justification).

522. Overall I am satisfied that Hitachi-GE have demonstrated, to an extent that is appropriate for GDA, that a NDE inspection can be developed and qualified that will meet the primary objective of detecting and rejecting defects of structural concern (planar defects larger than the QEDS). My judgement is essentially based on the following considerations: the design of the excavation has been considered so as to provide for a controllable inspection; the inspection techniques are based upon sound physical principles and use well understood and established techniques; the technical justification has been reviewed by IVC with all comments being satisfactorily addressed. In my opinion Hitachi-GE should consider the inclusion of cracks amongst the defects of concern in a similar way as for other VHI welds and I have raised an assessment finding to this effect.

AF-ABWR-SI-15 The inboard Main Steam Isolation Valve casings are very high integrity components; because the defect description proposed by Hitachi-GE are essential in defining whether repair welds will be necessary, confidence in these descriptions underpins confidence in the component. The licensee shall review and justify defect descriptions in the MSIV casing, including consideration of whether crack-like defects in the weld volume and heat affected zone should be considered.

523. In response to RQ-ABWR-1497, Hitachi-GE have confirmed that the reliability of the inspection will be confirmed through formal inspection qualification. Therefore eventually the inspection will be subject to the rigours of full qualification in accordance with the principles of the ENIQ methodology. Here the limited evidence presented in GDA will be expanded to consider more locations and to provide specific practical and theoretical evidence in support of the inspection capability.

4.4.1.16 CONCLUSIONS: QUALIFIED INSPECTIONS

524. I have sampled the evidence provided by Hitachi-GE that has been used to support the claim that end of manufacture NDE will provide high confidence that VHI components will enter service free of structurally significant defects (manufacturing defects that could potentially grow through service to a point where they could threaten structural integrity).

525. The basis of my assessment, within the context of GDA, has been to establish with sufficient confidence that ultimately a highly reliable NDE inspection could be developed to detect and reject structurally significant defects as determined by fracture mechanics with a significant safety margin. My criteria for high reliability are:

- End of manufacture NDE should apply sound physical principles and deploy techniques that are mature and for which there is significant industrial experience.
- Components, should be designed and manufactured, as far is reasonably practicable, to be readily inspectable. In practice, this means that the overall design (materials and geometry) and manufacturing process (for example surface finish) should facilitate access for the application of NDE.
- Manufacturing defects of concern should be detected and evaluated with good margins.

526. An important aspect of my assessment has been to consider whether the high reliability of the NDE could eventually be demonstrated through a rigorous process of qualification.

527. My conclusions with regard to the end of manufacture NDE are:

- Hitachi-GE have presented a good explanation of an approach to qualifying the end of manufacture NDE that is, in line with UK practice, and meets my expectations.
- The technical justifications, used by Hitachi-GE to present the inspection proposals and the evidence for their effectiveness, were produced according to the relevant ENIQ recommended practice. As such these documents form a good basis for taking the inspections through to full qualification.

528. I am satisfied that, in general, and within the context of GDA, Hitachi-GE have provided sufficient information within the technical justifications and in their responses to regulatory queries, to demonstrate that high reliability end of manufacture NDE could be developed for VHI components.

529. Additionally, I am satisfied that it should be possible to demonstrate the capability of the NDE techniques, when fully developed, through a rigorous process of inspection qualification.

530. Ultimately, the inspection will be subject to the rigours of full qualification in accordance with the principles of the ENIQ methodology. Here the limited evidence presented in GDA will be expanded to provide specific practical and theoretical evidence in support of the inspection capability.

531. There are some specific areas where I have identified that significant further development is needed. To ensure that these are taken forward and are satisfactorily resolved I have raised assessment findings to progress these matters.

Avoidance of Fracture Demonstration (Reconciliation) and Key Assessment Considerations

532. The themes of my assessment for RO-ABWR-0001 so far have included material properties, DTA and manufacturing inspection. In this part I consider how Hitachi-GE have reconciled evidence relating to these topics in its safety case for VHI Components, as reported in their reconciliation document (Reference 84). In addition, based upon the discussions in the preceding parts, I present my key assessment considerations and judgements relating to the evidence provided by Hitachi-GE in response to RO-ABWR-0001.

533. Hitachi-GE identified the VHI components, namely the RPV; a section of the main steam pipework from the RPV to the MSIV (inboard of containment); and the inboard MSIV. In my assessment I regard these as equivalent to the highest reliability components described in the SAPs. Therefore, I consider that the stringent principles EMC.1 to EMC.3 apply. An expectation under EMC.1 is that highest reliability structures and components are as defect free as possible and defect tolerant. RO-ABWR-0001 was raised to provide guidance to Hitachi-GE in meeting the UK regulatory expectations relating to their avoidance of fracture demonstration with the emphasis on demonstrating defects tolerance.

534. For each VHI component, Hitachi-GE developed a safety case that integrates qualified inspection with limiting defects sizes, determined by analyses that apply conservative material properties. The objective is to show that limiting defect sizes are larger than those that can be reliably detected and characterised. Hitachi-GE used the term DSM to describe the margin between these parameters with an objective to demonstrate a minimum DSM value of 2.0.

535. The target of a minimum DSM of 2.0 accords with approaches previously adopted in the UK. Hitachi-GE conclude that a minimum DSM of 2.0 has been demonstrated for all the limiting design locations evaluated in GDA and that the QEDS values are reconciled with the SLLDS values derived from DTA:

[redacted]

Note: SLLDS = ((ELLDS/2) - LFCG) and QEDS ≤ SLLDS

536. I note in all locations a DSM of at least 2 is available and for some locations a DSM significantly exceeding 2 as indicated by any additional margin between the QEDS and SLLDS values is claimed e.g. the MS piping weld (effective DSM of ≥ 3.5).

537. I welcome the demonstration of additional margins provided that the capability to undertake qualified inspections with high reliability using standard techniques is not compromised. I therefore questioned Hitachi-GE on the role of this additional margin in their sample DTA's. Hitachi-GE confirmed their priority was to establish a DSM of 2 with SLLDS at least equal to the QEDS value (Reference 96). I was satisfied with this response and note that in general the DTA results suggest good defect tolerance for the VHI components.

538. An exception was the RPV top head weld where with a minimum section thickness of the order of 100mm the QEDS was approximately 10% of the wall thickness and relatively small compared to the RPV beltline and bottom head locations. In my opinion a QEDS value of 10mm is achievable, but is approaching the limits for reliable detection using readily available NDT techniques.

539. I made this observation late in my assessment and asked my TSC, to undertake some R6 sensitivity studies to identify the cause(s) of the apparently low defect tolerance at the RPV top head weld. I determined, with TSC support, that the low value was primarily due to Hitachi-GE categorising the stud loads as a primary stress, which together with the stresses arising from internal pressure effectively doubled the membrane stresses local to the top head to flange junction. There is some uncertainty as to whether these stresses are primary or secondary in nature, but Hitachi-GE had adopted a conservative approach. In addition, they had introduced further conservatism in their DTA by assuming local rather than global collapse of the un-cracked ligament.

540. In RQ-UKABWR-1520 (Reference 96), I also raised several additional points relating to Hitachi-GE's reconciliation report. These included the need for a visible route to the evidence underpinning the material property input to the DTAs and a demonstration that these are conservative to support reconciliation; evidence that overall the Hitachi-GE DTAs retain adequate levels of conservatism; and clarification of the main basis of the Hitachi-GE claims for reconciliation. Hitachi-GE provided adequate responses to these questions to close RQ-UKABWR-1520 (Reference 96). In particular, Hitachi-GE highlighted the work of the expert panels in challenging the inputs to the DTAs to ensure that overall the approaches remained appropriately conservative.

541. I was also aware that ONR had raised UK-ABWR-RO-0068 which related to revised seismic modelling of the civil structures. I questioned Hitachi-GE on the implications, if any, for the plant design and the imposed seismic loads on VHI components used in their DTAs. Hitachi-GE confirmed the seismic loads were in general small compared to the pressure and mechanical loads such that the SLLDS (and hence QEDS) values were unlikely to be challenged. However, Hitachi-GE is committed to revisiting the effect of seimic analysis on Structural Integrity during site licensing. ONR will follow this up as part of normal business. I was satisfied with the Hitachi-GE response (Reference 96).

542. Key questions relating to Hitachi-GE's reconciliation on the NDT side included the need to provide evidence within the GDA that a revised QEDS of 20mm for the RPV beltline was achievable. This was progressed through RQ-UKABWR-1516 along with consideration of Hitachi-GE's revised proposals for qualification of the manufacturing inspections. In addition, Hitachi-GEs initial proposal not to qualify the MSIV weld repairs did not meet my expectations this was resolved through RQ-UKABWR-1497 and subsequent discussion in the manufacturing inspection section above.

543. From my preceding discussions salient points emerged which could also potentially affect the reconciliation. I list these points along with my considerations of their effect, if any, on the veracity of Hitachi-GE's reconciliation for the fracture avoidance demonstration below:

- uncertainty in the upper shelf fracture toughness values for the RPV weld metal and MSIV casting materials, Hitachi-GE performed sensitivity studies for the RPV beltline and MSIV, in both cases the change in the ELLDS value was gradual and modest with the absence of a 'cliff-edge' effect this together with the conservatism in the Hitachi-GE approaches provides a basis for confidence that there is some provision for reduced fracture toughness. These are commercial risks and I have raised Assessment Findings as appropriate for the future licensee to either justify a conservative lower bound upper shelf fracture toughness value consistent UK practice or show that there are adequate margins in the DTAs to accommodate any non-conservatism in the LFCG estimates.

- uncertainty in the FCG transient accounting - the LFCG values are small in the majority of cases and considering the DSM values are at least 2, and that Hitachi-GE have adopted conservative approaches, I expect that any increase in LFCG, if needed, can be accommodated without placing undue demands on the QEDS values, but have raised an Assessment Finding to progress this.

- the reduction in the QEDS from 30mm to 20mm for the RPV beltline was resolved through Hitachi-GE providing further evidence in a revised TJ that qualification of the lower QEDS value was likely to be achievable using readily available NDT techniques.

544. Overall I am satisfied that Hitachi-GE have provided adequate and sufficient evidence for reconciliation in their fracture avoidance demonstration to close action RO-ABWR-0001.A4. My conclusion is based on:

- The DSM margins in the DTAs – indicative of good defect tolerance.
- The comparison work of the IVB along with the challenge in the expert panels to ensure appropriately conservative DTA methods.
- The results of my independent comparative work.
- Adequate responses to my RQs.
- The conclusion that qualification of the QEDS values is achievable.

545. My assessment of the DTAs provided by Hitachi-GE for a sample of VHI welds is described above, where I conclude that the outcomes of these analyses provide an acceptable basis for reconciliation with the proposal for qualification of the NDT system for the GDA.

546. Consistent with UK good practice for treatment of the highest reliability components, Hitachi-GE applied the R6 procedure to determine limiting defect sizes for the set of VHI welds and nozzle crotch corner locations considered in GDA.

547. I have previously noted that the R6 procedure is generally less prescriptive than the methods used by Hitachi-GE. Thus application of R6 sometimes requires expert interpretation and judgement to confidently establish a valid result for which the degree of conservatism is well understood. I have observed that Hitachi-GE have improved its proficiency in using the R6 defect assessment procedure and in my opinion this will be of significant benefit when undertaking assessment of the increased number of VHI locations to be considered in future for site licensing. In particular, I consider that this will promote resolution of assessment finding AF-UKABWR-SI-08:

548. I conclude that it will be possible to undertake high reliability manufacturing inspection of VHI components and that qualification of the inspection system will be suitable to confirm that defects of structural concern, identified by fracture mechanics, will be detected and rejected.

549. I am therefore content that Hitachi-GE have provided an adequate avoidance of fracture demonstration for the sample of VHI welds considered in GDA. I consider that Hitachi-GE has applied, in general, conservative methods in their DTAs and sufficiently robust methods in the development of their TJs to provide a basis for confidence in the future qualification of the manufacturing inspections. In conjunction, these establish an acceptable margin between limiting defect sizes and the defect sizes that are expected to be reliably detected.

550. My assessment has been guided by ONR's SAPs, see Annex 1 and TAGs, see Annex 2. A notable example of RGP adopted by Hitachi-GE included the application of the R6

procedure in their DTAs and the application of the ENIQ methodology to confidently establish that NDT can be qualified in future.

551. Overall, I am satisfied that Hitachi-GE have provided adequate and sufficient evidence to close UK-ABWR-0001.

552. I have raised assessment findings in my assessment of RO-UKABWR-0001. These matters do not undermine my judgements regarding the generic safety submission and generally relate to future licensing.

4.4.2 RO-ABWR-0002

4.4.2.1 BACKGROUND

553. The control rod drive (CRD) penetrations through the bottom head of the reactor pressure vessel consists of:

554. A nickel alloy CRD stub tube welded via a full penetration weld to the nickel alloy cladding of the low alloy bottom head

555. A stainless steel CRD housing welded to the nickel alloy CRD stub tube welded via a partial penetration weld

- The pressure boundary welds associated with the CRD penetration are not classified as VHI welds, however, due to the complexity of the design it will be necessary to demonstrate the initial and through life integrity of this pressure boundary even as Standard Class 1 welds. The demonstration of initial and through life integrity will need to address a number of related aspects including:
- An explanation of loading mechanisms applied to the penetration including pressure, dead weight, thermal transients, and should include seismic loads and faulted load conditions.
- A demonstration of pressure vessel design code compliance, including stress, fatigue and any specific requirements associated with welding directly onto clad material.

556. An understanding of the inspection approaches to be employed both at manufacture and through life in order to demonstrate that the penetration has been designed to be adequately inspectable

557. A demonstration that the material choices and stress states of the materials are suitable for minimising the potential for through life degradation

558. An understanding of the operational experience with this design of penetration

559. I raised RO-ABWR-0002 in order obtain this information from Hitachi-GE and provide a route to demonstrating a suitable Safety Case. The Actions from the RO were as follows:

- **Action 1.1:** Hitachi-GE should provide evidence of operational experience (OPEX) from plants worldwide operating with this design of CRD, or substantially similar designs. This should include:
    - o Incidences of degradation
    - o Incidences of failure
    - o OPEX on inspection (e.g. ease of application, and confidence in results)
    - o Manufacturing OPEX

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- **Action 1.2:** Hitachi-GE should provide evidence that the stress states are suitable for minimising the potential for through-life degradation.
    - o Limiting locations should be identified.
    - o Likelihood of degradation for the limiting locations should be assessed
- **Action 1.3:** Hitachi-GE should identify where the demonstration of suitable materials choices has been made.

4.4.2.2 HITACHI-GE SAFETY SUBMISSION

560. Hitachi-GE's Safety Case. Hitachi-GE presented their responses to the RO through the following documents:

Action 1.1	CRD Penetration Design Justification Report (Reference 68).
Action 1.2	Stress Analysis Report of Control Rod Housing Penetration for GDA (Reference 69).
Action 1.3	Material Selection Report for Nuclear Boiler Systems (CRD Housing and CRD Stub Tube parts only) (Reference 70).

4.4.2.3 ONR ASSESSMENT

561. Hitachi-GE have proposed addressing Action 1.1 with the CRD Design Justification Report, of which there have been three revisions. This has been reviewed in detail by my TSC; no RQs were raised against them. From my review of the final revision of the CRD Design Justification Report I consider that adequate discussion has been provided to address the intent of the RO. This has covered explicitly the incidences of failure and degradation in the worldwide data and, in my judgement, the listings provided are adequately complete.

562. The discussion on operational experience for manufacture and inspection are at a lower level of detail than those relating to operational experience on degradation and failure. There is no significant discussion on the manufacturing operational experience of the CRDs, although I do note that a suitable manufacturing route has been demonstrated for Japanese deployments of the ABWR. Manufacturing confidence is presented by Hitachi-GE based upon reference to ASME compliance alone. Whilst ONR would not normally expect levels of Structural Integrity significantly in excess of ASME code compliance for components with a classification of standard class 1, the demonstration that this is ALARP is also expected. I consider that what has been supplied within GDA is sufficient, however, I consider that the licensee should look to provide a review of manufacturing OPEX before the purchase of CRD materials occurs as part of normal business. This will be f ollowed up as part of normal business.

563. The discussion on inspection covers only the Hitachi-GE plans, which are based on the ASME criteria and are not linked directly to the requirements of the Safety Case. The section on ease of application is at a high-level and simply states:

"Sufficient access is provided to CRD penetrations welds for performance of the visual VT-2, examination during the system leakage, and system hydrostatic examinations."

564. There is no discussion on whether visual inspection is adequate, how this relates to the Safety Case and what "*sufficient access"* means in practice. Given that the consequences of failure of these items are tolerable, I judge that this is not significant enough to prevent a DAC being issued. However, I consider that this will need to be

addressed in detail in the licensing phase. I have, therefore, raised the following Assessment Finding:

AF-ABWR-SI-16 – The inspectability of the Control Rod Drive penetrations provides a key part of their Safety Case. Because design for inspectability was not covered in sufficient detail during GDA, the licensee shall assess whether the inspectability of the Control Rod Drive penetrations is adequate to support the Safety Case, and whether it could be improved to reduce risks so far as is reasonably practicable.

565. The stress analysis report for the control rod housing was submitted by Hitachi-GE to address Action 1.2 of RO-ABWR-0002. This has been reviewed by my TSC, and comments were raised against the document in RQ-ABWR-1381. For the purposes of GDA, Hitachi-GE have submitted a response to this RQ, as reviewed in this report. I consider, therefore, on the basis of the responses provided to this RQ this Action to have been successfully closed.

566. Action 1.3 was raised to ensure that the evidence for this component was raised within GDA, but to ensure that duplicate submissions were not made between this submission and those for RO-ABWR-0035. I judge that the action has been completed appropriately through the submission of the Material Selection Reports.

4.4.2.4 CONCLUSION

567. I judge that Hitachi-GE have presented sufficient evidence for Step 4 of GDA, through the submitted documents and the responses to the associated RQs. The licensee should, however, seek to demonstrate suitable inspectability of the CRD penetrations during the site licensing phase, as outlined in the assessment finding above.

4.4.3 RO-ABWR-0003

4.4.3.1 BACKGROUND

568. During Step 2 of GDA, ONR raised RO-ABWR-0003. This related to the main pressure boundary and support structure of the Reactor Pressure Vessel (RPV). The RPV has been designated, by Hitachi-GE, as being a Very High Integrity (VHI) component, for which their failure is not tolerable. ONR's expectations for this component are, therefore, in line with the Safety Assessment Principles (SAPs), notably EMC.1 to EMC.3. The ONR SAPs contain explicit discussion of the RPV, thus:

The RPV will need to have a very low frequency of gross failure. However, such low frequencies cannot be demonstrated using actuarial statistics because of a lack of data, and cannot be plausibly or confidently estimated using theoretical modelling. Instead the approach is one of sound engineering practice that gives a high level of confidence in the ability of the vessel to deliver its safety functions throughout its life.

569. Part of this demonstration is for the RP to show that '*the metal component or structure is as defect-free as possible'*. This will require demonstration of control and understanding of the product form being used for the RPV, and how this minimises the occurrence of defects.

570. UK normal practice, both from SZB and other plant already having been through GDA, forgings have been used wherever possible. Hitachi-GE have, however, proposed the use of an RPV head made from plate material welded together into the head formation. Welds are, generally, areas of elevated defect occurrence compared with the parent material and so are expected to be minimised during the design of the plant. This is outlined in the SAPs via EMC.9, which states:

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.

571. There is a need, therefore, to ensure that the RPV pressure boundary and support structure has the highest level of integrity and that the appropriate product form is selected. This should minimise weld length, and hence defect rates, and facilitate inspection of the component.

572. In order to satisfy this need, it is my general expectation that the RPV will, where possible, be manufactured from 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.

4.4.3.2 SCOPE

573. RO-ABWR-0003 was raised to provide Hitachi-GE with ONR's expectations as to what data might be required to meet UK expectations. RO-ABWR-0003 stated that evidence will be required to show that:

- forgings have been chosen that minimise the number and length of the welds in the main pressure boundary of the Reactor Pressure Vessel and its support skirt, and that the weldments will avoid locations of high stress or neutron irradiation;
- consideration has been given to the use of integrally forged nozzles and flanges.

574. Should welded plate material be proposed for areas of the main pressure boundary, including the RPV head, or the support skirt, rather than forgings, then a detailed justification will be required to demonstrate why this proposal reduces risks So Far As Is Reasonably Practicable (SFAIRP) taking into account matters such as the material properties of the plate compared with forgings, the propensity for defects in a plate compared with forgings and the additional weld length required as a result of using multiple plates rather than larger forgings.

575. Two regulatory Actions were raised against Hitachi-GE for this RO:

- *Action 1.1. Where plate material is proposed for the main pressure boundary, including the RPV head, provide evidence that:*

  Hitachi-GE have balanced the safety risk from increased weld length and defect occurrence rates, with safety benefit from improved control of the production methodology, improved materials properties and any other factors, to demonstrate that the proposed design maintains risks ALARP. Hitachi-GE should give consideration to the fact that the RPV is a component of the highest reliability and apply the principle of gross disproportionality to any non-safety-related matters.

- *Action 1.2. Where nozzles are proposed that are not integral to the RPV, provide evidence that:*

  Hitachi-GE have considered the safety risk posed in the introduction of extra weld lines and evidence to show whether the proposed design maintains risks ALARP.

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576. Hitachi-GE provided one key submission to address these RO Actions, which was the Design Philosophy Report for the RPV (Reference 26). This provided a head document for other subordinate documentation to reference out from.

4.4.3.3 HITACHI-GE SAFETY CASE

577. The key submission from Hitachi-GE has been the RPV design philosophy report. New revisions of this document have been issued over the course of the GDA to address the GDA actions raised and to make a demonstration of the design maintaining risks ALARP. The final assessment has been made against Revision 3. A summary of the Hitachi-GE position is given here; full details are available through Reference 26.

578. The general design of the UK ABWR RPV is a development of BWR technology, which has been in operation since the 1960s. Hitachi-GE claim that the UK ABWR design is a significant improvement over previous designs in terms of safety and reliability. Hitachi-GE have taken cognisance of the UK expectations that classifications above the requirements of an ASME standard class 1. This is explained within the Structural Integrity classification report (Reference 7).

579. Hitachi-GE state that they have progressed their design philosophy through compliance with four matters:

1. Compliance with the ASME Code, Section III Appendix G
2. Designed to comply with R6
3. Consideration of thermal shock
4. Consideration of under-clad cracking

580. Avoidance of through-life degradation is also considered within the document, although this is considered more thoroughly in the materials selection process, covered under submissions made through RO-ABWR-0035 and discussed later within this report.

581. Specific sections have been included on design optimisation for the UK regulatory context. These address the optimisation of the number and location of RPV shell welds and the optimisation of the design of RPV penetrations. These relate directly to the actions in RO-ABWR-0003. Hitachi-GE consider that the contents of Reference 26 fully satisfy the requirements of RO-ABWR-0003, when considered in association with RQ-ABWR-1290 (Reference 71), which gave comments on the previous versions of the document.

582. I note that Reference 26 contains specific appendices on the certain matters raised by ONR, some through RO-ABWR-0003, and some through routine interactions. These appendices include:

B. RPV bottom head weld location design optimisation

C. Justification for use of Set-in nozzles

D. Feasibility study [into using forgings in the production of the top head]

4.4.3.4 ONR ASSESSMENT

583. RO-ABWR-0003 was raised because the differences between the proposed UK ABWR RPV design and the designs of previous UK plant, notably SZB, are quite different. It was not apparent that either the material product form, or the use of nozzles would meet UK expectations.

584. Following the initial submissions in support of closing Revision 0 of RO-ABWR-0003, I considered it appropriate to add actions to the RO to ensure that the specific UK

expectations for RPV design could be made more explicit and that the closeout of the RO could be made clearer.

585. The general comments made by ONR against the design philosophy document have been addressed via the RQ process, through RQ-ABWR-1290, outside of the RO. I do not cover those points in this section of the report. Moreover, I will consider the positioning of the bottom head weld, and its proximity to the core support shroud, outside of RO-ABWR-0003 in section 4.2.6. In this section I consider only whether the actions placed in the RO can be closed adequately.

586. Regarding Action 1.1: Hitachi-GE have proposed the use of the extant design in this instance and that this is ALARP. Hitachi-GE have provided evidence through the Design Philosophy Report (Reference 26), that the move to a single piece forging is not possible given current technology. I do not consider it ALARP to request that new manufacturing capabilities should be designed and commissioned for the purposes of production on the UK ABWR head alone. I also concur with Hitachi-GE that utilising what would be an unproven manufacturing route would carry with it a significant quality, and hence safety, risk. Moving to a series of forgings welded together would be possible. Hitachi-GE, however, state that the production of forgings would be difficult and could lead to defects orientated in the through-wall direction as opposed being parallel to the surface. This is demonstrated in appendix E of Reference 26 and I consider this demonstration to be reasonable. I consider, therefore, that Hitachi-GE have demonstrated that the balance of nuclear risk shows that remaining with a plate design is ALARP. I have discussed this with the head of specialism for Structural Integrity and he is in agreement.

587. Regarding Action 1.2: For the use of set-in and set-on nozzles, I note that the case presented in appendix C of Reference 26 is not as well formulated as for Action 1.1, above. This notwithstanding, I judge it to be adequate given the safety significance, generally of these nozzles. These nozzles are, generally of a safety classification below VHI, i.e. standard Class 1 components with failure that is within the design basis, and as such my general expectation is for compliance with the ASME code. This expectation does not, however, obviate the expectation that there should be design-for-inspection for all components, especially those with a significant safety role. Hitachi-GE have balanced the manufacturability against the inspectability and have, in all the cases presented demonstrated, at a high level, that the welds will be inspectable. The use of set-on nozzles, which reduce the inspectability of the component have been avoided completely.

AF-ABWR-SI-17 – The RPV nozzles have been designed according to the ASME code, but explicit demonstration of design for inspectability has not been given. Because of this, the licensee shall demonstrate that the nozzles are designed such that they facilitate inspection to support the Safety Case.

4.4.3.5 CONCLUSION

588. RO-ABWR-0003 has been ongoing since, nearly, the start of GDA step 2. It was originally due for closure at the end of step 3, but I revised it in line with the spirit of the original RO, in that it requested evidence. For this reason, I consider that it was not appropriate to close it until Step 4. At the end of Step 3, I added two Actions onto the new revision of the RO. I have received and reviewed documentation from Hitachi-GE, including independent review by my TSC and I am satisfied that the Actions have been closed out appropriately.

589. Hitachi-GE have provided specific documentation into the production of the RPV head, and into the use of set-in and set-on nozzles and their appropriateness for inspection. I consider that Action 1.1 is closed with no specific recommendations for licensing,

except normal surveillance appropriate for a component of the highest reliability (a VHI component).

590. For the use of set-in nozzles in the RPV, for the sample observed I am content that the intent of RO-ABWR-0003 is met. As per AF-ABWR-XC, during the licensing phase, detailed examination of the inspectability of these nozzles should be pursued as part of detailed design.

4.4.4 RO-ABWR-0004

4.4.4.1 BACKGROUND

591. RO-ABWR-0004 is entitled Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary. It was raised early in Step 2 of GDA within the Structural Integrity discipline, but acknowledges Reactor Chemistry as an associated area. Latterly, this RO, and some of the submissions made in association with the RO have been linked to RO-ABWR-0035. This is discussed later in this section of the Step 4 report.

592. The Reactor Pressure Vessel (RPV) for the UK ABWR will be made from a number of low alloy ferritic steel forgings and plate. These forgings will be subsequently welded together and clad to form the main pressure boundary of what will be a Very High Integrity (VHI) Component. It is my expectation that commensurate care will be needed to specify and control the manufacture of these forgings and the construction of this component.

593. For the forgings the detailed chemical specifications, forging processes, and quench and temper heat treatments are important in ensuring that they have tensile and toughness properties which exceed the minimum specified reasonably well throughout the thickness of the forging, and ensuring that through life properties are adequately maintained - for example that neutron irradiation embrittlement is minimised. Similarly careful control of welding and cladding material specifications and heat treatment processes will then be required to ensure the quality of the finished vessel.

594. Whilst the material specifications provided in the nuclear pressure vessel design codes will define the basis for these parameters, ONR's experience has shown that it is necessary for the Requesting Party to 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 vessel.

595. Examples of the types of evidence expected are:

i) Chemical Composition of Forgings:

- The detailed material composition for forgings, based on the design code specification along with any additional controls, and a justification of why these limits will achieve the necessary initial and through life properties, and deliver the Nuclear Safety Function throughout the entire volume.

- This justification should explain the basis for the limits and how these will affect aspects such as:

    - The microstructure and physical properties of the steel in the as-forged and heat-treated conditions;
    - The cleanliness of the steel, and other generic indicators of steel quality;
    - The susceptibility of the material to forging-related defects;
    - The development of the microstructure and properties following welding and/or cladding;

- o The susceptibility to radiation-embrittlement and other irradiation degradation mechanisms;
- o The susceptibility to thermal-embrittlement and other thermal degradation mechanisms.

This justification should consider setting compositional limits on:

- o Carbon, Manganese, Molybdenum, Nickel, Cobalt, Chromium, Copper, Vanadium, Aluminium, Titanium, Niobium (plus tantalum), Arsenic, Antimony, Boron, Tin, Calcium, Sulphur, Phosphorus, Silicon, Hydrogen and Nitrogen;
- o plus any other elements considered pertinent to control of properties.

ii) Casting and Forging Processes:

- The details of the casting and forging processes, an explanation of how these will be controlled and a justification of why they are adequate to deliver the Nuclear Safety Function throughout the entire volumes.

- This will potentially consider :

  - o Whether the steels will be fully-killed and confirmation of the process for de-oxidation;
  - o Ingot casting practice and how much material is discarded and from where (to remove segregation or inclusions);
  - o The degree of forging reduction achieved in manufacture of the various forged parts;
  - o The heat treatment sequences performed during manufacture;
  - o Mechanical property testing (tensile, Charpy, fracture toughness etc): application, property expectations and use of scatter in demonstrating adequacy of properties to deliver the Nuclear Safety Function throughout the entire volume.

iii) Welding and Cladding Processes:

- The detailed weld and clad material compositions and explanation of welding and cladding processes, including thermal controls, will be needed. There should be a justification of how these controls will achieve the necessary initial and through-life properties and degradation resistance, plus how these controls will lead to welds with suitable levels of initial integrity, both in terms of materials properties and defect sizes and distributions.

596. Evidence might typically include discussion of:

- Proposed welding processes and expected defect occurrence rates for selected processes;

- Compositional controls for the weld and cladding materials, including where these need to be controlled beyond code requirements;

- Thermal and parametric controls, and how these affect the final product through-life.

597. Hitachi-GE progressed with the production of a response within step 2 and 3 of GDA but, noting that there is a degree of evidence required in this RO, it was not appropriate to close out the RO until step 4. To provide guidance on ONR's expectations, I added three Actions to the RO, late in step 3. These were:

- **Action 1.1:** Regarding Materials compositions, Hitachi-GE should provide evidence that:
    - Relevant Good Practice regarding Materials compositions has been considered fully, notably from previous GDAs.
    - Irradiation Embrittlement surveillance programmes are adequate to mitigate the risk of irradiation embrittlement through-life.
- **Action: 1.2** Regarding Casting and Forging, Hitachi-GE should provide evidence that:.
    - Operational Experience (OPEX) from world nuclear plant, especially in the field of forging quality control, has been captured and measures implemented to prevent recurrences for the UK ABWR.
    - Evidence that the test specimens included in the materials qualification process will meet ONR's expectations in terms of being representative.
- **Action 1.3:** Regarding Welding and Cladding, Hitachi-GE should provide evidence that:.
    - How welding processes can be considered to be ALARP in terms of defect occurrence rates.

4.4.4.2 HITACHI-GE SAFETY SUBMISSIONS

598. Hitachi-GE's Safety Case. Hitachi-GE presented their responses to the RO through the following documents:

Action 1.1	Detailed Material Report for the RPV; Materials Testing Strategy Report for the RPV; Surveillance Program for Reactor Pressure Vessel in GDA.
Action 1.2	Detailed Material Report for RPV
Action 1.3	Manufacturing Report for RPV; Design Philosophy Report for RPV.

4.4.4.3 ONR ASSESSMENT.

599. In addressing the first part of Action 1.1, I have considered the detailed material reports, in the below, which have been produced as part of the RO-ABWR-0004. This information is also relevant to RO-ABWR-0035 and has been considered within that assessment as well.

Reactor Pressure Vessel Shell (Very High Integrity)

600. As part of my structural integrity assessment of the material selection, I assessed the detailed material justification for the Reactor Pressure Vessel (RPV) (Reference 154).

601. The detailed material report was submitted as part of the GDA assessments for steps 3 and 4. The reports were part of a suite of documents submitted, as the required evidence, for the closure of RO-ABWR-0004, but also linked to RO-ABWR-0035. Hitachi-GE provided information regarding the material composition and manufacturing techniques for the production of the Reactor Pressure Vessel. In my assessment, I identified that the submission does not provide sufficient evidence regarding the material composition and manufacturing techniques.

602. The choice of the RPV material as SA-508M Grade 3 Class 1 in itself is in line with my expectations as there is extensive operating experience using this material. My assessment focused on the additional Chemical Requirements for a UK ABWR RPV shell. These requirements were outlined as part of RO-ABWR-0004, where control of

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elements beyond the requirements of the ASME code should be in place. I raised RQ-ABWR-1194 to seek clarification of the material composition control for the RPV. Hitachi-GE stated that the compositional control of the RPV had been taken into account, and it ensured that the material exhibits the required toughness as well reducing the material's susceptibility to reheat cracking (Under Clad Cracking) and neutron irradiation damage. Hitachi-GE stated that the chemical control for the UKABWR will be in line with previous UK specific compositional experience (See Table A1-1 from Reference 154). I note that this is largely correct, but there are some exceptions: notably the maxima for Cr and Ni in the UK ABWR are higher than for UK practice, as defined by Hitachi-GE.

603. In the Detailed Material Report for the RPV, appendix 1 gives the basis for the chemical composition of the RPV. This relates directly to the first part of Action 1.1 from RO-ABWR-0004. Hitachi-GE state that, where their maxima are in excess of UK practice, the target values for composition will manage the chemical compositions to a lower level. I do not accept this argument. If lower values for any given element were required, then they should be specified and not targeted.

604. In specific reference to the value of nickel, the UK ABWR material has a set maximum 1.00% (w/w) compared with UK practice, which has been taken to be 0.85% (w/w) in previous GDAs. Previous designs to have undergone GDA (have utilised lower levels to improve weldability and mitigate irradiation embrittlement. These data are available on the ONR website. I note that Hitachi-GE will be controlling chemistry of the overall steel to control hydrogen cracking, reheat cracking and underclad cracking. I consider these to be the most significant threats to welding and fabrication of the RPV. By taking a holistic view of weldability rather than focus on certain individual elements, I consider it will be possible to maintain appropriate levels of weldability. I judge that the approach proposed is adequate, but that it will need monitoring during the licensing process. I have raised this further through an Assessment Finding in the following section. I further note that Hitachi-GE are monitoring the effect of chemical composition on irradiation embrittlement. This is discussed further below.

605. Hitachi-GE have also bench marked previous productions records of large forgings, demonstrating that the records of chemical compositions are lower than their target values (see Figure A1-3 from Reference 154). It can be seen that the target values for the trace elements composition are lower than their specified values.

606. Hitachi-GE have stated that their supplier for forgings, (Japan Steel Works, JSW), and the ladle degassing and stream degassing casting production process at JSW consistently produces ingot hydrogen contents below <0.8 ppm, previous experience of forging produced at JSW has shown no known reported incidence of hydrogen cracking in a JSW forging for nuclear applications. Overall, I judge that Hitachi-GE have provided enough evidence for the purpose of GDA and I am content with their processed to control the composition of the RPV material.

607. I am content that the chemical compositions proposed by Hitachi-GE are suitable for GDA, but note that control must be maintained of these limits throughout the licensing process. Consequently and in line with UK expectations, consistent with previous GDAs, and proportionate the highest levels of safety consequence for this component, I raise Assessment Findings later in this section.

608. Action 1.1 includes ONR's expectations for irradiation surveillance specimens. To address this matter I utilised the data available in the Surveillance Program for Reactor Pressure Vessel in GDA Report. During Level 4 communications with Hitachi-GE, this report was not highlighted to me as needed to close this RO Action.

609. I note that the surveillance specimens proposed are in excess of the minima specified within the ASME design code. I further note that, in general, the design of a BWR,

including that of the ABWR, leads to neutron fluences in the RPV material that are an order of magnitude lower than similar PWR designs. On this basis, I judge that the neutron irradiation embrittlement surveillance arrangements are adequate.

610. Regarding Action 1.2, I have sampled the detailed materials report for the RPV and note that there is a section specifically on the avoidance of hydrogen flakes. The control of Hydrogen as part of the manufacturing process for the UK ABWR was also part of my RQ-ABWR-1194. Hitachi-GE stated that appropriate steelmaking processes will be used in the casting to minimise the hydrogen levels in forgings. Hitachi-GE detailed the dehydrogenisation treatments during the steelmaking, and stated that during the manufacturing process, the hydrogen values for the UK ABWR would be measured. The levels of hydrogen for the forgings during product analysis will be kept to, generally, below the 0.8ppm level.

611. It was hydrogen flaking that was present at Doel 3 and Tihange 2 nuclear power stations in Belgium, and posed questions over the forging quality at those plants. I note that the production methodologies proposed have been selected so as to minimise the hydrogen levels and hence the likelihood of hydrogen flaking occurring. Moreover, following a visit to Japan Steelworks, performed in Step 3 (Reference 170), I have confidence that the steel supplier and their ability to deliver high quality steel. I accept that the 0.8ppm level is, generally, a suitable level to control to in order to minimise the risk of hydrogen flakes, but that this value lacks a scientific justification. I consider it proportionate, therefore, to address this as part of the assessment finding, below.

612. Hitachi-GE have not provided any in depth assessment of recent operational experience from Flamanville 3. I consider that this is not an issue for closure of GDA Step 4. This will best be addressed through the licensing stages and is discussed in the assessment finding below.

613. This notwithstanding, and in line with previous GDA assessments, I consider it incumbent upon any future licensee to demonstrate that the quality of components of the highest reliability are demonstrated individually. In this instance, GDA has provided suitable assurances that the component can be produced, but given that there can be no consequence case for failure of a VHI component, it is proportionate to raise the following Assessment Finding.

AF-ABWR-SI-18 – The RPV is a component of the highest reliability and as such all relevant international operational experience should be addressed when specifying it. During GDA, this area has developed and, hence, residual matters remain to be completed during licensing.

The licensee shall justify the proposals put forward by Hitachi-GE in terms of:

i) Materials properties

ii) Irradiation embrittlement

iii) Thermal embrittlement

iv) Dissolved gas control.

The licensee shall also demonstrate that these are achievable and can, in practice, be manufactured and meet the needs of the Safety Case throughout the thickness of the component.

614. The second part of Action 1.2 relates to the evidence needed that Hitachi-GE will produce materials testing results that are suitable and sufficient to support the Safety Case. Notably, in excess of the ASME code requirements, it is my expectation that

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fracture toughness testing is performed that supports the defect tolerance analyses performed as part of Hitachi-GE's response to the RO-ABWR-0001. The Materials Testing Strategy Report provides an outline of what additional testing is to be performed in excess of the code, as well as what materials archiving is expected from the licensee, also in excess of the code.

615. I note that, for this VHI component, Hitachi-GE have demonstrated that they have an understanding of the UK expectations for the production of data to support a Safety Case. The linkage to the requirements of the DTA is, however, not made explicitly.

616. As part of the licensing process, it is my expectation that any future licensee will complete the DTA of the areas having a VHI classification. As part of this analysis, it is my expectation that the linkage between the DTA and the materials testing be made explicit and a demonstration be made that the materials test results are capable of supporting the DTA. This has been discussed as part of ONR;s assessment of RO-ABWR-0001, above.

617. Action 1.3 relates to the choice of welding process and the demonstration that it can be shown to be ALARP, notably in terms of defect occurrence rates. The welding methodologies proposed are illustrated through the RPV manufacturing report. This report does not specify final production methods, but leaves it open to the producer to select from a number of different methods.

618. Of particular note, the main RPV shell welds have the option of being completed using Submerged Arc Welding (SAW). This process involves the use of a flux and, in my experience, can lead to slag inclusions and elevated rates of defects in the completed welds. Similarly for the weld overlay cladding of the bottom head dome, the option for using SAW has been maintained. It is not clear that a full comparison of the available methods has been performed and, given that a definitive selection of welding methodology has not been made, it is not apparent that the final welding methodology chosen will be ALARP.

619. The above notwithstanding, I judge that sufficient work has been performed within GDA to demonstrate that there are suitable welding methodologies available to manufacture the RPV. I, therefore, judge that Hitachi-GE have provided sufficient information to close Action 1.3 of RO-ABWR-0004. During licensing, and before the manufacture of the RPV commences, the final selection of welding process will be made. I therefore raise the following Assessment Finding:

AF-ABWR-SI-19 Hitachi-GE's GDA submissions have left the selection of fabrication processes for the Reactor Pressure Vessel to be selected during manufacture. Because welds, and the choice of fabrication technology affects plant risk, the licensee shall demonstrate that the fabrication processes selected for the reactor pressure vessel, including cladding processes, lower the plant risk so far as is reasonably practicable.

4.4.5 RO-ABWR-0034

620. RO-ABWR-0034 is entitled "Demonstrating the inclusion of a 'bottom drain line' in the UK ABWR design achieves inherent safety and reduces risks SFAIRP", and was raised by the Reactor Chemistry discipline. It does, however, contain strong links to the Structural Integrity discipline; hence it has been included in this report.

621. The UK ABWR includes a design feature referred to as the 'bottom drain line' (BDL). The line is located at the bottom of the RPV and is included in the Reactor Water Clean Up (RWCU), the 'bottom drain line' is highlighted in Figure .

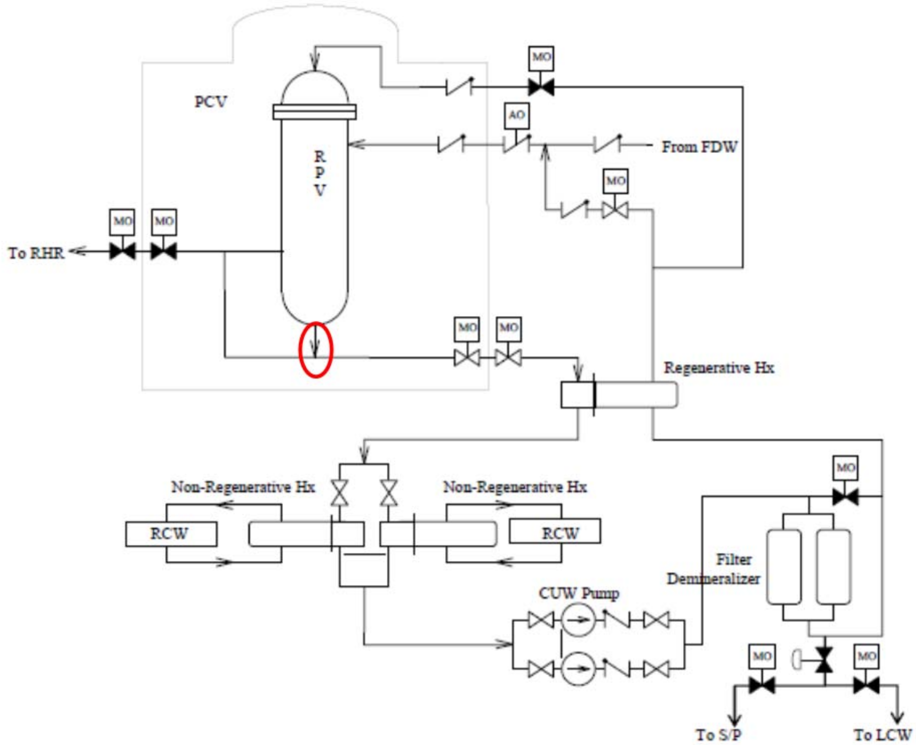

Figure 6 – Description of the RWCU for the UK ABWR

622. The Reactor Water Clean Up System is connected to the RPV via a bottom drain line and a mid-height connection. The 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.

623. During Step 2 of GDA of UK ABWR, RQ-ABWR-0082 was raised in the reactor chemistry area which requested Hitachi-GE to explain:

i) What the function(s) of the 'bottom drain line' in the UK ABWR design is(are), and based on their response to question i), to provide:

ii) A justification that the presence of a 'bottom drain line' in the UK ABWR design reduces risks SFAIRP.

624. The reason for this RQ was that ONR understood that the drain line is a significant contributor to operator dose during outages, and at that time Hitachi-GE were considering material changes to this line as a measure to mitigate the dose. The impact of the BDL on the dose to workers appeared consistent with other BWRs, a number of which have either permanently closed the BDL due to concerns regarding operator doses and/or materials degradation issues. Some have even purposefully designed out the BDL this includes internally pumped BWR designs similar to the UK ABWR. Furthermore, early in GDA Hitachi-GE were unable to clearly articulate the safety and/or operational functions and purpose(s) of the BDL. These were the contributory factors which led to RO-ABWR-0034 being raised.

625. Reference 155, "ALARP Consideration on RPV Bottom Drain Line" is Hitachi-GE's submission in response to RO-ABWR-0034. This is a detailed submission and provides an overall explanation of the approach taken for the ALARP demonstration as

well as a series of appendices which present Hitachi-GE's detailed evaluation of the different aspects which they have considered in their decision to include or exclude the BDL from the UK ABWR design.

626. Reference 155 identifies and explains the principal safety and operational functions the BDL performs. It also undertakes a thorough review of relevant worldwide OPEX to identify a credible list of alternative options to deliver these functions and identifies the relative benefits and detriments of the options. The report presents the outcome and rationale for scoring of these options that was undertaken during a workshop and a sensitivity analysis of the outcome. Having undertaken this work Hitachi-GE's report provides a series of conclusions. They conclude that for the UK ABWR design, there will be a BDL. In retaining the BDL, Hitachi-GE have also considered risk reduction measures to ensure that risks are reduced SFAIRP. They have changed the material selected for the bottom drain line in the UK ABWR, from carbon steel to low alloy steel. They have changed the UK ABWR operating chemistry from NWC (Normal Water Chemistry), which is utilised in Japanese deployments of the ABWR to HWC (Hydrogen Water Chemistry), OLNC (On-Line Noble Chemistry) and DZO (Depleted Zinc Oxide) injection. Hitachi-GE have stated that the implementation of remote inspection techniques will be done, to reduce the dose burden and to manage the conventional health and safety risks.

627. Although the BDL is actually part of the CUW (Clean-Up Water) System, this is why Hitachi-GE excluded the materials justification for it from the scope of Reference 156. I have considered the material selection for the BDL as part of my assessment for RO-ABWR-0034 in order to provide a complete picture of the BDL and the material choice.

628. I consider Hitachi-GE's approach, as presented in Reference 156, encompasses the major areas for consideration and I am satisfied that the process presented meets the relevant parts of ONR's guidance on demonstrating ALARP.

629. In Reference 156 Hitachi-GE states the BDL in the Japanese ABWR plant fulfils three principal functions and purposes, these are summarised below:

- Monitor the temperature in the bottom of the RPV, which has two main purposes
  - to support normal operations and plant control
  - to support fatigue monitoring.
- Remove CRUD from the RPV and monitor reactor water quality.
- Act as a point to be able to drain the RPV during construction, testing and maintenance activities.

630. Hitachi-GE stated that for the UK ABWR, they have identified the following principal functions that the BDL provides:

- Monitoring thermal stratification in the bottom of the RPV by measuring temperature.
- Monitoring reactor core coolant temperature.
- Monitoring and maintaining "primary coolant" water quality.
- Mitigating the risk of fuel failure by removing debris.
- Monitoring the water level in the RPV at a point below the fuel in severe accident conditions.

631. The two main detriments, or "risks", Hitachi-GE identify as a result of including a BDL in the UK ABWR design are:

- The impact on the risk of a LOCA. Including the BDL in the design requires an extra penetration at the very bottom of the RPV, a safety critical region.
- The impact on doses to workers.

632. Hitachi-GE argues the most important reasons (or "benefits") for including the BDL in the design are the purposes associated with the function of monitoring temperature in the bottom head region of the RPV – thermal stratification and plant control. From a structural integrity point of view, I agree with this conclusion. This does not prejudice the Reactor Chemistry assessment in this area.

633. Based on Hitachi-GE's identification of the main benefits and detriments associated with including a BDL in the UK ABWR design, my assessment of Reference 155 focussed on these areas.

634. As part of my structural integrity assessment I raised RQ-ABWR-1109 (Reference 157) asking for clarification of Hitachi-GE's claims regarding the significance of thermal stratification for the RPV and Reactor internal components.

635. Thermal stratification is a known phenomenon in Light Water Reactors and has been well researched and documented. It is the 'layering' of reactor water which is present in the RPV at different temperatures that can occur when the RIPs are unavailable. A number of RPV thermal stratification events have been reported for BWRs, thermal stratification is shown schematically below.

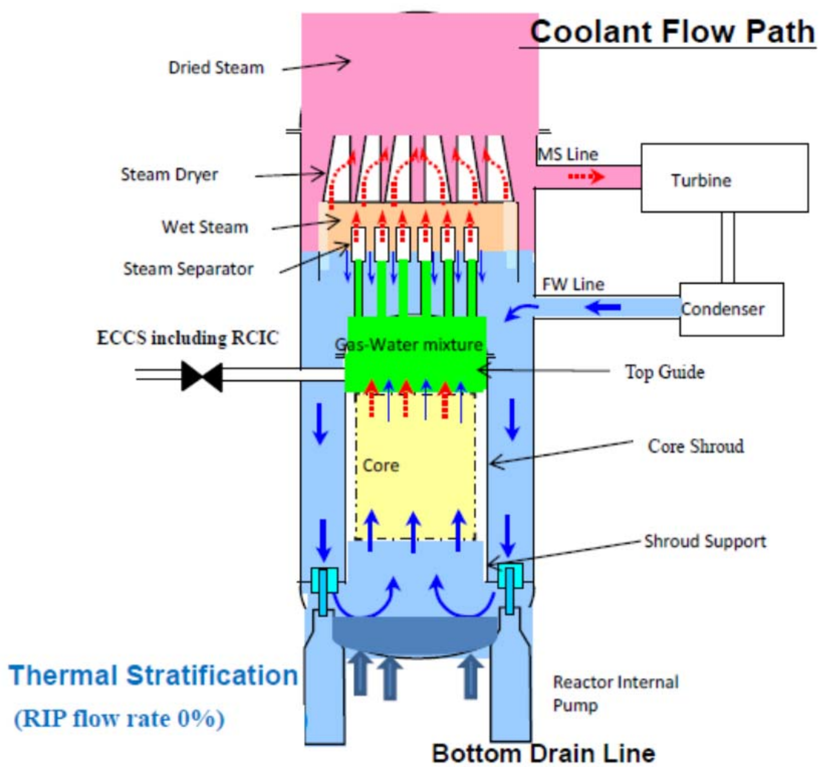

Figure 7 Schematic Representation of the Process of Thermal Stratification.

636. Thermal stratification of reactor coolant can lead to a thermal load in the system. In 2003, this was observed at Oskarsham Unit 3. The event occurred as cold water was introduced at the bottom of the RPV, whilst the RIPs (Reactor Internal Pumps) were unavailable. Due to the inadequate mixing due to natural circulation and following the RIPs re-start, there was a rapid increase in the temperature in the lower plenum of the RPV.

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637. Hitachi-GE's stated that the BDL needs to be part of the UKABWR design in order to monitor the temperature at the bottom of the RPV to mitigate the effects of thermal stratification. For UK ABWR this is done by monitoring the water temperature at the bottom of the RPV and the top of the water level and limiting the difference to 80°C.

638. Hitachi-GE stated that there are alternative means to provide measurements of the temperature at the bottom of the RPV and monitor and mitigate the risk of thermal stratification in the design. However, these options were discounted as Hitachi-GE stated that the reliability of the temperature measurements would not be adequate in order to appropriately control the temperature at the bottom of the RPV. Hitachi-GE state that all of the other available options don't give a temperature measurement as reliable and as quick as the one obtained via the BDL temperature measurement. In order to ascertain the suitability of this claim, I asked for further clarification on the reliability of the temperature measurements as part of RQ-ABWR-1109. Hitachi-GE stated that the function of the BDL, in avoidance of thermal stratification events for UK ABWR is twofold. Firstly, it will provide an impact in managing the effects, and secondly it will provide the lowest position for temperature monitoring at the bottom of the RPV. Hitachi-GE stated that due to natural circulation in the RPV, some of the cold water added to the RPV by the RCIC in a potential thermal stratification event is not fully mixed and would remain in the bottom of the RPV due to the density differences. I consider that Hitachi-GE's arguments of avoiding thermal stratification are appropriate, due to the structural integrity consequences of the thermal load. I note that, as part of the Structural Integrity investigation, the efficacy of different thermal measurement equipment has not been considered.

639. For comparison, Hitachi-GE considers the Fatigue Usage Factor (FUF) for the CRD stub tube, which is considered to be the worst case scenario, in terms of fatigue damage. The FUF for the CRD is calculated as [redacted] based on a peak stress of [redacted] and associated stress variations. Hitachi-GE considered a temperature gradient of [redacted] which will lead to a local increase in the peak stress equal to [redacted] which is equivalent to 1% elastic plastic peak strain and results in an FUF equal to [redacted] I consider that Hitachi-GE's consideration of the increase in the FUF together with the local increase in the peak stress for the limiting location of the CRD stub tube is an appropriate consideration of the consequences of thermal stratification. I also took into consideration the location of this potential damage mechanism and consider that Hitachi-GE's case to be appropriate for the avoidance of thermal stratification.

640. The Reactor Chemistry assessment considered the adequacy and accuracy of temperature measurement at the BDL location and raised RQ-ABWR-1347 (Reference 157). Overall their assessment concluded that based on the information provided, they are mostly satisfied that Hitachi-GE have demonstrated temperature measurements provided by the BDL can be considered as reliable. They highlighted that the one aspect they are not content with is Hitachi-GE's argument not to install a permanent flow meter in the BDL and have raised an assessment finding in this area. Further detail is present in the Reactor Chemistry Step 4 Assessment Report (Reference 159).

641. As part of my assessment I have also considered the material selection for the BDL. In the Japanese domestic ABWR design this pipework is constructed from Carbon Steel material which suffers from Flow Assisted Corrosion. As part of my structural integrity assessment I asked Hitachi-GE to consider material selection to avoid active degradation mechanisms such as FAC. In response Hitachi-GE stated that they propose to change the material selection to low alloy steel with Austenitic Stainless Steel once the pipework is outside of the vicinity of bottom of the reactor, where the concentration of pipework is high and inspection is difficult. From a structural integrity view point, the change in the material selection means that FAC as an active degradation mechanism in the region could be discounted. However, this does mean that there is a dissimilar metal weld (DMW) in the BDL pipework; however, Hitachi-GE

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stated that this DMW will be subject to appropriate inspections. Taking a holistic view of the system I consider this to be an appropriate position for GDA. RO-ABWR-0034 has now been closed.

4.4.6 RO-ABWR-35

4.4.6.1 BACKGROUND

642. The choice of materials for a particular System, Structure and Component (SSC) of a nuclear reactor is influenced by many competing factors, including:

- o the functional requirements of the SSC;
- o the tolerance/degradation of the SSC in its operating ‘environment’, and/or:
- o the potential hazards and risks, which must be either eliminated, reduced or controlled.

643. Considering the above factors, and potentially others, it is clear the justification of the most appropriate material selected for a particular SSC requires a balance to be struck. In reaching that balance, there should be a robust demonstration that all of the relevant risks have been considered and reduced SFAIRP.

644. Early in the GDA of the UK ABWR, Hitachi-GE provided some information related to material choices for the design. The information received in the submissions was relatively high-level principles, and was best described as design philosophy documentation.

645. As Step 3 of GDA progressed, Hitachi-GE planned to make a number of further, more detailed, submissions for particular SSCs, specifically to develop the structural integrity aspects of the safety case for UK ABWR.

646. Hitachi-GE’s plan was that these submissions would focus on those SSCs with the highest level of integrity claims. This approach meant that the justification of materials selected for UK ABWR would be focused on those SSCs where the consequences of failure were greatest. Wider consideration and justification of the materials selected for other SSCs, whose integrity claims may be lower, but the SSC still makes an important contribution to nuclear safety is, however, still necessary to meet ONR’s expectations.

647. Hitachi-GE’s approach at that time did not take full consideration of either:

i) the need to adequately justify materials selected for components below Very High Integrity (VHI);

ii) the need to balance the requirements of structural integrity, reactor chemistry and radiation protection. To be able to show that, on balance, the relevant risks have been reduced SFAIRP, the requirements of these, and potentially other disciplines; need to be factored in to the justification for the materials selected for UK ABWR.

648. ONR raised a Regulatory Observation (RO-ABWR-0035) to make clear the expectations regarding Hitachi-GE’s justification of the materials selected. This RO has been raised jointly by the reactor chemistry and structural integrity topics.

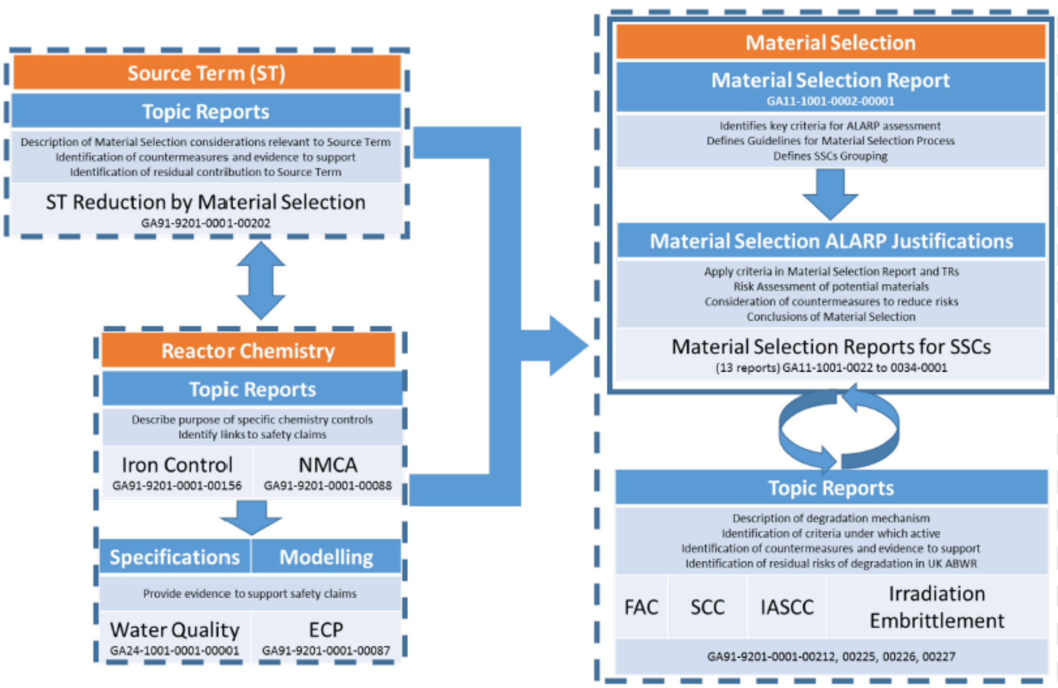
4.4.6.2 HITACHI-GE’S SAFETY SUBMISSION

649. The Structural Integrity PSR and Generic PCSR described a brief summary of the materials selected for the UK ABWR design and the degradation threats. It highlighted that Stress Corrosion Cracking (SCC), Irradiation Assisted SCC (IASCC) and Flow

Assisted Corrosion (FAC) are the three main threats and 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.

650. Hitachi-GE provided detailed demonstration of the UK ABWR material selection choices by reviewing the applicable degradation mechanisms and the specific material selection reports for specific Systems, Structures and Components (SSCs). The material choice for the UK ABWR has been subject to a Regulatory Observation (RO). The RO stated that the material choice should consider the functional requirement of the SSC, the through life degradation mechanisms which may impact the delivery of the safety function and the potential hazards and risks which must be addressed.

651. Hitachi-GE's format of an ALARP material selection Report, consisted of Material Selection Reports, highlighting the methodology for material selection and a suite of reports which addressed the components and systems classified as important to safety. Figure 1 details the material selection safety submission for the UK ABWR.



- **Figure 8 – Material selection for the UK ABWR.**

652. Overall the material selection reports for the systems and components of the UK ABWR consisted of the following:

System Category	Material Selection Report	System
Reactor Pressure Vessel Shell (VHI)	GA11-1001-0028-00001 1D-GD-0015	None
Main Steam Isolation Valve (VHI)	GA11-1001-0032-00001 1D-GD-0016	None

Main Steam Piping (VHI)	GA11-1001-0033-00001 1D-GD-0017	None
Nuclear Boiler Systems	GA11-1001-0031-00001 1D-GD-0018	• Reactor pressure vessel system • Nuclear boiler system • Reactor recirculation system
Control and Instrument Systems	GA11-1001-0030-00001 1D-GD-0019	• Control Rod Drive System (CRD) • Standby Liquid Control System (SLC)
Core Cooling Systems	GA11-1001-0029-00001 1D-GD-0020	• Residual Heat Removal system (RHR) • Flooding System of Specific Safety Facility (FLSS)
Reactor Servicing Equipment	GA11-1001-0034-00001 1D-GD-0021	• Fuel Storage Facility • Pool Miscellaneous Servicing
Reactor Auxiliary Systems	GA11-1001-0027-00001 1D-GD-0022	• Reactor water cleanup system (CUW) • Fuel Pool Cooling and Cleanup System (FPC) Suppression pool cleanup system (SPCU)
Radioactive Waste Systems	GA11-1001-0026-00001 1D-GD-0023	• Radioactive Drain transfer System (RD) • High Chemical Impurities Waste System (HCW)
Power Cycle Systems	GA11-1001-0025-00001 1D-GD-0024	• Condensate, Feedwater and Air Off Take System (C/FDW & AO)Feedwater Heater Drain and Vent System (HD/HV) • Condensate Filter Facility (CF) • Main Turbine • Moisture Separator Reheater (MSR) • Reactor Feedwater Pump • Turbine (RFPT) • Condenser • Off-gas System (OG)
Station Auxiliary Systems	GA11-1001-0024-00001 1D-GD-0025	• Makeup Water Condensate System (MUWC) • Reactor Building Cooling Water System (RCW) • HVAC Emergency

		Cooling Water System (HECW) • Reactor Building Service Water System (RSW) • Instrument Air System (IA)
Station Electrical Systems	GA11-1001-0023-00001 1D-GD-0026	• Emergency Diesel generator System (D/G)
Containment and Environmental Control Systems	GA11-1001-0022-00001 1D-GD-0027	• Primary Containment Vessel (PCV) • Standby Gas Treatment System (SGTS) • Atmospheric Control System (AC) • Filtered Containment Venting System (FCVS)

Table 7. Reports Submitted relating to Materials Selection

653. The submissions also contained the topic reports of the main degradation mechanisms, including Stress Corrosion Cracking (Report GA91-9201-0001-00225), Irradiation Induced Stress Corrosion Cracking (Report GA91-9201-0001-00226), flow Assisted Corrosion (Report GA91-9201-0001-00212).

4.4.6.3 ONR ASSESSMENT

654. As part of my assessment of the Material Selection for the UK ABWR, I have only considered aspects pertinent to the structural integrity of the pressure boundary. My assessment strategy has been to sample the components with the most significant structural integrity claims placed upon them. Nominally these are components which have been classified as Very High Integrity (VHI) or ASME class I components.

Regulatory Expectations

655. As part of my assessment, I have considered the structural integrity aspects of the safety submissions, I have taken cognisance of the applicable SAPs, including:

- o EAD.1 requires the safe working life of SSCs that are important to safety to be evaluated and defined at the design stage;
- o EAD.2 sets the requirement for adequate margins to exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on SSCs important to safety.

656. I have also taken cognisance of ONR's TAG, Integrity of Metal Components and Structures (Reference 163).

Assessment of the Degradation Topic Reports

Stress Corrosion Cracking

657. Austenitic stainless steels (SS) and nickel-base alloys (NBA) are widely used in high-temperature water systems, due to their resistance to general and localised corrosion.

However, under certain conditions they can be susceptible to Environmentally Assisted Cracking (EAC).

658. Austenitic stainless steels are widely employed in the construction of BWR reactor coolant system pressure boundary and as a structural material in reactor internals:

- o Pressure boundary applications include Piping, valve bodies, and pump casings.
- o Reactor internals applications include core plate, top plate, shroud support.

659. The main degradation mechanism of concern in BWR systems is Stress Corrosion Cracking (SCC). SCC can initiate and propagate through the material in either an intergranular or transgranular mode and, depending on the environment and material condition, the crack propagation mode can change.

660. For certain components in BWRs made of stainless steels or nickel-base alloys, such as the recirculation pipework, core internals, parts of the Reactor Pressure Vessel (RPV) such as the in-core monitor (ICM) housing, and the control rod drive (CRD) stub tubes, SCC has been the major degradation mechanism. However, in more modern designs such as the ABWR, some at risk components such as the recirculation pipework have been eliminated by using reactor internal pumps.

661. SCC requires a specific combination of susceptible material property, stress, and environment. The idea that a pre-existing defect is necessary for crack propagation is misleading, as cracking can start from as-machined or fabricated surfaces; which in-turn could be enhanced by the presence of cold work. This notwithstanding, the presence of laps and scratches on component surfaces are often implicated in SCC initiation.

662. I have reviewed the Topic Report on material degradation mechanisms – Stress Corrosion Cracking, (Reference 158) and consider it to provide a detailed description of the issue of Stress Corrosion Cracking and approaches being taken to mitigate against the threat of SCC. 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.

663. The report considers the three factors which influence the materials susceptibility to SCC and considers how they are each being managed for the UK ABWR. The three specifications considered are:

- Material Composition
- Stress Improvements and
- Environment

664. The detailed material choices and compositions used for the stainless steels and the nickel-base alloys considered for the UK ABWR have specified the use of nuclear grade low carbon stainless steels, 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. I consider that the use of nuclear grade material for the UK ABWR to be appropriate.

665. The management of residual stress arising from manufacturing processes and stress improvement techniques have also been proposed as part of the management of stress corrosion cracking in the UK ABWR. The management of stress and stress improvement techniques include;

- Limits on the bulk cold work,
- Limits on deformation and cold work due to manufacturing processes such as welding and bending, and
- Shot peening and surface modification to induce a compressive residual stress profile at the at-risk locations.

666. During my assessment of the report I asked Hitachi-GE a series of questions in Regulatory Query (RQ) 1416. Hitachi-GE's response to RQ-ABWR-1416 states that cold work is limited to [redacted] in Austenitic Stainless Steel components which come into direct contact with the reactor coolant. I followed this with another RQ, RQ-ABWR-1502 and sought further clarification as to how the impact of plant manufacturing and joining process impacts the control of cold work levels for the UK ABWR. Hitachi-GE responded that the cold work introduced by bending and forming is controlled with limits on the plastic strain or hardness. A limit of [redacted] bulk cold work (which is nominally equivalent to a hardness levels of [redacted]) will be applied for materials that contact reactor coolant. This limit applies to components which will not be Solution Heat Treated (SHT) prior to service entry.

667. Hitachi-GE responded that cold work imparted as a result of process such as welding and grinding (surface cold work) will not be controlled.

668. Overall, I consider that Hitachi-GE have provided a reasonable approach to the management of cold work for the UK ABWR for some of the processes which will be utilised in the manufacture of the plant components. However, there has been limited consideration of the impact of residual stress due to processes such as welding and the effectiveness of surface treatments in mitigating SCC over the 60 year plant life. I would expect a future licensee to give further consideration as the management of cold work, and the microstructural damage that implies, imparted onto the components which come into contact with reactor water. I consider this to be an assessment finding.

669. In their submissions Hitachi-GE's main concern has been with the deformation and SCC of Austenitic Stainless Steels and it is unclear if limits on levels of cold work also apply to components made from Nickel-Base Alloys. It is important to note that Nickel-Base Alloys, such as alloy 600 also suffer from SCC and limits of cold work are also applicable in this instance. I consider this to be an assessment finding.

AF-ABWR-SI-20 – Stress corrosion cracking susceptibility in austenitic stainless steel is linked to the levels of residual cold work, but this has not been considered by Hitachi-GE in sufficient detail during GDA. Therefore, the licensee shall justify the cold work levels for all Austenitic Stainless Steels and Nickel-Base Alloys as well as the methods utilised in the manufacture and construction of the UKABWR and apply appropriate remedial actions to reduce risks so far as is reasonably practicable.

670. Finally, and significantly, there is the water chemistry aspect which controls the environment. For the UK ABWR hydrogen water chemistry in conjunction with platinum injection (noble metal chemistry) and zinc injection will be utilised. 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 within the Reactor Chemistry GDA Step 4 Assessment Report (Reference 159), but its significance is noted here. I have also taken cognisance of the Reactor Chemistry RQ (RQ-ABWR-1324) considering the impact of Operating Chemistry on mitigating SCC.

671. Overall, I consider that the multi-faceted approach undertaken to mitigate the threat from the stress corrosion cracking degradation mechanism to be appropriate.

However, these steps do not all together eliminate the threat of SCC. Based on the proposed material selection, I consider a dedicated programme of In-Service Inspection constitutes relevant good practice for the management of degradation by SCC. I have not considered the proposed inspection regimes as part of my assessment, however, this has been considered as part of RO-ABWR-0001, above.

Irradiation Assisted Stress Corrosion Cracking

672. Hitachi-GE have highlighted Irradiation assisted stress corrosion cracking (IASCC) as an in-service degradation mechanism, which may affect some of the components in the UK ABWR.

673. IASCC is a phenomenon in which irradiation can affect the parameters which influence cracking, i.e. the environment, material, and stress. Irradiation can lead to an increase in the corrosion potential associated with radiolysis of water, which can lead to an increase in the rate of crack propagation. Radiation generated defects produce an increase in the yield strength. Irradiation can also influence the grain boundary chemistry by radiation-induced segregation (RIS). RIS arises from the flux of radiation-produced interstitial and substitutional solutes defects to sinks and is fundamentally different from thermal sensitisation. The influence of RIS on SCC susceptibility can be profound, as the process ultimately leads to a loss of chromium along the grain boundary.

674. Irradiation-induced grain boundary depletion of chromium is considered the main driving source of metallurgical change, which causes material susceptibility to IASCC. Therefore, in order to gain understanding of the microstructure and SCC susceptibility, material testing and high-resolution microscopy has been conducted on various materials. The effects of several metallurgical and electrochemical processes are under investigation. The variables considered include but are not limited to: neutron fluence, irradiation hardening, and irradiation induced chromium depletion, and water chemistry.

675. As part of my assessment I have considered the Topic Report on Material Degradation Mechanisms – Irradiation Assisted Stress Corrosion Cracking. The report provides a good overview of the degradation mechanism and provides a reasonable link to the issues impacting the UK ABWR. As part of my assessment I raised a number of queries in RQ-ABWR-1416. I asked Hitachi-GE to provide clarification on how the design of the UK ABWR has led to a reduced risk of IASCC. Hitachi-GE stated that the potential for IASCC in at risk locations, such as weld locations have been reduced by minimising welds in the highest neutron fluence areas (Reference 176). Furthermore, longitudinal welds have been eliminated from the core shroud region, as the use of cylindrical forgings has been proposed for the UK ABWR.

676. As part of my assessment I queried the control of alloying elements on radiation hardening and radiation induced segregation. In reference 176, Hitachi-GE stated that they have considered the role of alloying elements such as Nickel and Chromium as well as the small solute atoms such as Silicon, Phosphorus, Sulphur, Carbon and Oxygen. The report concludes that as Nuclear-Grade Low Carbon material will be utilised as part of the UK ABWR material selection, no further compositional considerations are needed.

677. I consider that Hitachi-GE's response is reasonable and designing out the susceptible weld locations in the UKAWBR design is a marked improvement.

Flow Assisted Corrosion and Erosion

678. The international experience in Light Water Reactors has identified flow assisted corrosion (FAC) as a potential degradation mechanism. FAC mainly affects pipework

and vessels, with numerous reported incidents, some of which have caused injury to personnel. I considered materials factors as the main influencing factor pertinent to my structural integrity assessment. FAC has affected carbon steel in both high energy and moderate energy systems**.**

679. FAC is the chemical process whereby the normally protective oxide layer on carbon or low alloy steel dissolves into a stream of flowing water (single phase) or a water/steam mixture (two phase). Normally, a protective oxide layer forms on the internal surface of the pipe, with the rate of formation and dissolution of the oxide being balanced.

680. However, in FAC an increase in localised mass transfer upsets the balance, leading to localised dissolution of the oxide and resultant metal loss. The flow velocity, pipe geometry, surface roughness, fluid density and steam quality can all play a significant role in FAC susceptibility. However, the rate of metal loss depends on a complex interaction between many parameters including water chemistry, temperature, material composition, and hydrodynamics. Single phase FAC most often occurs at locations where stable laminar flow is altered by pipe geometry such as bends, tee branches, valves and after orifices, which increase turbulence. Two phase FAC also most often occurs at similar locations. However, both types of FAC have been known to affect straight lengths of pipe. Two-phase FAC is generally considered to be more aggressive than single phase due to the more turbulent nature of wet steam.

681. Hitachi-GE's topic report stated that FAC affects carbon and low alloy steels, in cases where there are insufficient levels of Chromium (Cr) present. Hitachi-GE states that increasing Cr content in carbon steels above the minimum threshold of 0.04% provides an adequate level of protection against FAC.

682. In conducting my assessment of the Topic Report on FAC, I raised RQ 1416, seeking clarification of the Topic Report. I queried if Hitachi-GE have used the JSME rules for the management of FAC, and sought clarification as to how this compares with ASME. Hitachi-GE responded that no aspect of the UK ABWR design utilised JSME rules. The Reference to JSME only incorporates guidelines and data to inform the operating experience. Other sources of information such as EPRI have also been utilised for the UK ABWR.

683. Hitachi-GE's submissions attempted to justify the choice of increasing the Cr levels in Carbon steel (>0.10%Cr) as a means of having enough Cr to mitigate against FAC together with Active injection of Oxygen in the feedwater. This was also part of the query raised by the reactor chemistry assessment RQ-ABWR-1465. The outcome of this is treated in the Reactor Chemistry Assessment report (Reference 159). The data presented to support the reduction of FAC susceptibility ranged from many different sources, and although it came to describe some of the benefits which may be gained it does not eliminate the risk, compared with material such as Low Alloy Steel or Stainless Steels for components with high nuclear safety significance.

684. As part of my structural integrity assessment, I sampled the analysis of FAC degradation on the Class 1 sections of the feedwater pipework inside the RCCV, this was also part of my assessment of the Nuclear Boiler Systems (Reference 70). Hitachi-GE stated that, for the Class 1 feedwater pipework, the risk of FAC will be reduced by choosing Low Alloy steel as compared with carbon steel. Further considerations by Hitachi-GE concluded that the Low Alloy steel which has 1.25%Chromium should be selected for Class 1 pipework as part of the UK ABWR GDA. I consider this material choice to offer significant safety benefits as it will not require active mitigation by oxygen injection in the feedwater. I judge that this is a significant safety enhancement and is an acceptable position for the GDA assessment. I have also taken cognisance of the RQ 1479 raised by the Reactor Chemistry specialists as part of their assessment. This is detailed in Reference 159. Hitachi-GE stated that they have used the term FAC and Erosion-corrosion interchangeably and

they have taken cognisance of the risks arising from the mechanical impact (as applicable to corrosion-erosion) and FAC as part of their overall safety submission.

685. As part of my structural integrity assessment I have also considered the material selection for the Residual Heat Removal (RHR) System's main pipework. The RHR system's roles are to remove the decay heat during normal reactor shutdown and to cool the reactor core in the event of a Loss of Coolant Accident (LOCA) where the main condenser is not available. Hitachi-GE have considered the worldwide operating experience for RHR piping and has considered different material choice for the RHR pipework, including Austenitic Stainless Steels. Overall, the choice of material for the RHR piping will be Carbon Steel. I consider this to be appropriate as the introduction of Austenitic Stainless Steels increases the likelihood for Stress Corrosion Cracking in this system. Furthermore, risks arising from FAC can be adequately addressed in this area by taking into consideration the mitigations in place such as adequate inspection for FAC. Inspection for FAC is covered at a high level, but the licensee will need to develop adequate inspections for FAC as part of licensing through normal business. Overall, I am content with the material choice for the RHR pipework.

686. As part of my assessment of the material selection I have considered the Reactor Water Clean-up (CUW) System's main pipework as part of my structural integrity assessment. The CUW System's safety role is to clean the reactor coolant by removing impurities contained in the reactor water and maintain the quality of reactor water with the operating limits. I have focused my assessment in areas where active degradation mechanisms such as SCC and FAC have been identified as a risk mechanism. Hitachi-GE's approach to material selection for the CUW main pipe was to divide it into seven areas depending on the fluid condition, See Figure 3.6.1-1 and Table 3.6.1-1 from Reference 161. Hitachi-GE considered a range of options as part of their assessment process and will utilise a range of material options, including Austenitic Stainless Steel, Low Alloy Steel and Carbon Steel to construct the CUW system. It is unsurprising that Hitachi-GE's assessment has resulted in choosing a range of different options, as the chemistry in each section was a significant factor, overall, I am content with the material choice. I raised RQ 1416 and asked Hitachi-GE to consider the unintended consequences of their approach to material selection. This was also considered as part of the Reactor Chemistry Assessment in RQ 1332. Hitachi-GE explained that they have focused on areas where there are dissimilar metal welds (DMW) and have considered the weld quality, location, accessibility and potential degradation mechanism. They have also considered the risk of an accidental misuse of material. Overall, I am satisfied that for the purposes of a Generic Design, Hitachi-GE have given due consideration to material selection for this system and have adequately considered the unintended consequences of their material choice.

Material Choices for UK ABWR Systems

687. I have considered the detailed material reports, in the table presented above, which have been produced as part of the RO-ABWR-0004. However, the information is relevant for the material selection and as such has been utilised in my material selection report for RO-ABWR-0035:

Reactor Pressure Vessel Shell (Very High Integrity)

688. As part of my structural integrity assessment of the material selection, I assessed the detailed material justification for the Reactor Pressure Vessel (RPV) (Reference 154).

689. The detailed material report was submitted as part of the GDA assessments for steps 3 and 4. The reports were also part of a suite of documents submitted, as the required evidence, for the closure of RO-ABWR-0004. Hitachi-GE provided information regarding the material composition and manufacturing techniques for the production of the Reactor Pressure Vessel. In my assessment, I identified that the submission does

not provide sufficient evidence regarding the material composition and manufacturing techniques. I have not considered aspects of the cladding material choice.

690. The choice of the RPV material as SA-508M Grade 3 Class 1 in itself is in line with my expectations as there is extensive operating experience using this material. My assessment focused on the additional Chemical Requirements for a UK ABWR RPV shell. These requirements were outlined as part of RO-ABWR-0004, where control of elements beyond the requirements of the ASME code should be in place. I raised RQ 1194 to seek clarification of the material composition control for the RPV. Hitachi-GE stated that the compositional control of the RPV had been taken into account, and it ensured that the material exhibits the required toughness as well reducing the material's susceptibility to reheat cracking (Under Clad Cracking) and neutron irradiation damage. Hitachi-GE stated that the chemical control for the UKABWR will be in line with previous UK specific compositional limits (See Table A1-1 from Reference 154). Hitachi-GE have also bench marked previous productions records of large forgings, demonstrating that the records of chemical compositions are lower than their target values (see Figure A1-3 from Reference 154). It can be seen that the target values for the trace elements composition are lower than their specified values.

691. The control of Hydrogen as part of the manufacturing process for the UK ABWR was also part of my RQ 1194. Hitachi-GE stated that appropriate steelmaking processes will be used in the casting to minimise the hydrogen levels in forgings. Hitachi-GE detailed the dehydrogenisation treatments during the steelmaking, and stated that during the manufacturing process, the hydrogen values for the UK ABWR would be measured. The levels of hydrogen for the forgings during product analysis will be kept to, generally, below the 0.8ppm level. .

692. Hitachi-GE have stated that their supplier for forgings, (Japan Steel Works, JSW), and the ladle degassing and stream degassing casting production process at JSW consistently produces ingot hydrogen contents below <0.8 ppm, previous experience of forging produced at JSW has shown no known reported incidence of hydrogen cracking in a JSW forging for nuclear applications. Overall, I judge that Hitachi-GE have provided enough evidence for the purpose of GDA and I am content with their processes to control the composition of the RPV material. Note that this does not prejudice the discussions that have already been presented through RO-ABWR-0004.

Main Steam Piping (Very High Integrity)

693. I sampled the report titled detailed Material Report for MS Piping (Reference 175) as part of my assessment of the material selection for the UK ABWR. The report described the material specification including the requirement of chemical composition and mechanical properties for base metal and weld metal. As part of my assessment I raised RQ-ABWR-1193 and asked Hitachi-GE to explain the welding material choice as part of their submission. Hitachi-GE explained that the choice of the welding material was done according to the compatibility with the base material and proposed welding methodology. The piping material is made of Carbon Steel and as such Hitachi-GE proposes to use SFA – 5.18M and SFA – 5.1M for GTAW and SMAW welding. The welding material specifications have been chosen according to ASME code requirements, I consider that Hitachi-GE have taken cognisance of appropriate measures when choosing the appropriate welding material for the Main Steam Piping.

694. As part of this review I have sampled the VHI portions of the Main Steam Piping only. Sections of pipework at lower classifications have not been sampled in detail. The choice of product form for non-VHI portions of major pipework has been made implicitly through some sections of responses to RO-ABWR-0035. These choices will need to be justified as part of the licensing process.

AF-ABWR-SI-21 – Product form decisions for portions of safety significant pipework at classifications below Very High Integrity have been made implicitly within RO-ABWR-0035. Because it has not been demonstrated within GDA that these product forms minimises the overall plant risk, the licensee shall justify that the choice of product form for these items lowers the plant risk so far as is reasonably practicable.

Summary

695. Based on the outcome of my assessment of the "Material Choices and Degradation Mechanisms" I accept that the Hitachi-GE is taking a multi-faceted approach to mitigate the threat from different degradation mechanisms and consider that it is reasonable for Hitachi-GE to suggest a safety case based on a claim of minimising the likelihood of occurrence of material degradation, supplemented by an In-Service Inspection programme to detect any degradation before they become significant.

696. I have, in conjunction with Reactor Chemistry, closed RO-ABWR-0035 on the basis that the materials selection support the demonstration Structural Integrity for the ABWR. I note that the VHI components are treated further under RO-ABWR-0004.

4.4.7 RO-ABWR-0059

4.4.7.1 BACKGROUND

697. RO-ABWR-0059 was raised following discussions on the RCCV which have been pursued jointly between Structural Integrity and Civil Engineering, and ONR internal discussions with the severe accident discipline. RO-ABWR-0059 relates to the RCCV top head, which is an entirely metallic component of the containment boundary. ONR learned, through interactions with the Severe Accidents discipline, that the upper surface of the RCCV head will be flooded over during a severe accident in order to prevent the head flange from overheating and failing. This was an event that is believed to have happened during the Fukushima Daiichi accident.

698. The concern was raised by ONR that, if the Drywell head was contacted by cold water during a hot accident scenario, there could be a brittle failure. The drywell head is the largest metallic component in the RCCV and forms part of the containment. This would lead to an uncontained release and is, hence, unacceptable.

4.4.7.2 SCOPE

699. This issue has been treated under RO-ABWR-0059 and is separate to the discussions over the generic RCCV safety case. Interactions with the Civil Engineering discipline in this instance have been at a high level. Discussion in the section has been in the context of the level of demonstration needed for a low frequency event, such as is modelled in the severe accident discipline.

700. I have been utilising the general codes and standards outlined in Section 3. Specifically, I have used TAG 16 as my basis for considerations and have referred to the design code used by the requesting party. The design code utilised for this component has been the ASME boiler and pressure vessel code section NE. I note that, in general terms compliance with the code is required, but further note that demonstration that the component design is ALARP is also necessary.

701. In the RO, I set the following expectations:

1. Identify option to secure the Drywell Head containment function and hence afford additional protection to workers and the public under external cooling during a severe accident;

2. Evaluate the benefits and disbenefits associated with those options;

3. Provide a robust demonstration to justify whether or not it is reasonably practicable to implement one or more options to show that the risks relating to the structural integrity of the Drywell Head under external cooling during severe accident conditions are reduced SFAIRP (So Far As Is Reasonably Practicable).

4.4.7.3 HITACHI-GE SAFETY CASE

702. Hitachi-GE presented a single document in response to this RO, which was (Reference 162). This presented the options which Hitachi-GE considered applicable to answering the RO, these included:

1. Options to engineer an overflow system to prevent areas that had not been heat-treated on the drywell head from coming into contact with cooling water.

2. The option to perform a complete heat-treatment of the drywell head so as to manage the risk of a brittle failure to as low as possible.

703. There was also a significant amount of work presented which related to the original work presented within the Severe Accident discipline. I did not assess work related to how to mitigate the size of any potential release, nor the details of how any accident sequence would progress as these are areas that have been dealt with by the Severe Accidents discipline. This is covered through RO-ABWR-46.

704. Hitachi-GE presented the safety arguments as a complex interaction between the requirements of the Structural Integrity of the component and those of the Severe Accident under which the upper drywell head flooding would occur. A logical exposition of the requirements of the RO was not presented.

705. The range of engineering solutions for the delivery of cooling water, and how these might avoid contact of water with areas of the drywell head that are not heat-treated were presented, to some degree. The possible changes to civil structures that would be needed to facilitate water delivery to the exact needed level were discussed, albeit at a high level.

706. The costs and benefits of full drywell head heat treatment were also presented. This was presented to be a small overall cost and improved flexibility in how the water might be delivered, in addition to the benefit of no changes to the civil structures.

707. Hitachi-GE came to the conclusion that it is the best option to heat-treat the full head to address the intent of RO-ABWR-0059.

708. I note that Hitachi-GE repeat the arguments made under RO-ABWR-0023 Action 4.7.2, which was raised in the Severe Accidents area, that the gasket material for the drywell head has been changed from a silicone rubber in the reference design, to an EPDM (Ethylene Propylene Diene Monomer) rubber.

4.4.7.4 ONR ASSESSMENT

709. The safety case presented does not explicitly address the expectations set out within the RO. The response overall shows, in my opinion, an incomplete grasp of the Regulatory Observation process and was difficult to assess from a technical point of view. The level of discussion repeated from the previous Severe Accident RO, which dealt with the concerns outside the structural integrity discipline, was high. This obscured the responses to the RO presented. I will not comment on these matters in this report.

710. The different possible civil engineering changes that would be required were discussed at a high level, but the total impact on the design was difficult to assess. I could assess, however, that there would need to be some changes and these may be significant. I could not ascertain, with any certainty, how effective these would be in avoiding contact of cold water with the areas of the drywell head that have not been heat treated.

711. Hitachi-GE did not show a good understanding of the hierarchy of controls for risks.

712. The possibility of heat treating the entire drywell head was discussed at length. I agree that this would be effective in mitigating the risk of contact of cold water with areas that had not been heat treated, and hence having an unacceptably low fracture toughness, because there would be no such areas. I consider that this is a powerful argument. I further agree that full heat treatment of the head would help to facilitate water delivery by any means possible, as splashing would not be considered a problem.

713. As stated above, Hitachi-GE concluded that the ALARP position for the drywell head is to heat treat the entire head and not perform any changes to the civil structures. I judge that, although the presentation of the safety arguments for the other options is poor, and the level of extraneous discussion made by Hitachi-GE further obscures the safety arguments, the case that full heat-treatment of the head is ALARP is compelling.

714. The additional information on the change from a silicone rubber gasket to an EPDM rubber was reviewed and I concur that this material should offer an improvement in temperature resistance. However, this is not material to the RO presented and so has not informed closure.

4.4.7.5 CONCLUSION

715. I was content to close RO-ABWR-0059 on the basis that, although Hitachi-GE's presentation of the safety arguments was poor, the result was demonstrably ALARP. This was because the risk of brittle fracture would be greatly reduced by ensuring that all regions of the drywell head have adequate fracture toughness.

4.5 Comparison with standards, guidance and relevant good practice

716. The goal of the GDA Step 4 assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety, security and environmental case and for evidence to support that case to be presented and assessed. For this purpose, within ONR, assessment is undertaken in line with the requirements of the How2 Business Management System (BMS) document PI/FWD (Reference 164). Appendix 1 of Reference 164 sets down the process of assessment within ONR; Appendix 2 explains the process associated with sampling of safety case documentation.

717. In addition, the Safety Assessment Principles (SAPs) (Reference 32) 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 4 assessment of the UK ABWR. The SAPs 2014 Edition (Revision 0, November 2014) have been used in this assessment. The 2004 edition of the ONR SAPs, which are extensively similar in so far as they relate to Structural Integrity matters, have been benchmarked against IAEA standards.

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

719. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and enlarged on in the Technical Assessment Guide on Structural Integrity (Reference 163). This guide provides the principal means for assessing the Structural Integrity aspects in practice.

720. The overall presented case, as I have assessed it gives me no reason to believe that it will not meet the intent of the codes and standards listed above, and is in line with UK relevant good practice.

4.5.1 Safety Assessment Principles (SAPs)

721. The SAPs (Reference 32) 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.

4.5.2 Technical Assessment Guides (TAGs)

722. The following Technical Assessment Guide has been used as part of this assessment:

- o NS-TST-GD-016 Revision 4. March 2013. Integrity of Metal Components and Structures.
- o NS-TST-GD-016 Revision 5. March 2017. Integrity of Metal Components and Structures.
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4.6 Overseas regulatory interface

723. ONR has formal information exchange agreements with a number of international nuclear safety regulators, and collaborates through the work of the International Atomic Energy Agency (IAEA) and the Organisation for Economic Co-operation and Development Nuclear Energy Agency (OECD-NEA). This enables us to utilise overseas regulatory assessments of reactor technologies, where they are relevant to the UK. It also enables the sharing of regulatory assessment findings, which can expedite assessment and helps promote consistency.

724. ONR also represents the UK on the Multinational Design Evaluation Programme (MDEP).This seeks to:

- ■ Enhance multilateral co-operation within existing regulatory frameworks
- ■ Encourage multinational convergence of codes, standards and safety goals
- ■ Implication of MDEP products in order to facilitate the licensing of new reactors, including those being developed by Gen IV international Forum

725. In this assessment, the following information from overseas regulators has been used:

726. Interactions with the Belgian Regulator (FANC), and other international regulators from Japan, France, the USA, and Spain has been used to assess the quality of forged materials in nuclear power plant. This has arisen out of the findings of lamellar defects present at the Doel 3 and Tihange 2 power stations in Belgium, and its link back to hydrogen flaking which is believed to have occurred during the production of the forgings. These exchanges were progressed via regulatory working group discussions.

4.7 GDA Issues

727. During my assessment there were no residual matters identified as GDA Issues.

4.8 Assessment findings

728. During my assessment 21 residual matters were identified for a future licensee to take forward in their site-specific safety submissions. Details of these are contained in Annex 5.

729. These matters do not undermine the generic safety submission and are primarily concerned with the provision of site specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages. These items are captured as assessment findings.

730. I have recorded residual matters as assessment findings if one or more of the following apply:

- site specific information is required to resolve this matter;
- resolving this matter depends on licensee design choices;
- the matter raised is related to operator specific features / aspects / choices;
- the resolution of this matter requires licensee choices on organisational matters;
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

731. Assessment Findings are residual matters that must be addressed by the Prospective Licensee and the progress of this will be monitored by the regulator.

4.9 Minor shortfalls

732. During my assessment 3 items residual matters were identified as minor shortfalls in the safety case, but which are not considered serious enough to require specific action to be taken by the future licensee. Details of these are contained in Annex 6.

733. Residual matters are recorded as a minor shortfall if it does not:

- undermine ONR’s confidence in the safety of the generic design;
- impair ONR’s ability to understand the risks associated with the generic design;
- require design modifications;
- require further substantiation to be undertaken.

5 CONCLUSIONS

734. This report presents the findings of my Step 4 Structural Integrity assessment of the Hitachi-GE UK ABWR.

735. Hitachi-GE have provided a satisfactory safety demonstration for the UK ABWR by covering a representative series of Structural Integrity components. The rationale for the choice of these components has been presented and I am content with it. The areas covered by Hitachi-GE's safety submissions, and this assessment, have included the main pressure boundary of the reactor coolant circuit, including pressure boundary items in the turbine system, the Reactor internals, the metallic parts of the Primary Containment Vessel and the pressure boundary of the interim spent fuel storage.

736. I am content with the structure of the component Structural Integrity classification process applied by Hitachi-GE and have applied a proportionate approach to my assessment, focussing effort where the nuclear safety risk is greatest. I have raised assessment findings where there are shortfalls against my expectations within the scope of GDA.

737. I have reviewed the sections of the PCSR relevant to Structural Integrity and I am content that they are adequate for GDA purposes, when considered along with their supporting documents.

738. A safety case demonstration has been provided by Hitachi-GE for each of the representative components covering the scope of the Structural Integrity components.

739. To conclude, I am broadly satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for Structural Integrity. I consider that from a Structural Integrity view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits beings secured .

740. Several assessment findings (Annex 5) were identified; these are for future licensee to consider and take forward in their site-specific safety submissions. These matters do not undermine the generic safety submission and require licensee input/decision.

5.1 Key Findings from the Step 4 Assessment

741. I consider that from a Structural Integrity view point, the UK ABWR design is suitable for construction in the UK, at this present time, future permissions and permits beings secured. For this reasoning the UK ABWR should be awarded a DAC at this present time.

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149 UK ABWR GDA - 200797-0000-D20-RPT-0054 - QGE-GD-0022 - Rev 1 - IVC Review of End of Manufacture Technical Justifications MS Seamless Pipe to MSIV Weld TJ - 05 January 2017. (TRIM 2017/339150).

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156 HGNE COMMERCIAL - UK ABWR - GA91-9201-0003-00523 - SE-GD-0241 - Rev 2 - UK ABWR GDA ALARP Consideration on RPV Bottom Drain Line - 08 August 2016 (TRIM 2016/315197)

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174 HGNE COMMERCIAL - UK ABWR - GA91-9201-0003-02258 - 1E-GD-5103 - Rev 0 - Follow up on Review Comments on RQ-ABWR-1416 (Response to RQ-ABWR-1502) - 04 August 2017 (TRIM 2017/301397)

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Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
EMC.1	Integrity of metal components and structures: highest reliability components and structures. Safety case and assessment	The safety case should be especially robust and the corresponding assessment suitably demanding, in order that an engineering judgement can be made for two key requirements: a) the metal component or structure should be as defect-free as possible; b) the metal component or structure should be tolerant of defects.
EMC.2	Integrity of metal components and structures: highest reliability components and structures. Use of scientific and technical issues	The safety case and its assessment should include a comprehensive examination of relevant scientific and technical issues, taking account of precedent when available.
EMC.3	Integrity of metal components and structures: highest reliability components and structures: Evidence	Evidence should be provided to demonstrate that the necessary level of integrity has been achieved for the most demanding situations.
EMC.4	Integrity of metal components and structures: general. Procedural control	Design, manufacture and installation activities should be subject to procedural control.
EMC.5	Integrity of metal components and structures: general. Defects	It should be demonstrated that safety-related components and structures are both free from significant defects and are tolerant of defects.
EMC.6	Integrity of metal components and structures: general. Defects	During manufacture and throughout the operational life the existence of defects of concern should be able to be established by appropriate means.
EMC.7	Integrity of metal components and structures: design. Loadings	For safety-related components and structures, the schedule of design loadings (including combinations of loadings), together with conservative estimates of their frequency of occurrence should be used as the basis for design against normal operating, plant transient, testing, fault and internal or external hazard conditions.
EMC.8	Integrity of metal components and structures: design. Requirements for examination	Geometry and access arrangements should have regard to the requirements for examination.
EMC.9	Integrity of metal components and structures: design. Product form	The choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component.
EMC.10	Integrity of metal components and structures: design. Weld positions	The positioning of welds should have regard to high-stress locations and adverse environments.

EMC.11	Integrity of metal components and structures: design. Failure modes	Failure modes should be gradual and predictable.
EMC.12	Integrity of metal components and structures: design. Brittle behaviour	Designs in which components of a metal pressure boundary could exhibit brittle behaviour should be avoided.
EMC.13	Integrity of metal components and structures: manufacture and installation. Materials.	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility.
EMC.17	Integrity of metal components and structures: manufacture and installation. Examination during manufacture	Provision should be made for examination during manufacture and installation to demonstrate the required standard of workmanship has been achieved.
EMC.21	Integrity of metal components and structures: operation. Safe operating envelope	Throughout their operating life, safety-related components and structures should be operated and controlled within defined limits consistent with the safe operating envelope defined in the safety case.
EMC.23	Integrity of metal components and structures: operation. Ductile behaviour	For metal pressure vessels and circuits, particularly ferritic steel items, the operating regime should ensure that they display ductile behaviour when significantly stressed.
EMC.24	Integrity of metal components and structures: monitoring. Operation	Facility operations should be monitored and recorded to demonstrate compliance with the operating limits and to allow review against the safe operating envelope defined in the safety case.
EMC.27	Integrity of metal components and structures: pre- and in-service examination and testing. Examination	Provision should be made for examination that is reliably capable of demonstrating that the component or structure is manufactured to the required standard and is fit for purpose at all times during service.
EMC.28	Integrity of metal components and structures: pre- and in-service examination and testing. Margins	An adequate margin should exist between the nature of defects of concern and the capability of the examination to detect and characterise a defect.
EMC.29	Integrity of metal components and structures: pre- and in-service examination and testing. Redundancy and diversity	Examination of components and structures should be sufficiently redundant and diverse.
EMC.30	Integrity of metal components and structures: pre- and in-service examination and testing. Control	Personnel, equipment and procedures should be qualified to an extent consistent with the overall safety case and the contribution of examination to the Structural Integrity aspect of the safety case.

EMC.32	Integrity of metal components and structures: analysis. Stress analysis	Stress analysis (including when displacements are the limiting parameter) should be carried out as necessary to support substantiation of the design and should demonstrate the component has an adequate life, taking into account time-dependent degradation processes.
EMC.33	Integrity of metal components and structures: analysis. Use of data	The data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and the contribution to the safety case.
EMC.34	Integrity of metal components and structures: analysis. Defect sizes	Where high reliability is required for components and structures and where otherwise appropriate, the sizes of crack-like defects of structural concern should be calculated using verified and validated fracture mechanics methods with verified application.
EAD.1	Ageing and degradation. Safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD.2	Ageing and degradation. Lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components that are important to safety.
EAD.3	Ageing and degradation. Periodic measurement of material properties	Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties.
EAD.4	Ageing and degradation. Periodic measurement of parameters	Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.
ECS.1	Safety classification and standards. Safety categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety.
ECS.2	Safety classification and standards. Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regard to safety.

ECS.3	Safety classification and standards. Standards	Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.

Annex 2

Technical Assessment Guide

TAG Ref	TAG Title
NS-TAST-GD-005 Revision 7	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TST-GD-016 Revision 4. March 2013 NS-TAST-GD-016 Revision 5 March 2017	Integrity of Metal Components and Structures. Integrity of Metal Components and Structures.

Annex 3

National and International Standards and Guidance

National and International Standards and Guidance
Safety of Nuclear Power Plants: Design. Safety Requirements. International Atomic Energy Agency (IAEA). Safety Standards Series No. NS-R-1. IAEA. Vienna. 2000. www.iaea.org.
American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. With particular reference to: • ASME BPVC Section III - Rules for Construction of Nuclear Facility Components • ASME BPVC Section IX - Welding and Brazing Qualifications • ASME BPVC Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components
European Network for Inspection and Qualification: " European methodology for qualification of non-destructive testing third issue" https://publications.europa.eu/en/publication-detail/-/publication/34dc46ec-6c85-4e49-a672-7b382714457a

Annex 4

Regulatory Issues / Observations

RI / RO Ref	RI / RO Title	Description	Date Closed	Report Section Reference
RO-ABWR-0001	Insert Avoidance of Fracture – Margins based on the size of Crack-Like Defects	This RO underpins the demonstration of Fracture Avoidance for the UK ABWR. This is an expectation for components of the highest reliability within the UK, but is not at expectation for all countries with nuclear power plant. ONR raised this RO so as to ensure that Hitachi-GE understood the UK expectations in this area. Specific actions were placed on Hitachi-GE to provide claims arguments and evidence supporting the avoidance of fracture Safety Case in the following areas: • Material Properties, • Fracture Assessment, • Manufacturing Inspection, • Overall of Avoidance of Fracture Demonstration.		
RO-ABWR-0002	CRD Penetration Design	The pressure boundary welds associated with the CRD penetration may not be classified as VHIC welds (still work in progress), however, due to the complexity of the design it will be necessary to demonstrate the initial and through life integrity of this pressure boundary even as Standard Class 1 welds. The demonstration of initial and through life integrity will need to address a number of related aspects including: • Loading mechanisms • Design code compliance • Inspection • Materials choices • Operational experience		
RO-ABWR-0003	RPV Design	The main pressure boundary and support structure of the Reactor Pressure Vessel (RPV) will be constructed from a number of major component parts. There is a need to ensure that this pressure boundary and support structure has the highest level of integrity as the vessel will be classified as a Very High Integrity (VHI) Component.	14 June 2017	

		In order to satisfy this ONR's expectations, 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. Evidence will be required to show that: • Number and length of welds are minimised; • Nozzle design promotes Structural Integrity; • RPV head design is demonstrably ALARP.		
RO-ABWR-0004	Material/Forging/Weld/Clad Specifications for RPV Pressure Boundary	The Reactor Pressure Vessel (RPV) will be made from a number of low alloy ferritic steel forgings. These forgings will be subsequently welded together and clad to form the main pressure boundary of what will be a Very High Integrity (VHI) Component. Commensurate care will be needed to specify and control the manufacture of these forgings and the construction of this component. For the forgings the detailed chemical specifications, forging processes, and quench and temper heat treatments are important in ensuring that they have tensile and toughness properties which exceed the minimum specified reasonably well throughout the thickness of the forging, and ensuring that through life properties are adequately maintained - for example that neutron irradiation embrittlement is minimised. Similarly careful control of welding and cladding material specifications and heat treatment processes will then be required to ensure the quality of the finished vessel. Whilst the material specifications provided in the nuclear pressure vessel design codes will define the basis for these parameters, ONR's experience has shown that it is necessary for the Requesting Party to 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 vessel. Examples of the types of evidence expected are: 1. Chemical Composition of Forgings 2. Casting and Forging Processes 3. Welding and Cladding Processes:		
RO-ABWR-0035	Robust Justification for the Materials selected for UK ABWR	The choice of materials for a particular Structure, System or Component (SSC) of a nuclear reactor is influenced by many competing factors. Therefore, the justification of the most appropriate material selected for a particular SSC requires a balance to be struck.		

		In reaching that balance, there should be a robust demonstration that all of the relevant risks have been considered and reduced So Far as is Reasonably Practicable (SFAIRP). During the Generic Design Assessment (GDA) of the United Kingdom Advanced Boiling Water Reactor (UK ABWR), to date, Hitachi-GE have provided some information related to material choices for the design. The submissions received have been relatively high-level principles, best described as design philosophy documentation. As Step 3 of GDA progresses Hitachi-GE plan to make a number of further, more detailed, submissions for particular SSCs, specifically to develop the structural integrity aspects of the safety case for UK ABWR. This approach means the justification of materials selected for UK ABWR is focused on those SSCs where the consequences of failure are greatest. Wider consideration and justification of the materials selected for other SSCs, whose integrity claims may be lower, but the SSC still makes an important contribution to nuclear safety is, however, still necessary to meet ONR's expectations. This Regulatory Observation (RO) was, therefore, raised to make clear ONR's expectations regarding Hitachi-GE's justification of the materials selected for UK ABWR		
RO-ABWR-0059	Provision of Water Cooling for the RCCV Drywell Head of the UK ABWR	The drywell head forms part of the RCCV (Reinforced Concrete Containment Vessel) containment which, in an event such as a LOCA (Loss of Coolant Accident), will form the barrier against the release of radioactivity to the environment. In a severe accident scenario, Hitachi-GE plan to flood the cavity above the drywell head, to keep the flange of the head cool. Hitachi-GE have presented data and analysis aimed at demonstrating the structural integrity, and hence achievement of containment function, of the DW head under the Pressurised Thermal Shock (PTS) transient imposed during external cooling. The current drywell head design does not include Post Weld Heat Treatment (PWHT) of the structural welds above the flange region. Consequentially, under the combination of loadings arising from the internal pressure, the PTS transient and non-stress relieved welds the margin between the imposed SIF and the fracture toughness is likely to be significantly reduced. There would be increased confidence in the structural integrity of the Dry Well (DW) head under severe accident conditions if the overall		

		imposed SIF from the combination of the residual and thermal stress is reduced. Hitachi-GE's current approach does not give full consideration to the potential options to achieve this and whether or not it is reasonably practicable to implement one or more of those options.		

Annex 5

Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-SI-01	The safety functions of all classes of component are not defined and collated coherently in the GDA submissions. To properly explain the safety functions of all classes of component, the licensee shall link the safety demonstration of components of lower Structural Integrity classification to the safety functions they perform explicitly.	4.2.1
AF-ABWR-SI-02	In the safety documentation presented during GDA, the design code version is not controlled. The design code for the reactor plant forms the basis for good practice in design. The licensee shall institute controls over the edition of, and addenda to, the design code to be used.	4.2.1
AF-ABWR-SI-03	During GDA, Hitachi-GE have not demonstrated that the secondary consequences of failure of lower classification plant, in terms of missile formation, is captured properly during the classification process. The licensee shall review the classification of all Structural Integrity components inside containment relating to the potential formation of missiles and their effect on adjacent systems structure and components.	4.2.2
AF-ABWR-SI-04	Late on in GDA, ONR became aware that the bounding consequences of failure, for certain components, may not be during power operation but during shutdown conditions. Because Hitachi-GE's Structural integrity classification process assumes that operation provides the bounding condition, the licensee shall review the structural integrity classification for all systems structures and components and confirm if the classification is bounding.	4.2.2

AF-ABWR-SI-05	During GDA, Hitachi-GE have not demonstrated that the proposed inspection of reactor internals will meet the needs of the Safety Case. The licensee shall demonstrate that the inspection capability of the reactor internals matches the safety case requirements with regard to Stress Corrosion Cracking and Irradiation Enhanced Stress Corrosion Cracking. This must consider a demonstration of the capability of the in-service inspection taking into account the essential parameters of the defects and the inspection conditions.	4.2.8
AF-ABWR-SI-06	Because the fatigue usage calculation methodology used by Hitachi-GE is not demonstrably conservative for all cases, the licensee shall appropriately verify the fatigue usage factor calculations by other methodologies where margins to failure are low.	4.2.12
AF-ABWR-SI-07	Because some example toughness values produced by Hitachi-GE during GDA are below the values used within the Defect tolerance analysis, the assumptions used within the analyses must be justified within licensing. The Licensee shall, therefore, underpin the lower-bound upper-shelf materials toughness value for all Very High Integrity components and materials using representative testing, or justify the use of a lower value in their Defect Tolerance Analyses. More generally, materials data, across all classifications of component underpins the Structural Integrity safety cases. Confidence is needed, therefore, across a wide range of materials data, for all classes of component. The Licensee shall produce a comprehensive and appropriately justified material data set for the VHI, Class 1 and Class 2 components for use during the design and assessment process and also to support through life operation. This will need to cover all relevant data including the basic design data and the confirmatory batch and weld specific test data from the additional fracture toughness testing programme. It will need to be clearly presented such that the initial pedigree of the data can be traced following the literature trail with comparison to other international data sets where possible.	4.4.1.3

AF-ABWR-SI-08	Software that is the intellectual property of a third party has been used to perform defect tolerance analyses. Confidence in this software and the methodologies it embodies is critical to the safety case; because of this, the licensee shall ensure that robust verification and validation arrangements, incorporating adequate independent review and methods, are developed and implemented to underpin the fracture assessments for very high integrity and high integrity components.	4.4.1.3
AF-ABWR-SI-09	The fatigue damage calculation methodology, used by Hitachi-GE, is not demonstrably conservative. Because of this, the licensee shall demonstrate that either the Fatigue Crack Growth cycle counting approach is demonstrably conservative, or that sufficient conservatisms exist elsewhere in the Fatigue Crack Growth and Defect Tolerance Analysis calculations to compensate for any under-estimation of Fatigue Crack Growth.	4.4.1.3
AF-ABWR-SI-10	Hitachi-GE have provided example locations for defect tolerance analysis within GDA but not yet justified the whole plant sufficiently. Therefore, the Licensee shall undertake Defect Tolerance Analysis on a wider range of weld locations on the Very High Integrity components in order to demonstrate that the limiting locations have been assessed. The Licensee shall also undertake Defect Tolerance Analysis on the vulnerable areas of the parent forgings in order to demonstrate that the limiting locations have been assessed.	4.4.1.3
AF-ABWR-SI-11	The Reactor Pressure Vessel closure studs are standard ASME class 1 components when considered individually, but a component of the highest reliability, when considering multiple (common-cause) stud failure. Because Hitachi-GE have not considered common cause failure explicitly within GDA, it is required within licensing. The licensee shall provide a suitable defect tolerance demonstration for multiple stud failures with the depth of the demonstration informed by appropriate good practice, the fatigue usage factor, and the inspection/replacement policy.	4.4.1.3
AF-ABWR-SI-12	It is ONR's expectation that planar defects in major components should be rejected. Because Hitachi-GE have not provided sufficient confidence that this will be done, the	4.4.1.5

	licensee shall set appropriate inspection objectives for the end of manufacture Non-Destructive Testing to specify that all defects characterised as planar will be rejected.	
AF-ABWR-SI-13	The petal to bottom dome weld area of the reactor pressure vessel is an area of physically restricted access and the initial proposals for inspection of this area are not demonstrably adequate to support the Safety Case. Because of this, the licensee shall ensure that the inspection capability in this area is adequate to support the safety case.	4.4.1.11
AF-ABWR-SI-14	Because it is not apparent that the inspection conditions for very high integrity components reduce risks so far as is reasonably practicable, the licensee shall review all inspection conditions for all very high integrity welds to see where improvements could be made. This should include, but not be confined to, further investigations on the effect of cladding surface finish, the surface condition of the inspection surface and the locations of obstacles.	4.4.13
AF-ABWR-SI-15	The Main Steam Isolation Valve casings are very high integrity components; because the defect description proposed by Hitachi-GE are essential in defining whether repair welds will be necessary, confidence in these descriptions underpins confidence in the component. The licensee shall review and justify defect descriptions in the MSIV casing, including consideration of whether crack-like defects in the weld volume and heat affected zone should be considered.	4.4.15
AF-ABWR-SI-16	The inspectability of the Control Rod Drive penetrations provides a key part of their Safety Case. Because design for inspectability was not covered in sufficient detail during GDA, the licensee shall assess whether the inspectability of the Control Rod Drive penetrations is adequate to support the Safety Case, and whether it could be improved to reduce risks so far as is reasonably practicable.	4.4.2.3
AF-ABWR-SI-17	The RPV nozzles have been designed according to the ASME code, but explicit demonstration of design for inspectability has not been given. Because of this, the licensee shall demonstrate that the nozzles are designed such that they facilitate inspection to support the Safety Case.	4.4.3.4

AF-ABWR-SI-18	The RPV is a component of the highest reliability and as such all relevant international operational experience should be addressed when specifying it. During GDA, this area has developed and, hence, residual matters remain to be completed during licensing. The licensee shall justify the proposals put forward by Hitachi-GE in terms of: i) Materials properties ii) Irradiation embrittlement iii) Thermal embrittlement iv) Dissolved gas control. The licensee shall also demonstrate that these are achievable and can, in practice, be manufactured and meet the needs of the Safety Case throughout the thickness of the component.	4.4.4.3
AF-ABWR-SI-19	Hitachi-GE's GDA submissions have left the selection of fabrication processes for the Reactor Pressure Vessel to be selected during manufacture. Because welds, and the choice of fabrication technology affects plant risk, the licensee shall demonstrate that the fabrication processes selected for the reactor pressure vessel, including cladding processes, lower the plant risk so far as is reasonably practicable.	4.4.4.3
AF-ABWR-SI-20	Stress corrosion cracking susceptibility in austenitic stainless steel is linked to the levels of residual cold work, but this has not been considered by Hitachi-GE in sufficient detail during GDA. Therefore, the licensee shall justify the cold work levels for all Austenitic Stainless Steels and Nickel-Base Alloys as well as the methods utilised in the manufacture and construction of the UKABWR and apply appropriate remedial actions to reduce risks so far as is reasonably practicable.	4.4.6.3
AF-ABWR-SI-21	Product form decisions for portions of safety significant pipework at classifications below Very High Integrity have been made implicitly within RO-ABWR-0035. Because it has not been demonstrated within GDA that these product forms minimises the overall plant risk, the licensee shall justify that the choice of product form for these items lowers the plant risk so far as is reasonably practicable.	4.4.6.3

Annex 6

Minor Shortfalls

Minor Shortfall Number	Minor Shortfall Finding	Report Section Reference
MS-ABWR-SI-01	Supplier selection has played a role in defining the number of and length of welds in the RPV top head, with larger lengths of weld leading to possible higher numbers of defects. To ensure that supplier selection is not unduly influencing nuclear safety, the licensee should check to see if other credible suppliers are available, to manufacture plate for the RPV top head, who could supply material to a similar or higher level of safety.	4.2.5
MS-ABWR-SI-02	The licensee should ensure that relevant material property data to underpin the safety case for multiple RPV stud failure is included in a material property handbook.	4.4.1.3
MS-ABWR-SI-03	The licensee should ensure that consequences of multiple support failures for VHI components are classified and that an appropriate structural integrity case is developed.	4.4.1.3